GLOBAL Laser Enrichment

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Attn: Document Control Desk
Brian Smith, Chief
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Office of Nuclear Materials Safety & Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Subject: **REVISION 4 TO GLOBAL LASER ENRICHMENT LICENSE APPLICATION – PUBLIC VERSION**

Dear Mr. Frazier:

GE-Hitachi Global Laser Enrichment LLC (GLE) hereby submits revision 4 of the GLE License Application, chapters 1, 2, 3, 5, 8, and 11 (Enclosure 1). Non-Public version of the revised License Application has been prepared and will be submitted under separate enclosure.

If there are any questions regarding this letter and its contents, please do not hesitate to contact me at 910-819-4799 or at Julie.Olivier@ge.com.

Sincerely,

Julie A. Olivier

Julie Olivier
GLE Licensing Manager

NMSS22

Enclosures:

1. Revision 4 of the GLE License Application

Cc (without enclosures):
Tim Johnson (NRC)
Chris Monetta (GLE)
Jerry Head (GEH)
Ken Givens (GLE)
Lon Paulson (GEH)

CHAPTER 1
REVISION LOG

Rev.	Effective Date	Affected Pages	Revision Description
0	04/30/2009	ALL	Initial Application Submittal.
1	03/31/2010	8-9, 17, 19-29, 41, 43-45, 47, 55, 59, 61	Incorporate RAI responses submitted to the NRC via MFN-09-578 dated 09/04/2009 and MFN-09-801 dated 12/28/2009.
2	06/18/2010	14-18, 22, 26, 30, 22, 34, 38-41, 47, 49, 63	Revised Section 1.2.2.4 regarding nuclear liability insurance. Revised Section 1.1.3.1 regarding the transition period between construction and operations. Incorporated the latest natural phenomena information and updated the figures.
3	12/17/2010	23-25, 29, 32	Incorporate RAI responses from NRC letters dated October 5, 2010 and October 14, 2010. Revised Section 1.2.2.2 to update schedule for operations. Changed name President and CEO. Updated company assets to 2009 information Clarified that insurer determined amount of nuclear liability insurance Revised Section 1.2.5.6 to clarify definition of "basic component". Added description of procedure for inclement weather
4	03/30/2011	39, 48, 49	Changed two erroneous tornado "F3" references to "F2". Added revision number for referenced Reg Guide 1.76. Added revision number for referenced Reg Guide 1.198.

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FIGURES

1. GENERAL INFORMATION

This application requests a license from the U.S. Nuclear Regulatory Commission (NRC) to possess and use source material, special nuclear material (SNM), and byproduct material to construct and operate a commercial uranium enrichment facility. This application is filed by the GE-Hitachi Global Laser Enrichment LLC (GLE). GLE is requesting a license for a period of 40 years.

This chapter provides an overview of the GLE Commercial Facility. The facility enriches uranium for use in the manufacturing of nuclear fuel used in commercial power plants. This chapter provides a description of the facility and enrichment process along with a description of the GLE Site. Institutional information is provided to identify the applicant, describe the applicant's financial qualifications, and describe the proposed licensed activities.

This License Application (LA) is being submitted pursuant to the following:

- Atomic Energy Act of 1954, as amended *(Ref. 1-1)*,
- 10 CFR 70, *Domestic Licensing of Special Nuclear Material (Ref. 1-2)*,
- 10 CFR 40, *Domestic Licensing of Source Material (Ref. 1-3)*, and
- 10 CFR 30, *Rules of General Applicability to Domestic Licensing of Byproduct Material (Ref. 1-4)*.

1.1 FACILITY AND PROCESS DESCRIPTION

This section provides an overview of the GLE Site, the GLE Commercial Facility layout, and a summary of the GLE enrichment process.

1.1.1 Facility Location

The GLE Commercial Facility is located on an existing General Electric Company (GE) industrial site in Wilmington, North Carolina (herein referred to as the Wilmington Site). The Wilmington Site is a 1621-acre tract of land, located west of North Carolina Highway 133 (also known as Castle Hayne Road). The Wilmington Site lies between latitudes (North) 34° 19' 4.0"and 34° 20' 28.9" and longitudes (West) 77° 58' 16.4" and 77° 55' 19.8", and is approximately six (6) miles north of the City of Wilmington in New Hanover County, North Carolina (see Figure 1-1, *Wilmington Site and County Location*, and Figure 1-2, *Wilmington Site, New Hanover County, and Other Adjacent Counties*). The Wilmington Site is also the GLE "controlled area" (or "owner controlled area") for the purpose of meeting the requirements of 10 CFR 70.61(f), *Performance Requirements (Ref. 1-5)*.

The GLE Commercial Facility is located on approximately 100 acres of the Wilmington Site. In addition to the GLE Commercial Facility, the Wilmington Site contains the following GE facilities (see Figure 1-3, *Wilmington Site*):

- Global Nuclear Fuel – Americas, LLC (GNF-A) Fuel Manufacturing Operations (FMO) facility operated under the NRC SNM License-1097 *(Ref. 1-6)*;
- Wilmington Field Service Center (WFSC) in which used reactor control rod drive mechanisms are decontaminated, refurbished, and temporarily stored;
- GE Aircraft Engines (AE) facility which is not involved in nuclear fuel manufacturing operations;
- GE Services Components Operation (SCO) facility in which non-radioactive reactor components are manufactured;
- Fuel Components Operation (FCO) facility in which non-radioactive components for reactor fuel assemblies are manufactured; and
- Miscellaneous administrative and support buildings and site infrastructure, such as roads and parking lots.

To the east of the Wilmington Site border is North Carolina Highway 133 and some commercially and residentially developed properties. Located to the east of North Carolina Highway 133, is a GE-owned 24-acre parcel that is undeveloped, except for a GE employee park and a leased portion of property used as a transportation terminal. To the southwest of the Wilmington Site border is the Northeast Cape Fear River.

The majority of the north, northwest, and south perimeters are undeveloped forestlands. A small segment (approximately 1,000-feet of the north property line) borders the Wooden Shoe residential subdivision. A portion of the south property line is bordered by Interstate Highway 140 (otherwise known as the Wilmington Bypass). Residential properties are located directly south of the Wilmington Bypass.

The surrounding terrain is typical of coastal North Carolina with an elevation averaging less than 40 feet above mean sea level (msl). The terrain is characterized as gently rolling terrain consisting of forest, rivers, creeks, and swamps/marshlands.

1.1.2 Facility Description

The GLE Commercial Facility is shown on Figure 1-4, *GLE Commercial Facility Site Plan*. The GLE Commercial Facility includes the Operations Building where the enrichment processing systems and enrichment processing support systems are contained, several administrative and support buildings, a parking lot, retention basins, uranium hexafluoride (UF_6) cylinder pads, and connecting roadways. A cleared security buffer surrounds the entire GLE Commercial Facility and defines both the Restricted Area and the Protected Area of the facility. The major structures and areas of the facility are described below.

1.1.2.1 GLE Operations Building

The overall layout of the Operations Building is shown in Figure 1-4. The Operations Building includes the following process and support areas:

- Cylinder Shipping and Receiving Area,
- UF_6 Feed and Vaporization Area,
- Product Withdrawal Area,
- Tails Withdrawal Area,
- Cascade/Gas Handling Area,
- Blending Area,
- Sampling Area,
- Radioactive Waste Area,
- Heating, Ventilation, and Air Conditioning (HVAC) Equipment Area,
- Decontamination/Maintenance Area,
- Laboratory Area, and
- Laser Area.

The main process and support areas of the Operations Building and the associated operations are described below.

1.1.2.1.1 Cylinder Shipping and Receiving Area

The Cylinder Shipping and Receiving Area contains the necessary equipment to perform the following functions:

- Receive 30- and 48-inch cylinders from offsite;
- Weigh cylinders and perform other material control and radiological functions during receiving and when preparing for storage or offsite shipment;
- Provide interim storage of cylinders inside the Operations Building;
- Prepare cylinders and transfer them to onsite transfer vehicles (OSTVs) for transfer between the Operations Building and the UF_6 Cylinder Pads;
- Provide interim storage of product, feed, and sample/blend cylinders;

- Prepare cylinders and transfer them to OSTVs for transfer to other process areas within the Operations Building;

- Prepare product cylinders for offsite shipment and intra-site transfer; and

- Prepare 48-inch tails and heel cylinders for offsite shipment.

UF_6 feed is received at the GLE Commercial Facility in American National Standards Institute (ANSI) N14.1-compliant UF_6 cylinders on semi-trailer trucks, typically with one full 48-inch cylinder per shipping trailer. A compliant 48-inch feed cylinder contains a maximum of 12,501 kg of UF_6 *(Ref: 1-7)*.

When UF_6 cylinders are received at the GLE Commercial Facility, the cylinders are inspected, verified, and processed per approved written Operations, Security, and Radiation Protection (RP) procedures. Empty 30- and 48-inch cylinders are also received at the GLE Commercial Facility.

At the Cylinder Shipping and Receiving Area, cylinders are offloaded and transferred to an adjacent weighing and scanning area. After acceptance, feed cylinders are moved to an interim cylinder storage area inside the Cylinder Shipping and Receiving Area. From the interim cylinder storage area, feed cylinders may be moved to a feed station to begin processing, or to the In-Process Pad. An overhead bridge crane and transfer cart are used to handle the UF_6 cylinders.

Source material and SNM are used in this area.

1.1.2.1.2 UF_6 Feed and Vaporization Area

The UF_6 Feed and Vaporization Area contains the necessary equipment to perform the following operations:

- Receive UF_6 feed cylinders from the Cylinder Shipping and Receiving Area;

- Purge the light gases contained within the feed cylinders;

- Capture the light gases for disposal;

- Vaporize the UF_6 contained within the feed cylinders;

- Feed the vaporized UF_6 to the feed header between the Vaporization Area and the Cascade/Gas Handling Area within the Operations Building;

- Maintain design basis UF_6 feed rates to the feed header within the design basis temperature and pressure range; and

- Recover residual UF_6 from the feed cylinders to meet U.S. Department of Transportation (DOT) offsite cylinder shipping requirements for empty cylinders.

The UF_6 Feed and Vaporization Area is divided into feed vaporization chambers (FVCs). Each of the FVCs typically contains: solid feed stations (SFS) to vaporize the UF_6 feed; a cold trap purification station (CTPS) to remove light gases from the feed stream; a low temperature take-off station (LTTS) to remove feed cylinder UF_6 down to heel quantities; and a heated flow control valve box (HFCVB) for each SFS that contains the valves and pipe connections from each SFS.

Source material is used in this area.

1.1.2.1.3 Product Withdrawal Area

The Product Withdrawal Area contains the necessary equipment to perform the following functions:

- Receive empty 48 GLE UF_6 cylinders from interim storage within the Cylinder Shipping and Receipt Area;
- Maintain design basis UF_6 product withdrawal rates from the Cascade main discharge header;
- Separate the light gases from the UF_6 for disposal; and
- Provide filled 48 GLE cylinders with ≤ 8.00 wt% ^{235}U for interim storage and later disposition.

The Product Withdrawal Area contains: volume reducing compressor trains (VRCTs) that move UF_6 product material from the Cascade/Gas Handling System to the product Withdrawal Stations; LTTSs to collect the UF_6 product material; a CTPS to remove non-condensable light gases from the product stream; and a HFCVB for each LTTS that contains the valves and pipe connections from each LTTS.

SNM is used in this area.

1.1.2.1.4 Tail Withdrawal Area

The Tail Withdrawal Area contains the necessary equipment to perform the following functions:

- Receive empty UF_6 cylinders from interim storage within the Cylinder Shipping and Storage Area;
- Maintain design-basis UF_6 tails withdrawal rates from the enrichment system main discharge header;
- Separate the light gases from the UF_6 for disposal; and
- Provide filled UF_6 cylinders with ≤ 0.72 wt% ^{235}U for interim storage and later disposition.

The Tail Withdrawal Area contains: VRCTs that move UF_6 tails from the Cascade/Gas Handling System to the Tail Withdrawal Stations; LTTSs to collect the UF_6 tails material; a CTPS to remove non-condensable light gases from the tails stream; and a HFCVB for each LTTS that contains the valves and pipe connections from each LTTS.

Source material is used in this area.

1.1.2.1.5 Cascade/Gas Handling Area

The Cascade/Gas Handling Area contains the equipment necessary to perform the laser-based enrichment process. The UF_6 gas is exposed to laser-emitted light and two process streams are generated; one enriched in ^{235}U and one depleted in ^{235}U.

Technical details of the GLE laser-based enrichment process are proprietary, subject to export control by U.S. laws and regulations, and in many cases may also fall into the categories of security-related, safeguards, or classified information, access to which is further limited per U.S. laws and regulations.

Source material and SNM are used in this area.

1.1.2.1.6 Blending Area

The Blending Area contains the necessary equipment to perform the following functions:

- Receive 30- or 48-inch donor cylinders from interim storage within the Cylinder Shipping and Receiving Area;
- Purge the light gases contained within the cylinders;
- Capture the light gases for disposal;
- Vaporize the UF_6 contained within the donor cylinders;
- Feed the vaporized UF_6 to receiver cylinders;
- Recover residual UF_6 from the donor cylinders to meet DOT cylinder shipping requirements for empty cylinders; and
- Provide empty donor cylinders and filled receiver cylinders for interim storage.

The Blending Area contains blending donor stations (which are similar to the SFS) and blending receiver stations (which are similar to the product withdrawal LTTS) described under the Product Withdrawal Area above.

SNM is used in this area.

1.1.2.1.7 Sampling Area

The Sampling Area contains the necessary equipment to perform the following functions:

- Receive filled UF_6 cylinders from interim storage within the Cylinder Shipping and Receipt Area;
- Purge the light gases contained within the cylinders;
- Capture the reactive light gases for disposal and vent the nonreactive light gases;
- Homogenize and sample the UF_6 contained within the cylinders; and
- Maintain design basis UF_6 cylinder rates to support a six (6) million separative work unit (SWU) facility.

The function of the product liquid sampling system is to obtain an assay sample from filled product cylinders. The sample is used to validate the enrichment level of UF_6 in the filled product cylinders before the cylinders are sent to the fuel processor. This is the only system in the GLE Commercial Facility that converts solid UF_6 to liquid UF_6.

The Sampling Area contains: sample containment autoclaves (SCAs) to support liquefaction, sampling, and solidification of UF_6 in the cylinders; CTPS to remove light gases vented from the cylinders being sampled; LTTSs to capture UF_6 vented from the cylinders during sampling; HFCVB for each SCA that contains the valves and pipe connections between units within the sampling area; an autoclave surge tank (AST) that provides UF_6 surge capacity if an autoclave relief device actuates.

Source material and SNM are used in this area.

1.1.2.1.8 Liquid and Solid Radioactive Waste Areas

Quantities of radiologically contaminated, potentially contaminated, and non-contaminated aqueous liquid effluents are generated in a variety of the GLE Commercial Facility operations and processes. Aqueous liquid effluents are collected in tanks located in the Radioactive Liquid Effluent Collection and Treatment Room. The collected effluent is sampled and analyzed to determine if treatment is required before release.

Operation of the GLE Commercial Facility also generates refuse and other hazardous and nonhazardous solid wastes. These wastes may be designated as Resource Conservation and Recovery Act (RCRA) hazardous wastes, low-level radioactive waste (LLRW), high-activity waste, or low-level mixed waste (LLMW). Solid-waste systems are designed to process both wet and dry low-level radioactive solid waste. Solid radioactive waste material is accumulated, monitored for criticality control and other regulatory requirements, stored in temporary accumulation areas, and then transferred to one of the solid-waste storage buildings located on the GLE Site for storage pending eventual offsite shipment/disposition.

1.1.2.1.9 HVAC Equipment Areas

Various ventilation systems are used to condition the environment inside the buildings and areas to meet requirements for personnel, process equipment, and supporting systems and utilities. The HVAC systems also control the room pressure in different areas or zones of the buildings relative to adjacent areas and relative to the outdoors as part of the radioactive or hazardous material containment function.

The ventilation system requirements of each area are dependent on the process performed, and on variables such as the indoor air temperature, relative humidity, relative room pressure, and safety requirements.

Ventilation systems that have the potential to exhaust radioactive or hazardous materials interface with the Monitored Central Exhaust System (MCES). The MCES functions to remove uranium particulates as well as UF_6 and HF gas from process gas streams and room air during normal and abnormal events. The system maintains areas under negative pressure relative to ambient and adjacent areas. This prevents the release of radioactive or hazardous materials, which protects workers and the public. The MCES discharges through a monitored exhaust stack located in the Operations Building.

The ventilation and MCES equipment serving the Operations Building is located in various locations throughout the Operations Building.

1.1.2.1.10 Decontamination/Maintenance Area

The Decontamination/Maintenance Area provides a place for personnel to remove contamination from, and make repairs to, equipment and process components used in UF_6 systems, waste handling systems, and other areas of the facility.

Source material and SNM are contained in this area.

1.1.2.1.11 Laboratory Area

The Laboratory Area is located just north of the Cylinder Shipping and Receiving Area, on the east side of the Operations Building. Within the Laboratory Area there are areas for mass spectroscopy equipment, wet chemistry activities, safety and regulatory testing and analysis, standard analytical laboratory equipment, and fume collection and exhaust hoods.

Source material and SNM are used in this area.

1.1.2.1.12 Laser Area

The Laser Area contains the necessary equipment to operate the laser systems that are part of the GLE laser-based enrichment technology; and produce the specific wavelength of light required to affect the uranium isotope necessary for the enrichment process.

The Laser Area contains: lasers to generate the required wavelength of light needed for the enrichment process, and a laser repair shop located adjacent to the Laser Area to perform maintenance on the laser systems, including calibration, repair, and preventive maintenance.

No source material or SNM is used in this area.

1.1.2.2 UF_6 Cylinder Pads

The UF_6 Cylinder Pads include three outdoor cylinder pads each serving a different function. The three pads are described below. See Figure 1-4 for the location of the UF_6 Cylinder Pads.

1.1.2.2.1 Product Pad

The Product Pad is used to store product in 30-inch cylinders. The Product Pad is approximately 48,000 square feet and constructed similar to the other storage pads to provide for rainwater drainage. Saddles are used to store the cylinders and the cylinders are not typically stacked.

SNM is contained in this area.

1.1.2.2.2 In-Process Pad

The In-Process Pad is used to store feed material, as well as any cylinders containing heels and empty cylinders. It is approximately 130,000 square feet and constructed similar to the other pads to provide for rainwater drainage. Saddles are used to store the cylinders and the cylinders are not typically stacked.

Source material is contained in this area.

1.1.2.2.3 Tails Pad

The Tails Pad is designed to provide storage for 48-inch cylinders containing less than or equal to 0.72 percent weight ^{235}U. The Tails Pad is sized to accommodate the cylinders resulting from ten (10) years of facility operation.

The Tails Pad occupies approximately 465,000 square feet. The pad is sloped to provide drainage to the edges of the pad. The surrounding site is graded to provide collection and drainage of rainwater to an onsite retention basin. The cylinders may be stacked two high and are stored using saddles.

Source material is contained in this area.

1.1.2.3 Other Facility Buildings and Supporting Infrastructure

See Figure 1-4 for the location of the following buildings and supporting infrastructure.

There are three (3) administrative buildings. Two of the administrative buildings primarily contain office space for the GLE support staff and conference rooms. The third administrative building, the Operations Support Center, contains the personnel Entry Control Facility (ECF) and is located at the entrance to the Protected Area. Personnel requiring access to the Protected Area must pass through the ECF. The ECF is designed to facilitate and control the passage of authorized facility personnel and visitors. General parking is located outside of the Protected Area.

Waste storage buildings are used to store solid LLRW. The waste is packaged in transportation containers and surveyed prior to being stored in the warehouse.

An electrical substation and diesel generators provide electrical power to the GLE Commercial Facility. The diesel generators are used during short-term power losses to support an orderly shutdown of the enrichment processes upon loss of power or until normal electrical service is restored. A loss of GLE Site electrical power does not have any public safety implications.

Potable and process water supply lines run to the GLE Commercial Facility from the existing Wilmington Site water supply infrastructure. Sanitary waste, process wastewater, and treated liquid radiological wastewater are routed from the GLE Commercial Facility via underground lines to lift stations. The lift stations deliver the respective wastewaters to the existing Wilmington Site Sanitary Waste Water Treatment Facility (WWTF) and Final Process Lagoon Treatment Facility (FPLTF) through underground pipes.

Two retention basins receive stormwater runoff from the GLE Commercial Facility. The majority of the runoff from the GLE Commercial Facility, including the Operations Building, drains to a collection basin on the Wilmington Site. The remaining runoff, including runoff from the UF_6 Cylinder Pads, drains to a GLE Site retention basin.

There is a water tower, a firewater retention basin, and associated pumps and piping located on the GLE Site. The water in the tower is designated for process water, but has a reserved level for fire fighting. The firewater retention basin and associated diesel powered firewater pumps are designed as a backup source for fire protection systems.

The road leading to the entrance of the GLE Commercial Facility is located off of Castle Hayne Road (see Figure 1-3). There is also a road exiting the GLE Commercial Facility leading to the GNF-A FMO Facility. Both of these roads are located on the Wilmington Site and are maintained by GE.

1.1.3 Process Description

This section provides an overview of the GLE laser-based enrichment process. A more detailed description of the process is provided in the Integrated Safety Analysis (ISA) Summary. The ISA Summary also contains a description of the other systems supporting the GLE Commercial Facility including the utility systems; HVAC systems, process water system, and the various cylinder-handling systems used to move UF_6 cylinders.

1.1.3.1 Process Overview

The GLE Commercial Facility is a uranium enrichment facility that utilizes laser-based enrichment technology. The GLE Commercial Facility is designed to separate a feed stream containing the naturally occurring proportions of uranium isotopes into a product stream (enriched in the ^{235}U isotope) and a tails stream (depleted in the ^{235}U isotope).

The GLE Commercial Facility utilizes industry standard UF_6 containers and processes for material handling aspects of enrichment facility operations similar to those utilized at other uranium enrichment facilities. These similar UF_6 handling processes include the movement of uranium feed stock from its solid UF_6 form in cylinders to gaseous form used in the enrichment cascade via vaporization techniques, the filling of UF_6 cylinders with UF_6 gas condensed into solid UF_6 form after the enrichment process, and the blending of UF_6 gas of different enrichments to create specific desired product enrichments.

The GLE Commercial Facility uses the laser-based enrichment technology within an area of the facility known as the Cascade/Gas Handling Area. The process enriches natural UF_6, containing approximately 0.72 weight percent ^{235}U, to a UF_6 product containing ^{235}U enriched up to 8 weight percent. The nominal capacity of the facility is six (6) million SWU per year.

The uranium enrichment process utilized by the GLE Commercial Facility utilizes lasers tuned to specific frequencies to selectively excite UF_6 gas molecules to enable separation of the ^{235}U isotope in UF_6 feed stock. The result is a UF_6 product stream enriched in the ^{235}U isotope and a UF_6 tails stream in which the fraction of ^{235}U isotope is reduced or depleted. Technical details of the GLE laser-based enrichment technology are proprietary, subject to export controls by U.S. laws and regulations, and in many cases also fall into the categories of security-related, safeguards, or classified information, access to which is further limited per U.S. laws and regulations.

The phases of construction/initial operations include Early Construction, Phase 1 Construction (Initial Construction of one MSWU facility), Phase 2 Construction (Construction and Component Installation to Ramp up to six MSWU), and Full Operations at six MSWU. The facility described in this License Application assumes that the facility is operating at six MSWU. However, the facility will be operating at approximately one MSWU during the first year, two MSWU during the second year, three MSWU during the third year, four MSWU during the fourth year, five MSWU during the fifth year, and six MSWU during the sixth year and every year thereafter. The initial construction plan includes building the Operations Building in its entirety, and equipping it with the necessary equipment to generate one MSWU. During the first year, while the facility is operating at one MSWU, equipment/component installation will be occurring simultaneously. Similarly, for the second, third, fourth, and fifth years, operations and equipment/component installation will occur simultaneously.

During Early Construction (prior to receipt of an NRC license), the following activities will occur:

- Clearing 100 acres on the GLE Site,
- Site grading and erosion control,
- Installation of stormwater retention system,
- Construction of main access roadways,
- Placement of utilities (electricity, potable water, process water, water for fire suppression, sanitary sewer, natural gas), and
- Construction of parking lots and minor roadways.

During Phase 1 Construction, the following activities will occur:

- Construction of the Operations Building,
- Construction of the UF_6 Cylinder Pads,
- Construction of the guardhouses,
- Construction of ancillary buildings (includes waste storage facilities, vehicle maintenance facilities, warehouses, storage yards, utility buildings, etc.),
- Installation of security systems,
- Construction of the Administrative buildings,
- Installation of the fire protection and other safety systems, and
- Installation of components within Operations Building to support one MSWU production.

{{{Proprietary Information withheld from disclosure per 10 CFR 2.390}}}

During full operations at six MSWU, there is not anticipated to be further facility construction or component installation, with the exception of maintenance and repair activities. Any unanticipated construction/component installation will be evaluated per the 10 CFR 70.72 *Facility Changes and Change Process (Ref. 1-8)* to determine if an amendment to the license is required prior to initiating the activities.

1.1.3.2 *Process System Descriptions*

The GLE Commercial Facility enrichment process consists of the following four (4) major systems and two enrichment support systems:

Major Enrichment Process Systems

1. UF_6 Feed and Vaporization
2. Cascade/Gas Handling
3. Product Withdrawal
4. Tail Withdrawal

Enrichment Support Systems

1. Blending
2. Sampling

An overview of each process system or support system is provided below.

1.1.3.2.1 *UF_6 Feed and Vaporization System*

The major function of the UF_6 Feed Vaporization System is to provide a continuous supply of gaseous UF_6 from the feed cylinders to the Cascades. The nominal UF_6 feed flow rate is based on a six (6) million SWU/year facility capacity. Approximately 900 48-inch cylinders are processed annually.

The major equipment used in the UF_6 Feed Vaporization Process are the SFSs. Feed cylinders are loaded into SFSs; vented for removal of light gases, primarily air and hydrogen fluoride, and heated to sublime the UF_6. The light gases and UF_6 gas generated during feed purification are routed to the Feed Purification Subsystem where the UF_6 is de-sublimed. The Feed Purification Subsystem consists of UF_6 cold traps, a vacuum pump/chemical trap set, and a LTTS. The Feed Purification Subsystem removes any light gases such as air and hydrogen fluoride from UF_6 prior to introduction into the Cascade/Gas Handling Area. The UF_6 is captured in UF_6 cold traps and ultimately recycled as feed, while hydrogen fluoride is captured on chemical traps.

1.1.3.2.2 *Cascade/Gas Handling System*

After purification, UF_6 from the SFS is routed to the Cascade/Gas Handling Area. The gas is exposed to laser-emitted light, and the UF_6 gas is separated into two streams, one enriched in ^{235}U and one depleted in ^{235}U.

1.1.3.2.3 *Product Withdrawal System*

Enriched UF_6 from the Cascade/Gas Handling Area is de-sublimed in the Product Withdrawal LTTS. Pumps and compressors transport the UF_6 from the Cascade/Gas Handling Area to the Product Withdrawal LTTS. The heat of de-sublimation of the UF_6 is removed by cooling air routed through the LTTS. Filling of the product cylinders is monitored with a load cell system, and filled cylinders are transferred to the Product Cylinder Sampling System for sampling.

1.1.3.2.4 Tail Withdrawal System

Depleted UF_6 from the Cascade/Gas Handling Area is de-sublimed in the Tail Withdrawal LTTS. Pumps and compressors transport the UF_6 from the Cascade/Gas Handling Area to the Product LTTS. The heat of de-sublimation of the UF_6 is removed by cooling air routed through the LTTS. Filling of the tail cylinders is monitored with a load cell system, and filled cylinders are transferred to the Tails Pad.

1.1.3.2.5 Blending System

The primary function of the Blending System is to blend UF_6 donor cylinders with differing enrichments into a receiver cylinder. The assay in the receiver cylinder is one that meets customer specifications as well as transportation standards.

1.1.3.2.6 Sampling System

UF_6 sampling operations are performed in the Sampling Area. Current American Society for Testing and Materials (ASTM) International standards require that UF_6 samples be taken from homogenized UF_6. Therefore, the design criteria require liquefaction of UF_6 during sampling operations. In addition, sampling of a statistical basis set of feed and tails cylinders is required to support Material Control and Accounting (MC&A) requirements.

Autoclaves with heating and cooling capability are used to liquefy UF_6 in the cylinders, homogenize the liquefied material, obtain a representative sample of the contents of the cylinders, and then solidify the UF_6 in the cylinders before they are removed from the autoclave. The cylinders may be any approved UF_6 cylinder, per ANSI N14.1, *Nuclear Materials – Uranium Hexafluoride – Packaging for Transport (Ref. 1-9)*, which meets nuclear criticality safety (NCS) requirements. The autoclaves are designed to contain a UF_6 release in the autoclave. Electrically heated air is the heating medium and cold air is used for cooling.

1.1.4 Waste Management

1.1.4.1 Solid Wastes

Operation of the GLE Commercial Facility generates refuse and other nonhazardous solid waste, wastes designated as RCRA hazardous wastes, and LLRWs. No high-level radioactive wastes are generated by GLE Commercial Facility operations. GLE does not intend to generate mixed wastes. Low-level waste is expected to be Class A waste. The types, sources, and estimated quantities of solid wastes generated by GLE Commercial Facility operations are summarized in Table 1-1, *Typical Types, Sources, Quantities of Solid Wastes Generated by GLE Commercial Facility Operations*, and Table 1-2, *Management of Solid Wastes*.

GLE Commercial Facility operations generate an estimated 380 tons of municipal solid waste (MSW) per year. This waste is collected and placed in roll-off type containers. A commercial refuse collection service regularly collects the filled containers and transports the waste to a RCRA permitted Subtitle D landfill for disposal.

In addition to MSW, an estimated 107 tons of non-hazardous solid wastes are generated per year as a result of equipment maintenance for GLE Commercial Facility operations. Examples of these wastes are spent coolant and used filter media. These wastes are collected and temporarily stored in containers appropriate for the waste type. Depending on the composition of the non-hazardous waste, these materials are either shipped directly to a permitted RCRA Subpart D landfill for treatment and burial, or routed to other approved facilities for reuse, reclamation, or treatment.

The GLE Commercial Facility generates approximately 12 tons of RCRA hazardous waste per year. This waste is collected, packaged in DOT-approved shipping containers, and temporarily stored onsite for shipment to a RCRA-permitted Subtitle C treatment, storage, and disposal facility.

The sources and typical quantities of LLRW generated by GLE Commercial Facility operations are summarized in Table 1-1. LLRW is collected in containers appropriate for the waste form and shipped by truck to an approved disposal facility as indicated in Table 1-2.

1.1.4.2 Liquid Wastes

The sources and estimated quantities of wastewater generated by GLE Commercial Facility operations are summarized in Table 1-3, *Typical Types, Sources, and Quantities of Wastewater Generated by GLE Commercial Facility Operations*, and Table 1-4, *Management of Wastewater Generated by GLE Commercial Facility Operations*.

The liquid radioactive wastes generated in the Operations Building are collected in closed drain systems that discharge to an accumulator tank. The liquid is treated to remove uranium through precipitation; the liquid is then treated to remove fluoride through evaporation. The resulting solids are dried and disposed of as LLRW.

The treated wastewaters from the Radiological Liquid Effluent Treatment System (RLETS) are discharged to the existing Wilmington Site Sanitary WWTF and FPLTF. The FPLTF receives Wilmington Site process wastewater, including the treated effluent from the GNF-A Radiological Waste Treatment System. The treated effluent from the FPLTF is discharged via National Pollutant Discharge Elimination System (NPDES)-permitted Outfall 001 to the Wilmington Site effluent channel where it is combined with stormwater, discharging groundwater, and treated sanitary wastewater effluent. The effluent channel flows to the unnamed Tributary No. 1 to the Northeast Cape Fear River.

The cooling tower for the GLE Commercial Facility is a closed loop system that does not contact any uranium materials or uranium-contaminated wastewater streams. To minimize the amount of dissolved solids and other impurities in the circulating water, standard operating practice is to regularly remove a portion of the circulating water from the cooling tower loop and discharge the water to an evaporation pond (adding fresh water to the cooling tower loop to make up for corresponding water loss). Approximately 30,000 gallons per day (gpd) is removed and pumped directly to the existing Wilmington Site FPLTF.

Operation of the GLE Commercial Facility generates approximately 10,500 gpd of sanitary waste. The sanitary wastes are collected in a sewer system connected to the existing Wilmington Site Sanitary WWTF. This facility uses an Activated Sludge Aeration Process. The treated effluent from the Wilmington Site Sanitary WWTF is reused as process water.

Stormwater runoff from outdoor impervious surfaces within the GLE Commercial Facility is collected in drainage conduits and channels flowing into retention basins used for collection of runoff. The retention basins are routed to the unnamed Tributary No. 1, which flows into the Northeast Cape Fear River.

1.1.5 Depleted Uranium Management

Depleted uranium (also referred to as UF_6 tails) from GLE Commercial Facility operations is temporarily stored at the GLE Commercial Facility in 48-inch cylinders before being shipped offsite to a depleted uranium conversion facility. There is no onsite disposal of the UF_6 tails at the Wilmington Site. Section 3113 of the United States Enrichment Corporation (USEC) Privatization Act *(Ref. 1-10)* directs the U.S. Department of Energy (DOE) to "accept for disposal" depleted uranium, such as the UF_6 tails generated by the GLE Commercial Facility.

The Tails Pad is designed to provide storage capacity for approximately 9,000 48-inch cylinders, which is equivalent to ten years of facility operation. It is anticipated that DOE will have begun accepting possession of the UF_6 tails before the storage pad capacity is reached. The pad design layout permits double stacking of the 48-inch cylinders and allows the cylinders to be moved with gantry cranes and flatbed trucks. The storage pad occupies approximately 465,000 square feet. To provide stormwater drainage, the pad is sloped at the edges. The terrain surrounding the storage pad is graded to provide collection and drainage of stormwater to a retention basin.

Saddles are used to stack and store the cylinders above the Tails Pad surface. To transfer the UF_6 tails between the Cylinder Shipping and Receiving Area and the Tails Pad, dedicated diesel-powered flatbed trucks are used. At the Tails Pad, a diesel-powered, self-propelled gantry crane is used to unload the cylinder from the flatbed truck, move the cylinder to the appropriate storage location on the pad, and place the cylinder on its pad cradle. Work practices to manage the Tails Pad include periodic inspections and radiological surveys to ensure cylinder integrity. Operators are trained in safe cylinder handling and cylinder maintenance procedures.

1.1.6 Liquid and Air Effluents

1.1.6.1 Process Wastewaters

Uranium enrichment operations performed inside the Operations Building generate process wastewater from decontamination, cleaning wash water, and laboratory wastes. The waste streams contain small concentrations of uranium and are collectively referred herein as liquid radioactive waste. Liquid radioactive waste is treated to remove uranium and fluoride as described in Section 1.1.4, *Waste Management*.

The treated wastewaters from the RLETS are discharged to the existing Wilmington Site FPLTF. This facility currently receives Wilmington Site process wastewater, including the treated effluent from the GNF-A FMO Facility Radiological Waste Treatment System. The treated effluent from the FPLTF is discharged via NPDES-permitted Outfall 001 to the Wilmington Site effluent channel where it is combined with stormwater, discharging groundwater, and treated sanitary wastewater effluent. The effluent channel flows to the unnamed Tributary No. 1 to the Northeast Cape Fear River. The liquid leaving RLETS is monitored to ensure compliance with the 10 CFR 20, Appendix B, *Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage (Ref. 1-11),* limit. In addition, the liquid leaving the RLETS system is monitored to ensure compliance with the NDPES permit levels for fluoride, as well as other constituents specified in the permit. Other constituents may include total suspended solids, biological oxygen demand, oil and grease, total nitrogen, dissolved oxygen, and pH.

1.1.6.2 Air Effluents

The laser-based enrichment process is a closed process with no vents needed for routine venting of process gases. Some short-term gaseous releases occur inside the Operations Building during activities associated with operations such as the connection/ disconnection of UF_6 cylinders to process equipment and process equipment maintenance activities. These gaseous releases are routed through the building's ventilation system. The ventilation system air stream passes through a series of emissions-control devices consisting of high-efficiency particulate air (HEPA) filters and high-efficiency gas absorption (HEGA) filters. The exhaust air stream from these emission controls is vented to the atmosphere and monitored at the stack for uranium and fluoride. Table 1-5, *Typical GLE Air Emissions*, shows the typical air effluent concentrations from the Operations Building and the required regulatory limits. GLE shall comply with the requirements in 10 CFR 20, Appendix B, for uranium air effluents, and with the requirements specified in the North Carolina Department of Air Quality permit for monitoring of fluorides (as well as other operational controls/conditions specified in the permit).

1.1.7 Raw Materials, By-Products, Wastes, and Finished Products

The raw materials used in the laser-based enrichment process include UF_6 feed, gases used to support laser operation, oils used to support mechanical operations, process water, and solvents used in cleaning equipment. The by-product of the laser-based enrichment process is depleted uranium tails in the form of solid UF_6. The wastes from the laser-based enrichment process include solid wastes, process wastewaters, and air effluents. Further description of these wastes is contained in Section 1.1.4. The finished product from the laser-based enrichment process is solid UF_6 enriched in ^{235}U. GLE will not use or possess any moderator or reflector with special characteristics, such as beryllium or graphite.

GLE utilizes commercial natural UF_6 feed stock meeting the requirements of ASTM C787-06, *Standard Specification for Uranium Hexafluoride for Enrichment (Ref. 1-12)*. At this time, GLE does not intend to use "reprocessed UF_6" as feed stock, and consistent with ASTM C787-06, GLE requires that suppliers possessing feed cylinders contaminated with reprocessed UF_6 feedstock provide additional evidence of uranium purity that is backed up by statistical sampling of feed stock at GLE. As such, impurities in the feed are expected to be consistent with, or less than, those quantities specified in this standard. GLE shall produce enriched uranium meeting the requirements of ASTM C996-04, *Standard Specification for Uranium Hexafluoride Enriched to Less than 5 % ^{235}U (Ref. 1-13)*, for enriched commercial grade UF_6 and any additional customer specifications.

1.2 INSTITUTIONAL INFORMATION

This section describes the corporate identity, financial qualifications, type of license, and the requested special authorizations and exemptions.

1.2.1 Corporate Identity

The applicant name and address, corporate structure and ownership control, and physical location of the facility are provided below.

1.2.1.1 Applicant Name and Address

This application for an NRC license is filed by GE-Hitachi Global Laser Enrichment LLC. GLE is headquartered in Wilmington, North Carolina.

The full address of the applicant is as follows:

<u>Mailing Address</u>:

Global Laser Enrichment
P.O. Box 780, Wilmington, North Carolina 28402

<u>Physical Address</u>:

Global Laser Enrichment
3901 Castle Hayne Road, Wilmington, North Carolina 28401

1.2.1.2 Organization and Management of Applicant

The corporate ownership structure is shown in Figure 1-5, *GLE Ownership*. GLE is a Delaware limited liability company and currently the only subsidiary of majority owner GE-Hitachi Nuclear Energy Americas LLC (GEH), a global supplier of nuclear energy-related equipment and services, and which is itself a Delaware limited liability company and a wholly-owned subsidiary of GE-Hitachi Nuclear Energy Holdings LLC (Holdings). Holdings, a Delaware limited liability company, is a subsidiary of majority owner GENE Holding LLC (GENE), which is a Delaware limited liability company wholly owned by General Electric Company (GE), a U.S. corporation organized under the laws of the State of New York, and of minority owner Hitachi

America, Ltd., which is a wholly owned subsidiary of Hitachi Ltd., a Japanese corporation. GLE also has two minority owners, Cameco Enrichment Holdings, LLC ("Cameco Enrichment"), with 24% ownership interest in GLE, and GENE, which owns 13.5% of GLE. Cameco Enrichment is a Delaware limited liability company wholly owned by Cameco US Holdings, Inc., a Nevada corporation, which is in turn wholly owned by Cameco Corporation, a Canadian corporation.

In this ownership structure, GE maintains an indirect majority, that is 51% ownership, controlling interest, and no foreign entity has the ability to exercise control over GLE operations and management or has access to, or use rights in, GLE's nonpublic enrichment technology, including classified information. GLE Governing Board resolutions and, as applicable, Governing Board member voting proxies are utilized to assure that only Governing Board members who are U.S. citizens with appropriate U.S. government clearances have access to, or exercise control over activities affecting the protection of, classified information. Foreign ownership, control, and influence (FOCI) information is initially submitted, and periodic updates thereto are provided, to the NRC in accordance with 10 CFR 95, *Facility Security Clearance and Safeguards of National Security Information and Restricted Data (Ref. 1-14)*.

The current principal officers of GLE and their citizenship are listed below:

- Chris Monetta, President and Chief Executive Officer — United States
- Craig M. Steven, Chief Financial Officer — United States
- Harold J. Neems, Secretary and General Counsel — United States

GLE's immediate parent, GEH, is the parent company of NRC licensees that are licensed under 10 CFR 50, *Domestic Licensing of Production and Utilization Facilities (Ref. 1-15)*, 10 CFR 70, and 10 CFR 72, *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor Related Greater Than Class C Waste (Ref. 1-16)*, at facilities in Sunol, California and Morris, Illinois. GLE's affiliate, GNF-A, also a controlled subsidiary of GE, is the current holder of an NRC license under 10 CFR 70 for an existing facility on the Wilmington Site.

1.2.1.3 Address of Facility and Site Location Description

The address of the facility is the same as the physical address of the applicant. A description of the facility site location is provided in Section 1.1.1, Facility Location.

1.2.2 Financial Qualifications

1.2.2.1 Capital Cost Estimate

GLE estimates that the total capital investment required to construct a six million SWU facility is approximately **{{{Proprietary Information withheld from disclosure per 10 CFR 2.390}}}** (in 2009 dollars), excluding capital depreciation, UF_6 tails disposition, decommissioning and any replacement equipment required during the life of the facility. The basis for the cost estimate is provided in Table 1-6, *GLE Commercial Facility Capital Cost Estimate*.

The cost estimate is based on a phased construction approach that is expected to take approximately eight (8) years from the time the license is issued to reach the full six (6) million SWU capacity. The first phase of the GLE Commercial Facility will be a one (1) million SWU facility, followed by incremental addition of 1 million SWU per year, starting one year after the initial 1 million SWU begins operating. GLE is expected to start production on the initial 1 million SWU facility approximately three (3) years from the issuance of the NRC license that GLE is seeking through this application. The ramp u phase (from 1 to 6 million SWU) is expected to leverage efficiencies gained from the initial deployments to expedite the construction process and increase the SWU capacity that can be deployed at one time..

1.2.2.2 Funding Commitments

Construction of the initial 1 million SWU facility shall not commence before funding is fully committed. Of this full funding (equity and/or debt), GLE will have: (1) minimum equity contributions of 30% of project costs from the parents and affiliates of the partners; and (2) firm commitments ensuring funds for the remaining project costs. The construction of the ramp up phase will have the same requirements listed for the first phase, except, that expected profits from sales may be used as a funding source.

GLE shall not proceed with the project unless it has in place long-term conditional enrichment contracts (that is, five (5) years or longer) with price expectations sufficient to cover operating costs (including facility depreciation and decommissioning), with a return on investment.

The foregoing funding commitments, which will be in place prior to GLE Commercial Facility construction and operation, as applicable, are consistent with the license condition approved by the NRC in previous uranium enrichment facility licensing proceedings. See CLI-97-15, 46 NRC 294, 309 (1997) (Claiborne Enrichment Center); CLI-04-3, 59 NRC 10, 23 (2004) (National Enrichment Facility); and CLI-04-30, 60 NRC 426, 437 (2004) (American Centrifuge Plant).

GLE LA Chapter 10, *Decommissioning,* describes how reasonable assurance is provided that funds will be available to decommission the facility as required by 10 CFR 70.22(a)(9), *Contents of Applications (Ref. 1-17)*, 10 CFR 70.25, *Financial Assurance and Recordkeeping for Decommissioning (Ref. 1-18)*, and 10 CFR 40.36, *Financial Assurance and Recordkeeping for Decommissioning (Ref. 1-19).*

1.2.2.3 Financial Resources

GLE is currently funded by three parent companies, General Electric, Hitachi, and Cameco. The parent organizations have contributed cash and notes to fund the project through the design validation stage of the program and stand committed to provide additional funding pending the successful validation of the design concept. GLE currently expects to fund the construction costs through additional equity contributions provided by the parent companies. However, GLE may explore other funding options including, but not limited to additional equity owners (pending approval of the current parent companies) or long-term debt instruments.

A summary of the parent companies' total assets and net income for 2009 are provided below. All three of the parent organizations are publicly traded and additional information, including annual reports, are available on the companies' respective websites.

For the year ending December 31, 2009, GE had total assets (U.S. Dollars) of $781,818,000,000, with cash assets of $72,260,000,000. GE's net income in 2009 was $10,725,000,000.

For the year ending December 31, 2009, Hitachi had total assets (Japanese Yen) of JPY9,403,709,000,000, with cash assets of JPY807,926,000,000. Hitachi had a net loss in 2009 of JPY787,337,000,000.

For the year ending December 31, 2009, Cameco had total assets (Canadian Dollars) of C$7,342,102,000, with cash assets of C$1,101,229,000. Cameco's net income in 2009 was C$1,099,422,000.

1.2.2.4 Liability Insurance

GLE shall, in accordance with 10 CFR 140.13b, *Amount of Liability Insurance Required for Uranium Enrichment Facilities (Ref. 1-20)*, and prior to and throughout operation of the GLE Commercial Facility, have and maintain nuclear liability insurance in the amount of up to $200 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 material or SNM. The amount of $200 million was determined by the insurer (American Nuclear Insurers).

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

- An effective facility form (non-indemnified facility) policy of nuclear energy liability insurance from nuclear facility underwriters;
- Such other type of nuclear energy liability insurance as the NRC may approve; or
- A combination of the foregoing.

GLE will provide proof of insurance to the NRC no later than October 15, 2010.

1.2.3 Type, Quantity, and Form of Licensed Material

GLE proposes to acquire, deliver, receive, possess, produce, use, transfer, and/or store source material and SNM meeting the criteria of SNM of low strategic significance as described in 10 CFR 70.4, *Definitions (Ref. 1-20)*. Details of the SNM are provided in Table 1-7, *Type, Quantity, and Form of Licensed Special Nuclear Material*. It is anticipated that other source and by-product materials will be used for instrument calibration purposes. These materials will be identified during subsequent design phases and the LA will be revised, as necessary.

GLE utilizes commercial natural UF_6 feed stock meeting the requirements of ASTM C787-06. At this time, GLE does not intend to use "reprocessed UF_6" as feed stock, and consistent with ASTM C787-06, GLE shall require that suppliers possessing feed cylinders contaminated with reprocessed UF_6 feed stock provide additional evidence of uranium purity that is backed up by statistical sampling of feed stock at GLE. As such, GLE expects to possess only trace amounts of other radionuclides consistent with the natural decay of uranium.

1.2.4 Requested Licenses and Authorized Uses

GLE is engaged in the production and sale of uranium enrichment services to electric utilities or fuel fabrication facilities for the purpose of manufacturing fuel to be used to produce electricity in commercial nuclear power plants. GLE also may purchase and enrich uranium for direct sale to fuel fabrication facilities. In addition, GLE may provide enrichment services for the U.S. government under certain contractual agreements.

This GLE LA is necessary for licenses issued under 10 CFR 30, 10 CFR 40, and 10 CFR 70 to construct, own, use, and operate facilities described herein as an integral part of the GLE Commercial Facility. This includes licenses for byproduct material, source material, and SNM. The license requested is for a 40 year period. See Section 1.1, *Facility and Process Description*, for a summary description of the GLE activities.

1.2.5 Special Authorizations and Exemptions

1.2.5.1 Authorized Guidelines for Contamination-Free Articles

GLE requests authorization to use the guidelines, contamination, and exposure rate limits developed by the NRC and included as Appendix A of this chapter titled *Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material,* for decontamination and survey of surfaces or premises and equipment prior to abandonment or release for unrestricted use. These guidelines are included as a regulatory acceptance criterion in NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility (Ref. 1-22).*

1.2.5.2 Exemption to Posting Requirements

GLE requests authorization to post areas within Radiological Controlled Areas (RCAs) in which radioactive materials are processed, used, or stored with a sign stating "Every container in this area may contain radioactive material," in lieu of the labeling requirements in 10 CFR 20.1904, *Labeling Requirements (Ref. 1-23).*

The requested exemption is authorized by law because there is no statutory prohibition on the proposed posting of a single sign indicating that every container in the posted area has the potential for internal contamination. Indeed, to reduce unnecessary regulatory burden, the NRC issued a final rule in 2007 that, in part, modified 10 CFR 20.1905, *Exemptions to Labeling Requirements (Ref. 1-24)*, thereby exempting certain containers holding licensed material from the labeling requirements of 10 CFR 20.1904 if certain conditions are met. Although the 2007 rulemaking only applied to facilities licensed under 10 CFR 50 and 10 CFR 52, *Licenses, Certifications, and Approvals for Nuclear Power Plants (Ref. 1-25)*, the rationale underlying the rule supports the exemption request. Exempting GLE from this requirement will reduce licensee administrative and information collection burdens, but serve the same health and safety functions as the current labeling requirements. Therefore, the exemption does not affect the level of protection for either the health and safety of workers and the public or for the environment; nor does it endanger life or property or the common defense and security.

The NRC approved a similar exemption from 10 CFR 20.1904 requested by a prior uranium enrichment facility license applicant. In approving the exemption, the NRC concluded:

"Under 10 CFR 20.2301, the Commission may grant exemptions from the requirements of the regulations, if it determines that the request will be authorized by law and will not result in undue hazard to life or property. Also, 10 CFR 20.1905(c) already exempts containers from 10 CFR 20.1904, if the containers are attended by an individual who takes the precautions necessary to prevent the exposure of individuals in excess of the limits established. The staff agrees that it would be impractical to label each and every container in restricted areas at this facility because of the large number of potential containers. Labeling each container may also reduce radiation safety by desensitizing the worker to radiation warning signs. Since there is no statutory provision prohibiting the granting of this exemption, the staff concludes that the request is authorized by law. Also, the exemption request is consistent with those approved previously at the gaseous diffusion plants and other fuel cycle facilities. Experience at facilities that have received the exemption from the labeling requirement demonstrates that the applicant's request will provide an equivalent amount of safety, and will not result in an undue hazard to individuals. Accordingly, the staff finds that the request will not be an undue hazard to life or property. Therefore, exemption to the requirements of 10 CFR 20.1904 is recommended." (Ref. 1-24)

1.2.5.3 Exemption to Decommissioning Funding Requirements

The following proposed exemption from the requirements of 10 CFR 70.25(e) and 10 CFR 40.36(d) addressing the decommissioning funding requirements is identified in the Decommissioning Funding Plan (DFP) and GLE LA Chapter 10, *Decommissioning*.

10 CFR 70.25(e) and 10 CFR 40.36(d) require, in part, that *"The decommissioning funding plan must also contain a certification by the licensee that financial assurance for decommissioning has been provided in the amount of the cost estimate for decommissioning..."*. In accordance with the DFP, GLE will incrementally fund that portion of its total decommissioning costs associated with the disposition of UF_6 tails generated by facility operation. Specifically, GLE will provide financial assurance for the disposition of UF_6 tails based on the expected amount of UF_6 tails to be generated annually, in a forward-looking manner. The NRC has previously approved the same incremental decommissioning financial assurance approach for USEC's American Centrifuge Project (ACP) and Louisiana Energy Services', L.P. (LES) National Enrichment Facility (NEF).

This exemption is justified for the following reasons:

- It is authorized by law because there is no statutory prohibition on incremental funding of decommissioning costs.
- The requested exemption will not endanger life or property or the common defense and security because UF_6 tails are generated incrementally over the life of the plant. GLE will provide financial assurance for UF_6 tails already generated that require disposal and the projected UF_6 tails to be generated in the next year. As such, requiring financial assurance for the disposition of UF_6 tails to be generated over the full licensed operating life of the enrichment facility – at the time of initial license issuance – would impose an unnecessarily large financial burden on the licensee.
- Granting this exemption is in the public interest for the same reasons stated above. Moreover, by eliminating an unnecessarily large financial burden on the licensee, the exemption will facilitate the deployment of an advanced, next-generation enrichment technology in the United States, in furtherance of important national energy objectives.

Finally, providing financial assurance for UF_6 tails disposition on an incremental basis is justified in view of GLE's commitments to: (1) provide full financial assurance for facility decommissioning (assuming a six MSWU facility) at startup (startup refers to when GLE receives licensed material); (2) update its UF_6 tails dispositioning cost estimate annually, on a forward-looking basis, to ensure that the financial assurance reflects the current projected inventory of UF_6 tails at the facility (including any previously-generated tails still requiring disposition); and (3) adjust other decommissioning costs periodically, and no less frequently than every three years. This approach will allow GLE to consider available operating experience and other relevant information, including actual UF_6 tails inventory values and generation rates, and to ensure that sufficient decommissioning financial assurance is available at any point during the licensed operating life of the facility.

1.2.5.4 Authorization to Use ICRP 68

GLE requests authorization to use the derived air concentration (DAC) and annual limit on intake (ALI) values based on dose coefficients published in International Commission on Radiological Protection (ICRP) Publication No. 68, *Dose Coefficients for Intakes of Radionuclides by Workers (Ref. 1-26)*, in lieu of the values in Appendix B of 10 CFR 20, in accordance with approved written procedures.

The ICRP 68 guidance was promulgated after the 10 CFR 20, Appendix B criteria were established, and provides an updated and revised internal dosimetry model. Use of the ICRP 68 models provide more accurate dose estimates than the models used in 10 CFR 20, and allows GLE to implement an appropriate level of internal exposure protection. The NRC has established precedent for this exemption request from 10 CFR 20 in SECY-99-077.

1.2.5.5 Authorization to Make Changes to License Commitments

1.2.5.5.1 Changes Requiring Prior Approval

GLE shall not make changes to the License Application that decreases the effectiveness of commitments, without prior NRC approval. For these changes, GLE will submit to the NRC, for review and approval, an application to amend the license. Such changes shall not be implemented until approval is granted.

1.2.5.5.2 Changes Not Requiring Prior Approval

Upon documented completion of a change request for a facility or process, GLE may make changes in the facility or process as presented in the License Application, or conduct test or activities not presented in the License Application, without prior NRC approval, subject to the following conditions:

1. There is no degradation in the safety commitments in the License; and
2. The change, test, or activity does not conflict with any condition specifically stated in the License Application.

Records of such changes shall be maintained, including technical justification and management approval, in dedicated records to enable NRC inspection upon request at the facility. A report containing a description of each such change, and appropriate revised sections to the License Application, shall be submitted to the NRC within three (3) months of implementing the change.

1.2.5.6 Exemption from 10 CFR 21.3 Definitions

GLE requests authorization to replace the definitions of basic component, commercial-grade items, critical characteristics, dedication, and dedicating entity as they apply to facilities licensed pursuant to 10 CFR 70 with the following:

Basic Component: A structure, system, or component (SSC) designated as an item relied on for safety (IROFS), or part thereof that affects the IROFS function, that is directly procured by the licensee of a facility or activity subject to the regulations in 10 CFR 70 and in which a defect or failure to comply with any applicable regulation or this chapter, order, or license issued by the U.S. Nuclear Regulatory Commission (NRC) could create a substantial safety hazard (i.e., exceed the performance requirements of 10 CFR 70.61). In all cases, basic components include IROFS-related design, analysis, inspection, testing, fabrication, replacement of parts, or consulting services that are associated with the component hardware, whether these services are performed by the component supplier or others.

Commercial-Grade Item: A structure, system, or component (SSC), or part thereof that affects its IROFS function, which is not designed and manufactured as a basic component. Commercial-grade items do not include items where the design and manufacturing processes require in-process inspections and verifications to ensure that defect or failures to comply are identified and corrected (i.e., one or more critical characteristics of the item cannot be verified.)

Critical Characteristics: Those important to design, material, and performance characteristics of a commercial-grade item that, once verified, will provide reasonable assurance that the item will perform its intended IROFS function.

Dedication Process: An acceptance process undertaken to provide reasonable assurance that a commercial-grade item or service to be used as a basic component will perform its intended IROFS function and, in this respect, is deemed equivalent to an item designed and manufactured under a 10 CFR 50, Appendix B, Quality Assurance Program. This assurance is achieved by identifying the critical characteristics of the item and verifying their acceptability by inspections, tests, or analyses performed by the purchaser or third-party dedicating entity after delivery, supplemented as necessary by one or more of the following: commercial grade surveys; product inspections or witness at holdpoints at the manufacturer's facility, and analysis of historical records for acceptable performance. In all cases, the dedication process must be conducted in accordance with the applicable provisions of 10 CFR 50, Appendix B. The process is considered complete when the item is designated for use as a basic component.

Dedicating Entity: The organization that performs the dedication process. Dedication may be performed by the manufacturer of the item, a third-party dedicating entity, or the licensee itself. The dedicating entity, pursuant to 10 CFR 21.21(c), *Notification of Failure to Comply or Existence of a Defect and its Evaluation (Ref. 1-27),* is responsible for identifying and evaluating deviations, reporting defects and failure to comply for the dedicated item, and maintaining auditable records of the dedication process. In cases where the Licensee applies the commercial-grade item procurement strategy and performs the dedication process, the Licensee would assume full responsibility as the dedicating entity.

1.2.5.7 CAAS Exemption on the Cylinder Storage Pads

GLE requests exemption from the use of a Criticality Accident Alarm System (CAAS) to cover the UF_6 Cylinder Storage Pads (MPF-106, -107, and -108), Trailer Storage Area, and UF_6 Cylinder Staging Area. The exemption is based on the full discussion presented in GLE LA Section 5.3.5.1 and is summarized as follows:

In the UF_6 Cylinder Storage Yards, most of the storage is provided for source material, not special nuclear material (SNM). Only 30B model cylinder containing SNM at 5 wt% ^{235}U, or less, is stored on the Product Pad. Storage of 30B model cylinders is short term and involves fewer cylinders than Tails or In-Process Storage thus further reducing the total likelihood for mishaps. Installation of CAAS to cover these storage yards will require detection clusters mounted high over the pads and require increased traffic into the storage yards for maintenance, functional testing, and calibration activities. This introduces additional hazards to the worker working at heights and presents an increased cylinder damage hazard from falling items and collapsing lift equipment.

1.2.6 Security of Classified Information

GLE has requested a facility security clearance, in accordance with 10 CFR 95, in a separate submittal. The use, processing, storage, reproduction, transmission, transportation or handling of classified information necessary to support this license application is currently controlled under the NRC authorized GNF-A facility security clearance at the Secret Restricted Data (SRD) level. As a result, access to restricted data (RD) or national security information (NSI) for the GLE Commercial Facility shall continue to be controlled by GNF-A in accordance with 10 CFR 25, *Access Authorization (Ref. 1-28)*, 10 CFR 95, and any other requirements that the NRC imposes through the issuance of Orders, until such time NRC processes GLE for an approved facility security clearance at the SRD level. Classified information associated with this LA, but not part of the facility security clearance request has been transmitted in a separate submittal.

1.3 SITE DESCRIPTION

This section contains a summary description of the Wilmington Site and surrounding areas. The GLE Environmental Report (ER) *(Ref. 1-29)* contains more detailed information regarding the site and its environs.

1.3.1 Site Geography

This section contains information regarding the site location, including nearby highways, bodies of water, and other geographical features.

1.3.1.1 Site Location Specifics

The GLE Commercial Facility is located on an existing industrial site in Wilmington, North Carolina. The existing Wilmington Site is situated on a 1621-acre tract of land, located west of North Carolina Highway 133 (also known as Castle Hayne Road). The Wilmington Site lies between latitudes (North) 34° 19' 4.0"and 34° 20' 28.9" and longitudes (West) 77° 58' 16.4" and 77° 55' 19.8", and is approximately six (6) miles north of the City of Wilmington in New Hanover County, North Carolina (see Figure 1-1 and Figure 1-2). For further information, see Section 1.1.1.

The southeastern corner of the Wilmington Site is adjacent to the interchange of Interstate 140 with Castle Hayne Road. Current access to and from the Wilmington Site by trucks and other vehicle traffic is from Castle Hayne Road. Northbound Castle Hayne Road from the Interstate 140 interchange bordering the Wilmington Site is a four-lane road that continues for approximately one-half mile before narrowing to two lanes. The Wilmington Metropolitan Planning Organization designated Castle Hayne Road as an urban principal arterial south of Interstate 140 and as an urban minor arterial north of the Interstate 140 interchange.

1.3.1.2 Features of Potential Impact to Accident Analysis

The surrounding terrain is typical for coastal North Carolina. The terrain has an average elevation of less than 40 feet above msl and is characterized by gently rolling land, with rivers, creeks, swamps, and marshlands. Approximately 182 acres of the southwest portion of the Wilmington Site are classified as swamp forest. There are no mountain ranges nearby. The terrain of the GLE Site is very gently sloping (gradients less than 2 percent) with little relief; therefore, landslides are not credible events. There is no volcanic or glacial activity in the region or vicinity of the Wilmington Site.

The elevation of the GLE Site is above the 500-year coastal still water flood elevation (coastal still water elevations factor in potential impacts from storm surge, including tidal and wind setup effects). The GLE Commercial Facility is located outside both the 100- and 500-year flood plains and there are no dams in the vicinity that could contribute to a rapid flood event. The site may be subject to a maximum probable flood event resulting from combined river flooding of the Cape Fear and Northeast Cape Fear Rivers. This type of event would be very slow moving thus allowing ample warning for safe shutdown. GLE will have procedures for determining what actions to take in the event of inclement weather (i.e., whether to shut down operations). Additionally, the design of systems and components within the facility are evaluated for the flooding to ensure any accidents that could result are "Highly Unlikely", and will not cause any accident scenarios resulting in consequences exceeding the performance criteria in 10 CFR 70.61.

Due to the curvature of the coastline in the area, the ocean lies approximately 10 miles east and 26.4 miles south of the Wilmington Site. The Federal Emergency Management Agency defines the geographic threshold for concern regarding a tsunami as one (1) mile inland from the coast with an elevation of 25 feet above msl. Given the distance of the Wilmington Site from the ocean, there are no direct threat effects of a potential tsunami. Because of the distance of the Wilmington Site upstream from the Atlantic Ocean (approximately 23 river miles) and the height of the GLE Site above the 500-year floodplain, the indirect effects of flooding from a tidal bore in the Northeast Cape Fear River induced by a tsunami are minimal.

The Mid-Atlantic Coastal Plain province counties in North Carolina are in a low potential zone for the presence of radon gas relative to other regions in the state.

Soil samples collected at the GLE Site typically do not have high amounts of natural organic material. In addition, no peat deposits that could be a potential source of methane gas have been identified at the GLE Site. There are no municipal landfills on or in the immediate vicinity of the Wilmington Site that could generate methane gas; therefore, methane gas buildup beneath the Wilmington Site is not credible.

The projected lowering of the potentiometric surface at the GLE Site, as a result of the groundwater withdrawals from the aquifer on and in the vicinity of the Wilmington Site, is minimal, and no greater than the historical seasonal fluctuations observed in groundwater levels. In addition, the absence of a thick or regionally continuous confining bed at the GLE Site further minimizes the potential for subsidence as a result of lowered groundwater levels; therefore, subsidence due to dewatering is not credible. Likewise, there are no active mines adjacent to the Wilmington Site or known economic deposits of minerals, stone, or fuel materials that could cause subsidence at the GLE Site.

1.3.2 Demographics

This section provides the current census results (calendar year [CY] 2000) for the area surrounding the Wilmington Site, to include specific information about populations, public facilities, and industrial facilities. Land use and nearby bodies of water are also described.

1.3.2.1 Latest Census Results

According to the U.S. Census Bureau's 2000 Decennial Census *(Ref. 1-30)*, a total of 321 census blocks fall within a five-mile radius of the Wilmington Site. The majority of these census blocks (261) is within New Hanover County and includes 12,997 persons and 4,953 households. A total of 57 Pender County census blocks are within the five-mile radius, with a combined population of 3,305 persons and 1,274 households. An examination of census block data from CY 2000 reveals a total of three census blocks in Brunswick County with some portion of the total area inside the five-mile radius. The total population of these three (3) census blocks is 36 persons in 17 households. Blocks with any portion of their area inside the five-mile radius were included in this population count. (See GLE ER Section 3.10.1 for additional information.)

1.3.2.2 Description, Distance, and Direction to Nearby Population Area

The region around the site is lightly settled with large areas of heavily timbered tracts of land. Farms, single-family dwellings, and light commercial activities are located along North Carolina Highway 133. In the eastern and southern vicinities of the Wilmington Site, residential uses are dominant due to the presence of the Wrightsboro (south), Skippers Corner (east), and Castle Hayne (northeast) communities. Wrightsboro has a population of approximately 4500, Skippers Corner has a population of approximately 1200, and Castle Hayne has a population of approximately 1100. (See GLE ER Section 3.1 for additional information.)

1.3.2.3 Proximity to Public Facilities

Figure 1-6, *Community Characteristics Near the Wilmington Site*, shows the location of schools and parks with respect to the five-mile Wilmington Site radius. There are a total of 90 public and private elementary, middle, and high schools in the three-county region. In addition to these primary and secondary schools, colleges such as the University of North Carolina at Wilmington (UNCW), Brunswick Community College, and Cape Fear Community College are located in the region. Out of the 90 schools in the region, one is within a four-mile radius of the GLE Site (Wrightsboro Elementary) and 21 schools are within an eight-mile radius of the GLE Site. The nearest hospital, New Hanover Regional Medical Center, is approximately six (6) miles from the Wilmington Site.

No state or federal parks are located within five (5) miles of the Wilmington Site. There are 18 parks, three trails, and three gardens maintained by New Hanover County. Four of the parks are located within a five-mile radius of the Wilmington Site.

1.3.2.4 Nearby Industrial Facilities

The Northeast Cape Fear River borders the Wilmington Site to the west, and industrial land uses are dominant on the opposite (west) side of the river. The BASF Corporation, Elementis Chromium Facilities, and the L.V. Sutton coal-fired power plant operated by Progress Energy are examples of industrial operations located in this area. The industrial area sits between the Northeast Cape Fear River and the main branch of the Cape Fear River.

1.3.2.5 Land Use within a Five Mile Radius

The land use in the vicinity of the Wilmington Site is discussed below and generally covers the five-mile radius around the Wilmington Site. The Wilmington Site is a 1,621-acre parcel, owned by the GE, located west of Castle Hayne Road (otherwise known as North Carolina Highway 133). The property is currently zoned I-2, which is described in the New Hanover County zoning code as intended for heavy industrial uses. No portion of the property is currently used for agricultural purposes.

Immediately north of the Wilmington Site is a large parcel of approximately 4,069 acres owned by Hilton Properties. The current zoning designation for this property is Rural Agricultural, which is designed for low-density residential development with an emphasis on farming and open-space preservation. This parcel is locally known as the Sledge Forest and is currently used for timber management and as a private hunting area. Access to the Sledge Forest is provided via a private, unpaved road that intersects with Castle Hayne Road and closely follows the northern property line of the Wilmington Site.

The Northeast Cape Fear River borders the Wilmington Site to the west, and industrial land uses are dominant on the opposite (west) side of the river. The BASF Corporation, Elementis Chromium facilities, and the L.V. Sutton coal-fired power plant operated by Progress Energy are examples of industrial operations located in this area. The industrial area sits between the Northeast Cape Fear River and the main branch of the Cape Fear River. In the eastern and southern vicinities of the Wilmington Site, residential uses are dominant due to the presence of the Wrightsboro (south), Skippers Corner (east), and Castle Hayne (northeast) communities.

Three (3) public schools are located within five (5) miles of the Wilmington Site: Wrightsboro Elementary School, Emma B. Trask Middle School, and Emsley A. Laney High School. Trask Middle School also serves as an emergency shelter for New Hanover County.

The Wilmington International Airport (ILM) is located approximately five (5) miles south-southeast from the Wilmington Site. The New Hanover County Landfill is located approximately four (4) miles southwest of the Wilmington Site.

1.3.2.6 Land Use Within One Mile of the Facility

As described above, the Wilmington Site is bordered on the north by the Sledge Forest and on the west by the Northeast Cape Fear River. Castle Hayne Road borders the eastern portion of the site. Further north along Castle Hayne Road, are four (4) mobile homes located on the opposite side of the street from the Wilmington Site. Adjacent to the site on the northeast side is the Wooden Shoe residential subdivision. Located adjacent to the Wilmington Site's eastern boundary across Castle Hayne Road, are the North Carolina State University Horticultural Crops Research Station, a truck parking lot, and a small recreational park for use by Wilmington Site employees (owned by GE). Directly south of the site is the Interstate 140, and beyond the interstate is a small residential area.

1.3.2.7 Uses of Nearby Bodies of Water

A portion of the Wilmington Site borders the Northeast Cape Fear River. Both commercial and recreational fishing occur on the Northeast Cape Fear River. Commercial fishing is more prevalent downstream of the Wilmington Site and in the Cape Fear River Estuary.

1.3.3 Meteorology

1.3.3.1 Primary Wind Directions and Average Wind Speeds

On an annual basis, the wind direction (direction from where the wind is blowing) at Wilmington International Airport is predominantly southwesterly *(Ref. 1-31)*; thus, reflecting the general synoptic scale wind pattern. In contrast, the predominant wind direction during the fall and winter is often northerly, due largely to the influence of invading polar air masses and changes in global circulation *(Ref. 1-31; Ref. 1-32)*. Figure 1-7, *Wind Rose for Wilmington International Airport*, shows the overall wind rose for Wilmington International Airport. The annual prevailing wind speed at the airport is 10.4 mph (9 knots) *(Ref. 1-31)*.

1.3.3.2 Annual Precipitation – Amounts and Forms

The mean annual precipitation in eastern North Carolina is heaviest in the southeast corner of the state and steadily decreases toward the north and west. The higher precipitation amounts are due to higher levels of moisture provided by the Atlantic Ocean. The area along the North Carolina coast experiences afternoon showers and thunderstorms often during the summer months. These storms form along a sea breeze front as it moves inland from the coast. The mean annual precipitation for the area around the GLE Commercial Facility is approximately 55.0 inches/year according to the 1948 to 1995 dataset *(Ref. 1-31)* and 57.1 inches/year according to the 1971 to 2000 dataset *(Ref. 1-33)*.

Due to the moderate climate, Wilmington receives very little snowfall, except on rare occasions. On average, only about 2.1 inches of snowfall occurs annually. December and January are expected to receive the most average snowfall, at 0.6 inches *(Ref. 1-33)*. Wilmington also receives only a small amount of sleet. The mean recurrence interval for measurable sleet in Wilmington, North Carolina, is approximately 4.6 years, or an annual probability about 22 percent. Sleet greater than 0.25 inches has a mean recurrence interval of only once every 46 years, or an annual probability of about 2 percent *(Ref. 1-34)*. Freezing rain usually poses a higher risk to power systems and trees than sleet. Freezing rain does not occur often in Wilmington, although it occurs more often than sleet *(Ref. 1-34)*. Measurable accumulations occur in Wilmington with a mean recurrence interval of about 1.5 years, or an annual probability of 67 percent. More significant accumulations of less than 0.25 inches occur with a mean recurrence interval of 7.7 years, or an annual probability of 13 percent. Accumulations of less than 0.5 inches, which are very likely to affect power lines and trees, are expected to occur in Wilmington at a mean recurrence interval of 46 years, or an annual probability of 2 percent.

1.3.3.3 Severe Weather

1.3.3.3.1 Extreme Temperature

The highest recorded temperature at Wilmington International Airport for the period of record is 104.0°F, which occurred during June 1952 *(Ref. 1-33)*. The lowest recorded temperature of 0.0°F occurred in December 1989 *(Ref. 1-33)*. This shows that the maximum annual temperature range at the Wilmington Site is about 104.0°F.

1.3.3.3.2 Extreme Precipitation

Tropical storms and hurricanes occur in and around the southeastern United States, making Wilmington prone to high amounts of rainfall over a short time period. The highest recorded 24-hour rainfall amount of 13.38 inches at Wilmington International Airport occurred during September 1999 due to the effects of Hurricane Floyd making landfall on the North Carolina Coast *(Ref. 1-33)*. The maximum one-time extreme rainfall resulting from Hurricane Floyd is considered the deterministically defined maximum extreme rainfall event. Considering the expected precipitation intensity, Wilmington International Airport has a 1 in 50 annual exceedance probability (AEP) of receiving precipitation at a rate of 11.86 inches/hour for a duration lasting five minutes. The AEP for precipitation with a rate of 16.05 inches/hour occurring for five minutes is about 1 in 1,000. Generally, the intensity of rainfall that could occur for a given AEP decreases as the duration of the precipitation event increases *(Ref. 1-35)*. Based on GLE site elevation and facility design, a severe local storm that meets the deterministically defined maximum extreme rainfall event would not flood the GLE facility site, nor impact the design of the structures.

On rare occasions, Wilmington can receive large snowfall amounts. During a storm event in late December 1989, the area received 9.6 inches of snow in a 24-hour period *(Ref. 1-32 and 1-36)*. This December 1989 storm also matched a previous record snow depth of 13 inches and is used as the deterministic design basis snow load event. The roof design parameters for the GLE Commercial Facility as required by the International Building Code (IBC) for the region exceed the expected loadings from snow and ice. However, the highest drift snow load may exceed the normal snow load, which could impact the live load roof capacity at roof locations where there is an interface between roof elevation changes. For the roof decking in these interface areas, the snowdrift load could cause the decking to first sag and eventually fail, allowing snow and water to enter the building. It is important to note that in the locations where these failures could occur, licensed material or hazardous chemicals are not present, thus this type of roof failure does not represent a high or intermediate consequence event. As a result, the impact of a severe snow event, including up to the snow loads in the design basis snow load event, is determined to be "Highly Unlikely" to result in consequences in excess of the performance criteria in 10 CFR 70.61.

1.3.3.3.3 Extreme Winds

Extreme winds may occur at Wilmington International Airport due to localized events, such as thunderstorm downdrafts, microbursts, or tornadoes. In addition, the airport lies in a particularly vulnerable location for hurricane-force winds. As of 1995, the highest wind gust measured at the airport was approximately 78 mph (68 knots) *(Ref. 1-31)*; however, since that time, Wilmington has experienced Hurricanes Fran (1996), Floyd (1999), and Charley (2004). Hurricane Fran had a peak gust of approximately 86 mph (75 knots) measured at the Wilmington International Airport. Hurricane Floyd similarly caused a wind gust of approximately 86 mph (75 knots) at the airport *(Ref. 1-37)*. Hurricane Charley had somewhat lower wind gusts of approximately 74 mph (64 knots) at the airport *(Ref. 1-38)*. The likelihood and consequences of design basis wind velocities are discussed further in Section 1.3.3.3.7.

1.3.3.3.4 Thunderstorms

Rainfall in the region during the summer months comes primarily from thunderstorms. These storms occur on approximately 33 percent of days during June through August in the vicinity of the Wilmington Site and are scattered and uneven in coverage *(Ref. 1-31)*. Although the inland advance of the sea breeze front often causes summer thunderstorms, other primary causes of thunderstorms in the Wilmington area are tropical storms or hurricanes approaching from the south and southeast, and large-scale synoptic fronts approaching from the north and west. The latter two causes of thunderstorms also increase the chance of severe weather. For example, hail is observed in the Wilmington area on an average of about once per year *(Ref. 1-31)* and is most likely to be associated with synoptic frontal thunderstorms. Severe thunderstorms may produce damaging straight-line winds greater than 57 mph (50 knots). According to the National Severe Storms Laboratory (NSSL) *(Ref. 1-39)*, the area surrounding the Wilmington Site experiences approximately four days per year of damaging thunderstorm winds or winds less than 57 mph (50 knots) due to a thunderstorm.

1.3.3.3.5 Lightning

Another hazard of thunderstorms is lightning, which can strike miles from a thunderstorm and often occurs without warning. Besides the obvious danger to personnel working outside, lightning can disrupt electrical circuits and cause fires. The region surrounding the Wilmington Site has experienced a lightning flash density ranging from 4 to 8 flashes/km^2/year over the period from 1996 through 2000.

1.3.3.3.6 Tornados

Fifteen (15) tornadoes are known to have touched down in New Hanover County, North Carolina, between 1950 and 2004, including waterspouts in the sound and on the Atlantic Ocean. The strongest of these tornadoes occurred on June 13, 1962 in the western part of the county and measured F2 on the Fujita scale (meaning it was capable of producing considerable damage). Wind speeds associated with an F2 tornado are between 113 and 157 miles per hour (mph).

Based on evaluation of data from the National Severe Storms Laboratory *(Ref. 1-39)*, a tornado would be expected to occur within 25 miles of the Wilmington Site on 0.4 to 0.6 days per year. The ocean covers a significant portion of the area within 25 miles of the Wilmington Site; therefore, some of these tornadoes could occur as waterspouts. From a probabilistic perspective, tornado design basis guidance indicates that tornadoes in the Wilmington area would be expected to have up to 230-mph maximum winds at an exceedance probability of 10^{-7} per year *(Ref. 1-40)*. This change in expected intensity would not be abrupt, but due to the coarse nature of the grid cells used in Regulatory Guide 1.76, *Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants (Ref. 1-41)*, to calculate the intensity regions, there is a sharp demarcation between regions. Nevertheless, using this approach, the likelihood of a tornado of this magnitude is "Highly Unlikely".

Because there is no record of an F4 or F5 tornado in NC, and none of the tornadoes in the Wilmington area were stronger than a F1 tornado, from a deterministic perspective, a conservative tornado for the GLE site would be a F2 tornado (118 to 161 mph 3-second gust speed equivalent). NUREG/CR-4461, *Tornado Climatology of the Contiguous United States (Ref. 1-42)*, indicates that the tornado wind speed with an annual probability of 10^{-5} is 140 mph (3-second gust speed) for the region (Region II) in which the GLE site is located. In accordance with the performance requirements of 10 CFR 70.61 in Subpart H a tornado with an annual probability of 10^{-5} can be considered as a "Highly Unlikely" event. Since the 140 mph tornado wind speed compares with the wind speeds associated with the deterministic F2 tornado for the site, the deterministic tornado wind speed for the site is 140 mph (3-second gust speed). This magnitude wind is bounded by the wind speed identified for hurricanes as described below.

1.3.3.3.7 Tropical Storms and Hurricanes

The area of New Hanover County could expect the following return periods for each category of hurricane passing within approximately 86 miles (75 nautical miles):

- Category 1, 6 to 10 years;
- Category 2, 23 to 30 years;
- Category 3, 33 to 44 years;
- Category 4, 79 to 120 years; and
- Category 5, 191 to 250 years (*Ref. 1-40*).

Because winds are stronger on the right side of the storm's eye, causing more wind damage and higher storm surges, the greatest meteorological threat to New Hanover County comes from hurricanes that strike land in the approximate area between the South Carolina border and the outlet of the Cape Fear River. In addition, the strongest bands of rain occur in front of a hurricane as it approaches, resulting in a great deal of heavy, flooding rain in New Hanover County when a storm approaches this area of coastline. Between 1954 and 2004, three hurricanes, ranging from Category 1 through Category 3, made landfall in the area. Two of the hurricanes, Hurricanes Hazel (1954) and Fran (1996), were Category 3 storms that made landfall with winds between 111 to 130 mph.

Based on the above, the most severe hurricane recorded in proximity to the Wilmington Site is a Category 3 hurricane, however a Category 4 hurricane is used as the deterministic hurricane for GLE facility design. The wind speeds for a Category 4 hurricane range from 131 to 155 mph. When comparing the various contributors to wind speeds (thunderstorms, tornados, hurricanes), the hurricane is the source of the highest wind speed of up to 155 mph, and thus this value is the design basis wind velocity. The wind speed defined in ASCE 7-05, *Minimum Design Loads for Buildings and Other Structures (Ref. 1-43)*, to be applied for the GLE facility on the Wilmington Site is 140 mph. Implementation of the wind design requirements in ASCE 7-05 requires the use of a loading factor of 1.6 to wind loads, which is equivalent to using a wind speed of 177 mph for the design. Because the equivalent design wind of 177 mph for the GLE facility is larger than the design basis wind velocity of 155 mph, the design basis wind event is considered to be "Highly Unlikely" and will not cause any accident scenarios resulting in exceeding the performance criteria in 10 CFR 70.61.

According to the examination of NOAA storm surge data *(Ref. 1-44)*, most portions of the Wilmington Site at an elevation of 25 feet above msl, including the GLE Commercial Facility would not be directly affected by the highest storm surge. This is further supported by the storm surge potential from hurricanes being estimated at 21.94 feet as presented in Regulatory Guide 1.59, *Design Basis Floods for Nuclear Power Plants (Ref. 1-45)*. As a result, the event potential from a hurricane induced storm surge event as "Highly Unlikely."

1.3.3.3.8 Floods

The GLE Site does not fall within 100-year or 500-year floodplains *(Ref. 1-46)*; however, some of the low-lying areas on the Wilmington Site contain swamp forest that borders the Northeast Cape Fear River. Much of this swamp forest is in the floodplain and may flood upstream during extreme rain events. As a result, the GLE site may be subject to a maximum probable flood event as discussed in Section 1.3.1.2.

1.3.4 Hydrology

The section contains descriptions of nearby water bodies, groundwater on and near the Wilmington Site, and design basis flood events.

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

Bodies of water in the vicinity of the Wilmington Site are the Northeast Cape Fear River (which borders the Wilmington Site to the west) and its associated tributaries and creeks. The Northeast Cape Fear River is a blackwater river with relatively low levels of dissolved oxygen and higher turbidity than the Cape Fear River. The Northeast Cape Fear River and its tributaries have a naturally low pH and are classified as swamp water by the North Carolina Department of Environment and Natural Resources Division of Water Quality. At the Wilmington Site, the river is tidally influenced. Salinity concentrations vary with the rate of freshwater input and the amount of tidal exchange.

On the Wilmington Site, there are three (3) streams that provide habitat to aquatic wildlife. Two of the streams, unnamed Tributaries No. 1 and No. 2 (located in the Swamp Forest community in the Western Site Sector), drain to the Northeast Cape Fear River. The remaining stream is located on the Eastern Site Sector and drains northward to Prince George Creek. The first two are unnamed tributaries to the Northeast Cape Fear River and are classified as freshwater streams, but their lower reaches are tidally influenced by the river. The third stream, the unnamed tributary to Prince George Creek, is a freshwater stream and is not tidally influenced within the Wilmington Site. All streams are capable of accommodating the aquatic species associated with the neighboring Northeast Cape Fear River. However, the tidal variations in dissolved oxygen and salinity may affect the suitability of the habitat for some species.

In addition, there are three (3) small ephemeral ponds in the Western Site Sector and North-Central Site Sector, along with wetland areas throughout the Site that provide habitat. These areas provide a water source for wildlife found on the Wilmington Site.

1.3.4.2 Depth to the Groundwater Table

On the Wilmington Site, the water table is generally located near the land surface averaging approximately nine (9) feet below ground surface (bgs) with a range from 0 to 20 feet bgs.

1.3.4.3 Groundwater Hydrology

The Wilmington Site is within the North Carolina Coastal Plain physiographic province, which extends from the Piedmont eastward to the North Carolina coast. The coastal aquifer system is an eastward-dipping and eastward-thickening wedge of depositional sediments and sedimentary rock underlain by a crystalline, eroded surface of igneous and metamorphic rock (Precambrian or Early Paleozoic age). Six (6) regional aquifers are present in the region surrounding the Wilmington Site, including the Surficial Aquifer, Castle Hayne Aquifer, Peedee Aquifer, Black Creek Aquifer, and the Upper and Lower Cape Fear Aquifers. The aquifers are water-yielding formations that are more permeable than the finer-grained formations (confining units) that are typically above and/or beneath these coastal aquifers. In most areas, a less-permeable confining unit, with the exception of the Surficial Aquifer, overlies each aquifer that is under water-table conditions. The aquifers and confining units consist of sands, conglomerates, silts, clays, shell hash, and fossiliferous limestones deposited in nearshore and deltaic to offshore marine environments *(Ref. 1-47)*.

1.3.4.4 Characteristics of the Uppermost Aquifer

The Surficial Aquifer includes undifferentiated, stratified sediments. These sediments typically include terraced and barrier beach deposits, fossil sand dunes, and stream channel deposits. The sediment texture varies from medium- to fine-grained sands to silts and clays. This aquifer is recharged directly by rainfall, and the water table is generally located relatively near the land surface (approximately averaging nine (9) feet bgs with a range from 0 to 20 feet bgs). The hydraulic conductivity of the Surficial Aquifer has been estimated to be approximately 130 feet/day.

The Surficial Aquifer discharges into streams, drainage canals/ditches, and the low-lying swampy areas on the Wilmington Site. In addition, the Surficial Aquifer recharges groundwater into the underlying Peedee Aquifer (referred to as the Principal Aquifer). Due to yield limitations, water supply from the Surficial Aquifer is primarily restricted to domestic use.

The Wilmington Site wells produce from the Peedee Aquifer, which is the principal aquifer under the site. Groundwater is used at the existing Wilmington Site for industrial process water and drinking water. The average annual withdrawal is approximately 1.0 million gpd. Water levels measured in wells that tap the Peedee Aquifer at the Wilmington Site were evaluated in terms of the long-term sustainability of the water resource. The water levels in the aquifer do not show a long-term downward trend. A review of potential future changes to the withdrawal rates indicate that the existing water use and future estimates (approximately 10 percent increase) do not exceed the sustainable yield of the aquifer in this area (See GLE ER). The hydraulic conductivity of the Peedee Aquifer has been estimated to be approximately 38 feet/day.

1.3.4.5 Design Basis Flood Events Used for Accident Analysis

The GLE Commercial Facility is located on a high bluff, outside the 100-year (10^{-2}) and 500-year (2×10^{-3}) floodplains (that is, 0.2% chance of a catastrophic flood occurring at the level of a 500-year floodplain during any year). These flood levels occur at approximately 20 to 25 feet above msl. The Operations Building first floor elevations are above 25 feet msl.

1.3.5 Geology and Seismology

This section describes the geology and seismology at the Wilmington Site, including soil characteristics, earthquake magnitudes and return periods, and other geologic hazards.

1.3.5.1 Characteristics of Soil Types and Bedrock

Generally flat topography characterizes most of the Wilmington Site's physiography; however, the GLE Site is positioned on a topographic high compared to the adjacent land in that area of the Wilmington Site. The ground surface begins to gently roll into small low hills in the Northwestern Wilmington Site Sector, suggesting the presence of possible sand dune or remnant terrace deposits from shoreline migration in the recent geologic past. The Northeast Cape Fear River and its floodplain are the most prominent physiographic features bordering the Western and Northwestern Wilmington Site sectors. High bluffs and extensive estuarine areas along this reach of the river help protect the GLE Site from flooding events. The area west of the river channel scar, which is clearly visible in aerial images, marks an ancient flow boundary of the Northeast Cape Fear River. The abandoned part of the channel is today an estuarine area of low topographic relief bordering the current river's edge.

Surficial sedimentary deposits at the Wilmington Site are interpreted to be mostly a result of deposition in the geologic past associated with the ancient Northeast Cape Fear River system. These surficial deposits overlie the Peedee Formation at the Site and are largely undifferentiated and unconsolidated alluvial sands, clayey sands, and clays. Some of these deposits are previously deposited marine sediments that were reworked and re-deposited by alluvial processes.

The sedimentary sequence in the GLE Site is comprised of 10 to 30 feet of thin layers of silty fine sands, silty fine clayey sands, fine sandy silts, and fine sandy clays that overlie the Peedee Formation. Surficial sands are present in the area with an apparent average thickness of less than 5 feet. Thicker surficial sand deposits of approximately 10 feet thick are present in some areas. Surficial sediments in the uppermost 4 to 10 feet of this sector range from dark brown and black sand with some organic material to gray and tan fine- to medium-grained sand with minimal gravel. Beneath these sands, a dark gray, very silty and clayey fine sand is present in some locations.

At the base of the surficial deposits in many locations on the Wilmington Site lies a substantial marine clay layer considered to be part of the Peedee Formation. The Peedee Clay layer is encountered at a typical depth range of 20 to 30 feet. Hydraulically, the Peedee Clay forms an important semi-confining unit overlying the Peedee Aquifer, which is the source of process water for the existing Wilmington Site. The presence of glauconite throughout the Peedee Clay and the absence of reworked sediments more characteristic of shallower alluvial deposits suggest the Peedee Clay is of marine origin; therefore, this marine clay layer is stratigraphically considered part of the Peedee Formation. The Peedee Clay varies in both thickness and distribution across the Site.

Field observations of samples collected during investigations of the GLE Site indicate that the consistency of the Peedee Clay is generally firm, but can be softer if located near the ground surface. In general, this clay layer contains more silt than sand and is easily distinguished from other surficial alluvial clays present in some areas of the GLE Site by the uniform presence of glauconite and the Peedee Clay's characteristic gray to dark gray color.

The potential for differential settlement, or the difference in settlement across a foundation, was considered when preparing facility and roadway engineering designs. No soil types on the GLE Site pose any construction concerns.

Previous geotechnical investigations on the Wilmington Site found that soil conditions required the use of a specialized structural in-ground support system. A geotechnical design investigation to determine the structural in-ground support system necessary to support the estimated heavy loading will be completed prior to commencement of construction. The geotechnical design investigation will be performed using the applicable regulatory guidance in Regulatory Guide 1.132, *Site Investigations for Foundations of Nuclear Power Plants (Ref. 1-48).*

1.3.5.2 Earthquake Magnitudes and Return Periods

Earthquake epicenters in the southeastern United States generally extend in a northeasterly orientation along the axis of the Appalachian Mountain range. In North Carolina, the vast majority of seismic activity is concentrated in the western mountainous regions, where sutures and faults are predominantly associated with North American collisional tectonics. There are clusters of events scattered throughout South Carolina, and a few isolated occurrences of singular events along the coast. A small number of events are recorded along the Mid-Atlantic Coastal Plain physiographic province. In summary, seismicity levels are low outside of the Charleston region and the mountains to the west. In the Wilmington Site region, seismicity levels are relatively low.

Since the mid-1990s, the U.S. Department of the Interior has published probability of exceedance maps for ground shaking at one (1) and five (5) hertz (Hz) for a 50-year time span *(Ref. 1-36)*. A spectral acceleration of one Hz represents low frequency ground shaking (appropriate for Rayleigh and Love surface waves), whereas a five-Hz spectral acceleration represents high-frequency ground shaking related to body waves (P-waves and S-waves). For many cases of interest, the primary controlling earthquake is the postulated event that governs the spectral accelerations in the five-to-ten Hz range *(Ref. 1-49)*. The maps are developed for peak horizontal ground acceleration or spectral accelerations with 2 percent, 5 percent, or 10 percent probability of being exceeded in 50 years on uniform firm-rock site conditions (Vs30 = 760 m/s). These data present the peak acceleration for earthquakes believed to be likely near a given site. The Wilmington Site has a peak acceleration of approximately 0.1 g at two percent probability for five Hz wave over 50 years. This corresponds to a peak acceleration of approximately 0.03 g for a ten percent probability of exceedance in 50 years (500-year earthquake).

There are no significant geological features in the Wilmington region that would produce a major earthquake. The IBC has identified this area as Zone 1 and considers seismic events of minor magnitude (Mercalli VI, Richter 5.5 – 6.0).

The Charleston, S.C., earthquake of 1886 was felt in Wilmington, producing effects equivalent to Mercalli V– VI (Richter 4.8 – 5.4). Since then there have been nine recorded seismological events in the Wilmington area, all of which have been minor in nature, producing effects no greater than Mercalli IV (Richter 4.5). The U.S Geological Survey predicts the probability of a Richter 4.75 event at 2 x 10-4 and a Richter 5.0 at 2 x 10-5.

Based on the U.S. Geological Survey, documented historical events, the IBC design criteria, and the design margins used both in establishing the IBC criteria and the building designs to meet the IBC, it is improbable that an earthquake would affect the structures on the GLE Commercial Facility Site in such a way as to cause an accident scenario resulting in consequences exceeding the performance criteria in 10 CFR 70.61.

1.3.5.3 Other Geologic Hazards

As described in Section 1.3.1.2, other geologic hazards are not present at the Wilmington Site. There are no mountain ranges nearby. The terrain of the GLE Site is very gently sloping (gradients less than two percent) with little relief; therefore, landslides are not credible events. There is no volcanic or glacial activity in the region or vicinity of the Wilmington Site.

Soil samples collected at the Wilmington Site typically do not have high amounts of natural organic material. In addition, no peat deposits that could be a potential source of methane gas have been identified within the GLE Site.

The projected lowering of the potentiometric surface in the GLE Site as a result of the groundwater withdrawals from the aquifer on and in the vicinity of the Wilmington Site is minimal, and no greater than the historical seasonal fluctuations have been observed in groundwater levels. In addition, the absence of a thick or regionally continuous confining bed on the GLE Site further minimizes the potential for subsidence as a result of lowered groundwater levels; therefore, subsidence due to dewatering is not credible.

There are no active mines adjacent to the Wilmington Site or known economic deposits of minerals, stone, or fuel materials that could cause subsidence at the GLE Site.

Using the soil information from the geotechnical design investigation mentioned in Section 1.3.5.1, the following activities will be conducted:

- The assessment of liquefaction potential of subsurface soils will be completed using the applicable guidance contained in Regulatory Guide 1.198, *Procedures and Criteria for Assessing Seismic Soil Liquefaction as Nuclear Power Plant Sites (Ref. 1-50).* The Ground Motion Response Spectra used for the liquefaction analysis will be based on guidance contained in the *International Building Code (Ref. 1-51).*

- Allowable bearing pressures for shallow and deep foundations will be evaluated using established geotechnical engineering methods. Methods anticipated for use include those contained in the following publications: NAVFAC DM 7, *Naval Facilities Engineering Command Design Manual (Ref. 1-52)*; *Foundation Engineering Handbook (Ref. 1-53)*; *Foundation Analysis and Design (Ref. 1-54)*; and FHWA-IF-99-025, *Drilled Shafts: Construction Procedures and Design Methods (Ref. 1-55).*

The evaluation of total and differential settlement for structure foundations will be completed using established geotechnical engineering methods. Methods anticipated for use include those contained in the following publications: NAVFAC DM 7, Foundation Engineering Handbook; and Foundation Analysis and Design.

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Table 1-1. Typical Types, Sources, Quantities of Solid Wastes Generated by GLE Commercial Facility Operations.

Waste Type	Waste Source	Estimated Average Annual Quantity Generated
Municipal Solid Waste (MSW)	General worker operations, maintenance, and administrative activities not involving the handling of or exposure to uranium	380 ton/yr
Nonhazardous Industrial Wastes	Nonhazardous wastes from equipment cleaning and maintenance activities (for example, used coolant, nonhazardous caustic, and filter media) that are recyclable or not accepted by MSW landfill	107 ton/yr
Resources Conservation and Recovery Act (RCRA) hazardous waste	Wastes designated as RCRA hazardous wastes from equipment and maintenance activities (for example, used cleaning solvents and used solvent-contaminated rags)	12 ton/yr
Low-Level Radioactive Waste (LLRW)	Laboratory waste from UF_6 feed sampling and analysis	97 lb/yr
	Combustible, uranium-contaminated used items (for example, worker personal protection equipment, swipes, step-off pads)	92 ton/yr
	Noncombustible, uranium-contaminated, used items (for example, spent filters from HVAC systems, liquid radiological waste treatment system, and area monitors) and corrective maintenance items (defective pigtails, valves, and other safety equipment that needs replacement)	863 yd^3/yr
	Liquid radiological waste treatment system filtrate/sludge	670 lb/yr

Table 1-2. Management of Solid Wastes.

Solid Waste Source	Onsite Waste Management	Offsite Waste Treatment/Disposal
Municipal solid waste (MSW)	Collected and temporarily stored in roll-off containers	Filled roll-off containers transported by commercial refuse collection service to an approved disposal site
Non-hazardous wastes from operations equipment cleaning and maintenance activities that are recyclable or not accepted by MSW landfill	Collected and temporarily stored in containers	Filled containers transported by truck to an approved disposal site[a]
Wastes designated as Resource Conservation and Recovery Act (RCRA) hazardous wastes	Collected and temporarily stored in containers	Filled containers transported by truck to an approved disposal site[b]
Laboratory waste from UF_6 feed sampling and analysis	Collected and temporarily stored in containers	Either transported by truck to an approved disposal site or transported to an approved uranium recovery vendor.
Combustible used or spent uranium-contaminated materials	Collected and temporarily stored in containers	Either transported by truck to an approved disposal site or transported to an approved uranium recovery vendor.
Noncombustible used or spent uranium-contaminated materials	Collected and temporarily stored in boxes	Filled boxes transported by truck to an approved disposal site[c]
Liquid Radiological Waste Treatment System filtrate/sludge	Collected and temporarily stored in metal cans	Filled cans transported by truck to an approved disposal site

[a] Licensed RCRA Subpart D landfill.

[b] Licensed RCRA Subpart C Treatment, Storage, and Disposal Facility (TSDF).

[c] Licensed Low-Level Radioactive Waste Disposal Facility.

Table 1-3. Typical Types, Sources, and Quantities of Wastewater Generated by GLE Commercial Facility Operations.

Wastewater Type	Wastewater Source	Typical Average Daily Quantity Generated
Process liquid radiological waste	Wastewaters from the Operations Building Decontamination/Maintenance Area; process area floor drains, sinks, sumps, and mop water; Laboratory Area floor drains, sinks, sumps, and mop water; change room showers and sink; and aqueous process liquids that have the potential to contain uranium	5,000 gpd
Cooling tower blowdown	Operations Building HVAC cooling tower	30,000 gpd
Sanitary Waste	Sanitary waste from building areas used by GLE personnel (for example, restrooms and break rooms)	10,500 gpd
Stormwater	Stormwater runoff from impervious surfaces (for example, building roofs, parking lots, service roads, outdoor storage pads, and other maintained areas)	Variable depending on local precipitation

Table 1-4. Management of Wastewater Generated by GLE Commercial Facility Operations.

Wastewater Type	Onsite Waste Management	Offsite Waste Treatment/Disposal
Process liquid radiological waste	Wastewaters collected in closed drain system connected to Radiological Liquid Waste Treatment System (RLETS). Treated radiological waste effluent discharged to existing Wilmington Site process wastewater aeration basin and Final Process Lagoon Treatment Facility (FPLTF)	Treated effluent from the Wilmington Site FPLTF is discharged at NPDES-permitted Outfall 001 to the onsite effluent channel
Cooling tower blowdown	Blowdown pumped from cooling tower to existing Wilmington Site FPLTF	Treated effluent from the Wilmington Site FPLTF discharged at NPDES-permitted Outfall 001 to the onsite effluent channel
Sanitary Waste	Sanitary waste collected in sewer system connected to existing Wilmington Site Sanitary Wastewater Treatment Plant. Waste stream treated by activated sludge aeration process.	Treated effluent from the Wilmington Site Sanitary Wastewater Treatment Plant is discharged at NPDES-permitted Outfall 002 to the onsite effluent channel
Stormwater	Stormwater runoff collected in drainage conduits and channels flowing to onsite retention basins.	Stormwater from onsite retention basins is discharged per requirements of NPDES stormwater permit.

Table 1-5. Typical GLE Air Emissions.

Constituent	Amount	Regulatory Limit
Uranium	8×10^{-15} µCi/mL [a]	3×10^{-12} µCi/mL [b]
Hydrogen Fluoride	< 0.50 lb/day	~0.50 lb/day [c]

a. Per Global Laser Enrichment Environmental Report, December 2008.

b Per 10 CFR 20, Appendix B.

c Best estimate provided as the actual limit is specified on the North Carolina Department of Environment and Natural Resources air permit to be issued prior to operations.

Table 1-6. {{{Proprietary Information withheld from disclosure per 10 CFR 2.390}}}

Table 1-7. Type, Quantity, and Form of Licensed Special Nuclear 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	140,000,000 kg
Uranium enriched in isotope ^{235}U up to 8 percent by weight and uranium daughter products	Physical: solid, liquid, and gas Chemical: UF_6, UF_4, UO_2F_2, oxides and other compounds	2,600,000 kg
^{99}Tc, transuranic isotopes and other contamination	Any	Amount that exists as contamination as a consequence of historical feed of recycled uranium at other facilities.

Figure 1-1. Wilmington Site and County Location.

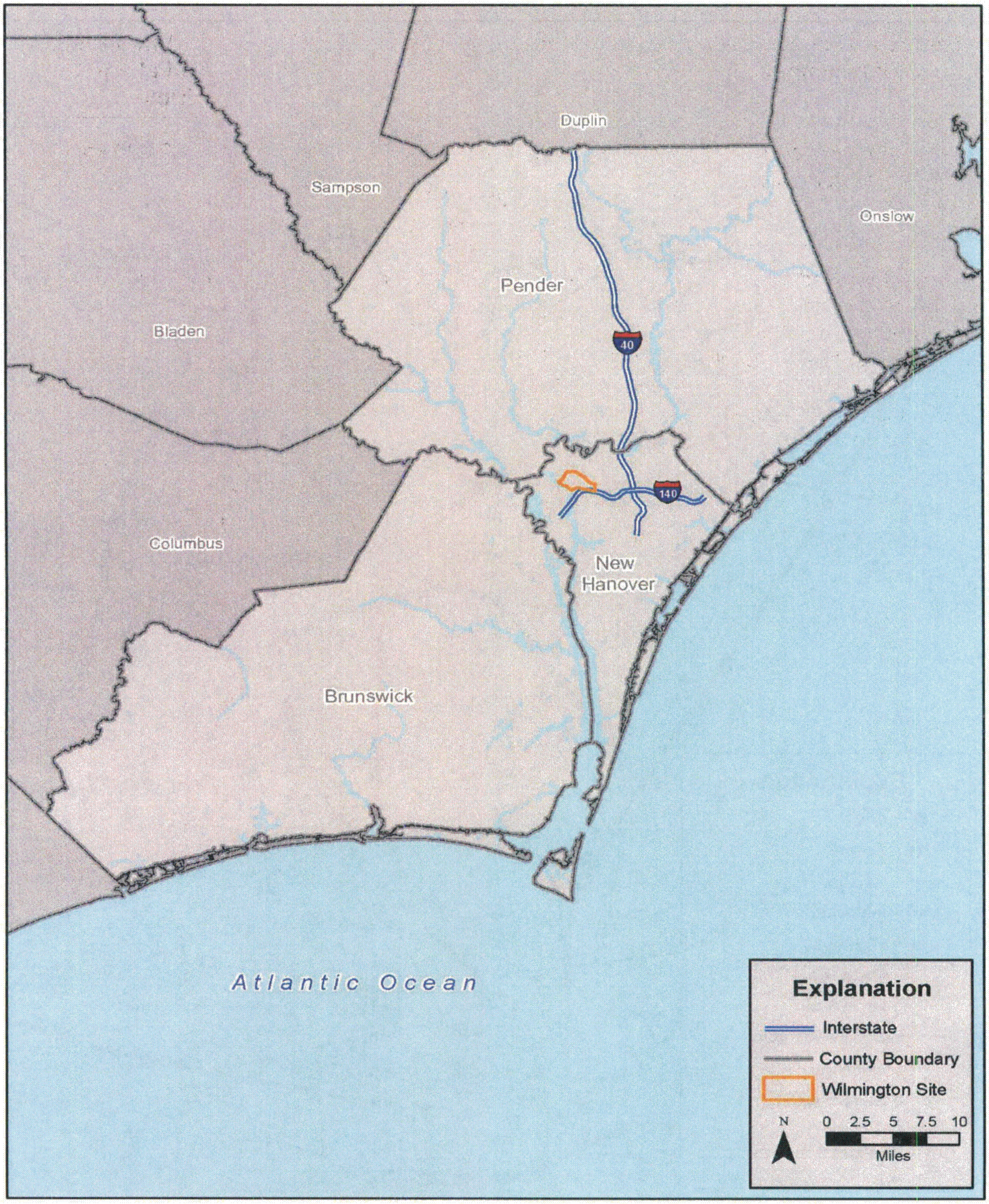

Figure 1-2. Wilmington Site, New Hanover County, and Other Adjacent Counties.

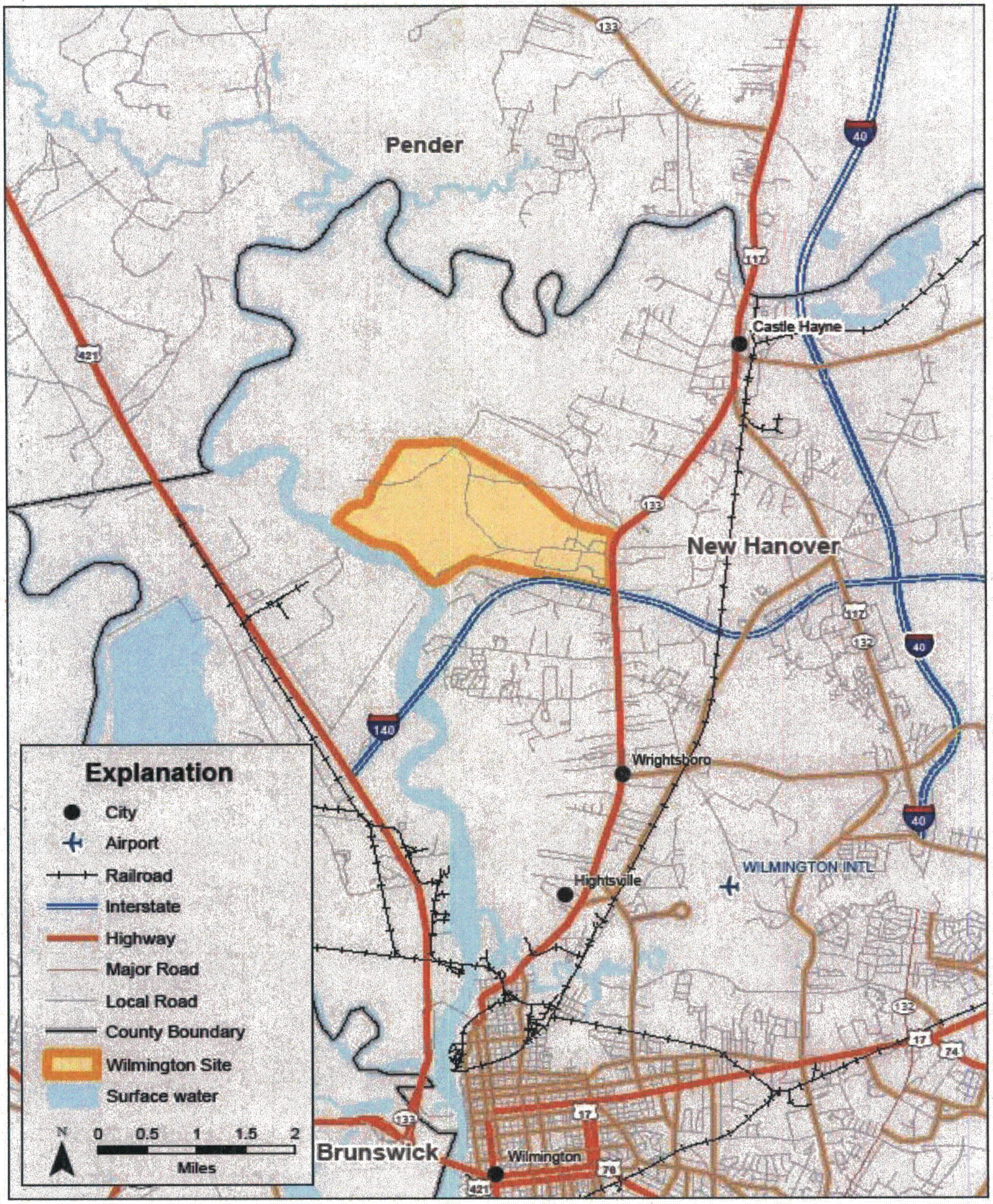

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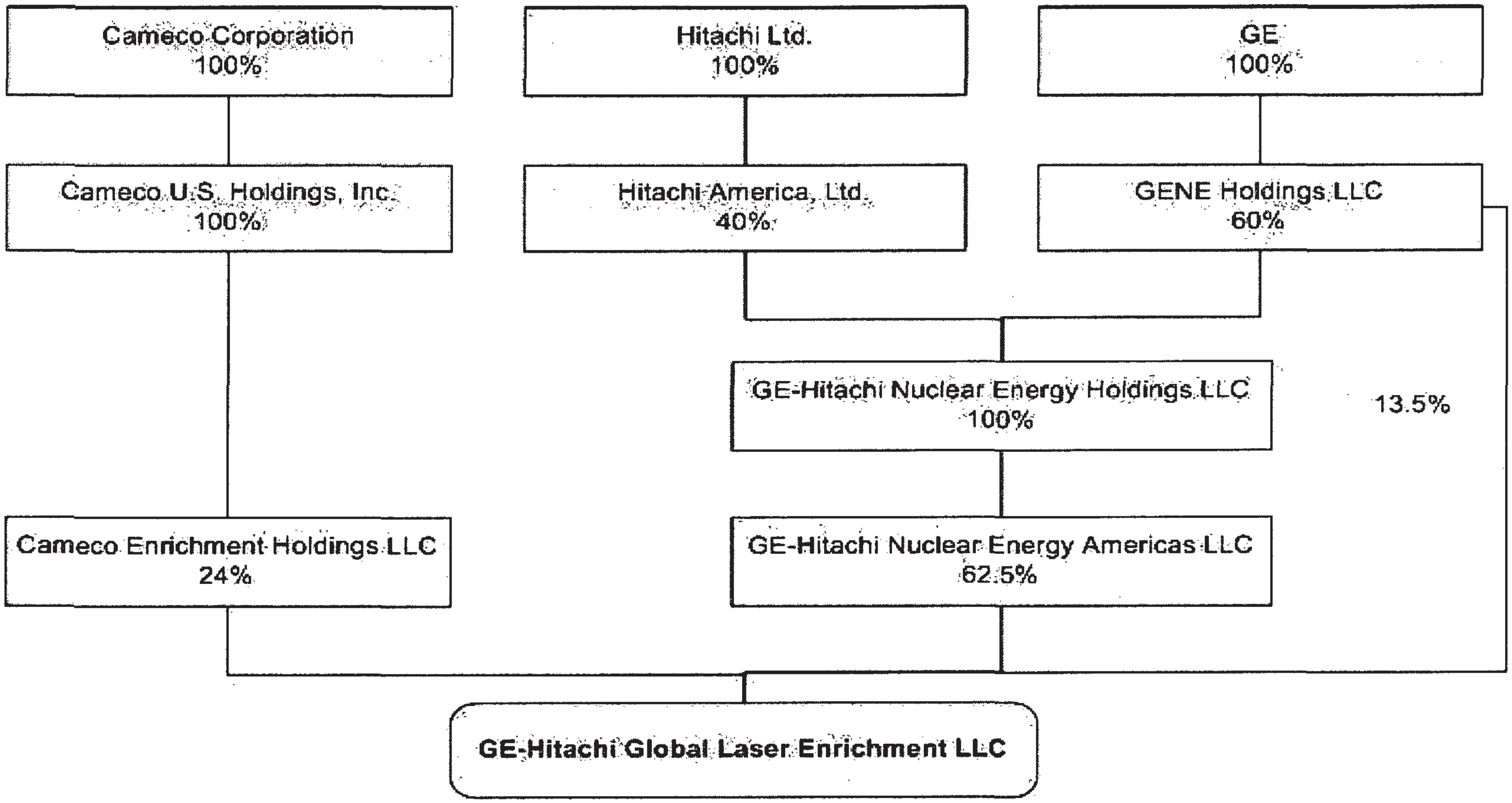
Figure 1-3. {{{Proprietary Information withheld from disclosure per 10 CFR 2.390}}}

Figure 1-4. {{{Proprietary Information withheld from disclosure per 10 CFR 2.390}}}

Figure 1-5. GLE Ownership.



GE Indirect Membership Interest:	51%	(60% x 62.5% + 13.5%)
Hitachi, Ltd. Indirect Membership Interest:	25%	(40% x 62.5%)
Cameco Corporation Indirect Membership Interest:	24%	

Figure 1-6. Community Characteristics Near the Wilmington Site.

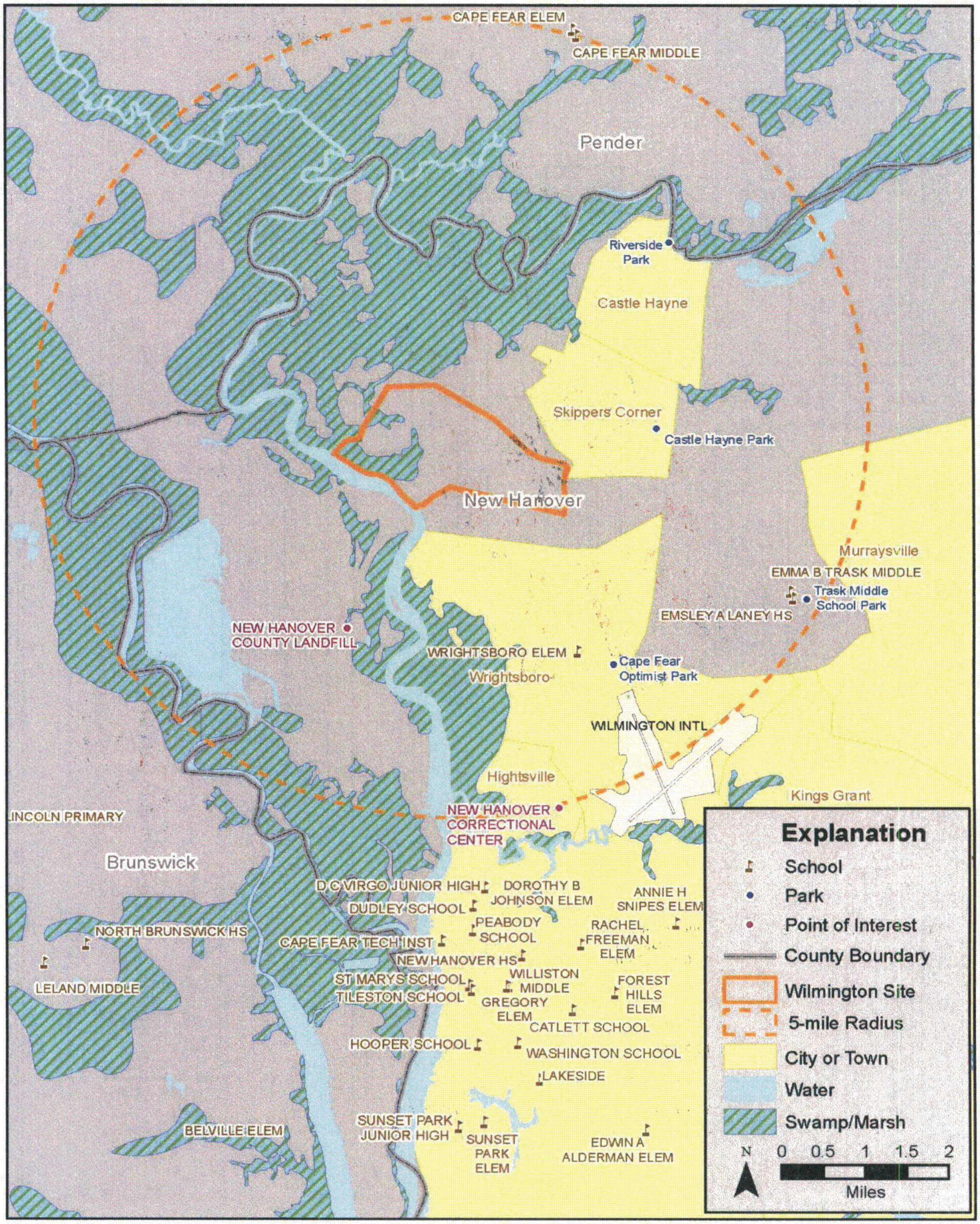
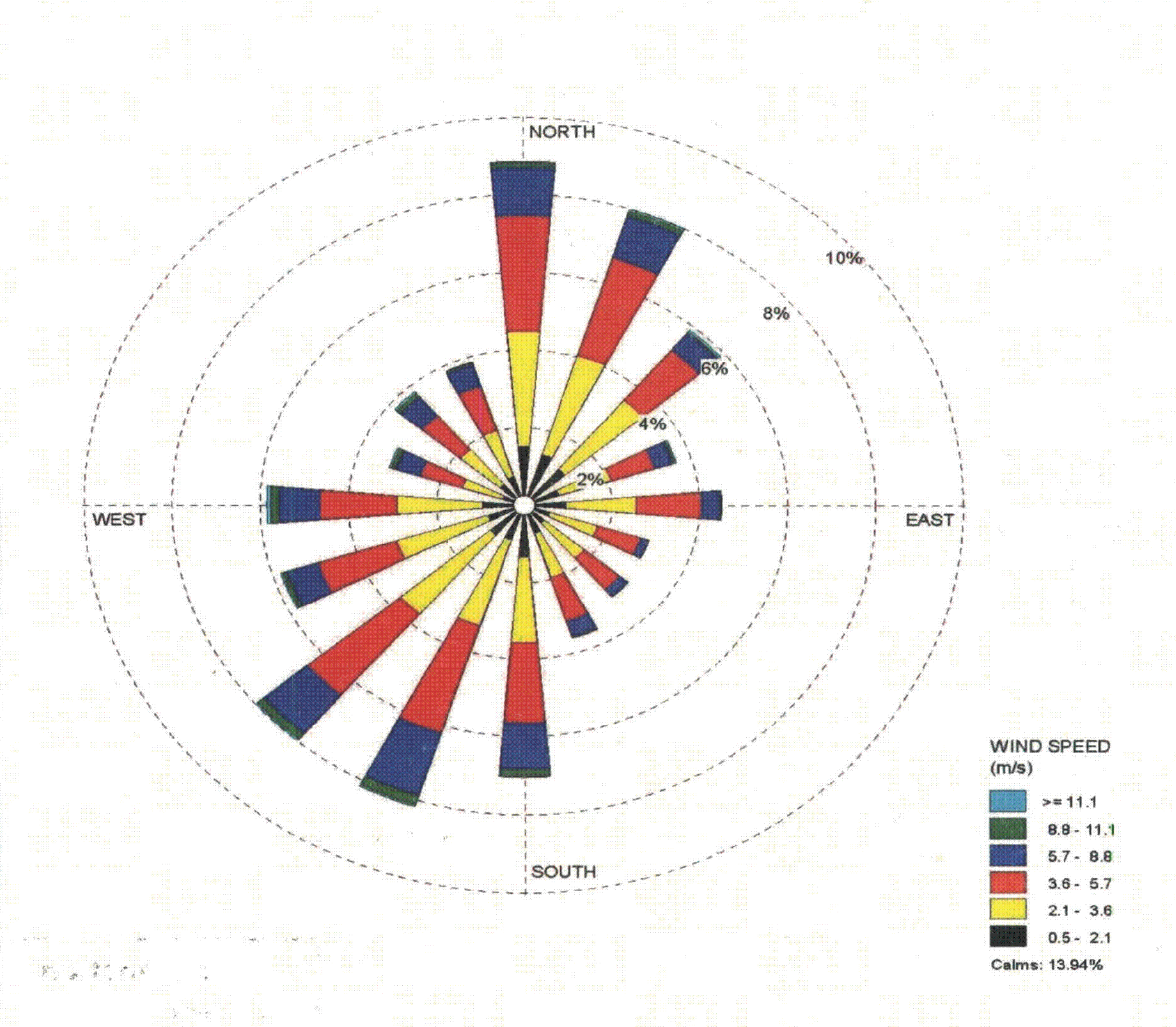


Figure 1-7. Wind Rose for Wilmington International Airport.

WIND ROSE PLOT:
Station #13748 - WILMINGTON/NEW HANOVER COUNTY, NC

DISPLAY:
Wind Speed
Direction (blowing from)



COMMENTS: Source of Data: DS-3505.	DATA PERIOD: **1990-2006** **Jan 1 - Dec 31** **00:00 - 23:00**		
	CALM WINDS: **13.94%**	TOTAL COUNT: **143191 hrs.**	
	AVG. WIND SPEED: **3.57 m/s**	DATE: **7/7/2007**	PROJECT NO.:

WRPLOT View - Lakes Environmental Software

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APPENDIX A -

GUIDELINES FOR DECONTAMINATION OF FACILITIES AND EQUIPMENT PRIOR TO RELEASE FOR UNRESTRICTED USE OR TERMINATION OF LICENSES FOR BYPRODUCT, SOURCE, OR SPECIAL NUCLEAR MATERIAL

U.S. Nuclear Regulatory Commission
Division of Fuel Cycle Safety
and Safeguards
Washington, DC 20555
April 1993

APPENDIX A

The instructions in this guide, in conjunction with Table 1, specify the radionuclides and radiation exposure rate limits which should be used in decontamination and survey of surfaces or premises and equipment prior to abandonment or release for unrestricted use. The limits in Table 1 do not apply to premises, equipment, or scrap containing induced radioactivity for which the radiological considerations pertinent to their use may be different. The release of such facilities or items from regulatory control is considered on a case-by-case basis.

1. The licensee shall make a reasonable effort to eliminate residual contamination.

2. Radioactivity on equipment or surfaces shall not be covered by paint, plating, or other covering material unless contamination levels, as determined by a survey and documented, are below the limits specified in Table 1 prior to the application of the covering. A reasonable effort must be made to minimize the contamination prior to use of any covering.

3. The radioactivity on the interior surfaces of pipes, drain lines, or ductwork shall be determined by making measurements at all traps, and other appropriate access points, provided that contamination at these locations is likely to be representative of contamination on the interior of the pipes, drain lines, or ductwork. Surfaces of premises, equipment, or scrap which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement shall be presumed to be contaminated in excess of the limits.

4. Upon request, the Commission may authorize a licensee to relinquish possession or control of premises, equipment, or scrap having surfaces contaminated with materials in excess of the limits specified. This may include, but would not be limited to, special circumstances such as razing of buildings, transfer of premises to another organization continuing work with radioactive materials, or conversion of facilities to a long-term storage or standby status. Such requests must:

    a. Provide detailed, specific information describing the premises, equipment or scrap, radioactive contaminants, and the nature, extent, and degree of residual surface contamination.

    b. Provide a detailed health and safety analysis which reflects that the residual amounts of materials on surface areas, together with other considerations such as prospective use of the premises, equipment, or scrap, are unlikely to result in an unreasonable risk to the health and safety of the public.

5. Prior to release of premises for unrestricted use, the licensee shall make a comprehensive radiation survey which establishes that contamination is within the limits specified in Table 1. A copy of the survey report shall be filed with the Division of Fuel Cycle Safety and Safeguards, U. S. Nuclear Regulatory Commission, Washington, DC 20555, and also the Administrator of the NRC Regional Office having jurisdiction. The report should be filed at least 30 days prior to the planned date of abandonment. The survey report shall:

   a. Identify the premises.

   b. Show that reasonable effort has been made to eliminate residual contamination.

   c. Describe the scope of the survey and general procedures followed.

   d. State the findings of the survey in units specified in the instruction.

   Following review of the report, the NRC will consider visiting the facilities to confirm the survey.

APPENDIX A

TABLE 1
ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDES[a]	AVERAGE[bcf]	MAXIMUM[bdf]	REMOVABLE[bef]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α/ 100 cm^2	15,000 dpm α/ 100 cm^2	1,000 dpm α/ 100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1000 dpm/100 cm^2	3000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5,000 dpm βγ/ 100 cm^2	15,000 dpm βγ/ 100 cm^2	1,000 dpm βγ/ 100 cm^2

[a]Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b]As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c]Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d]The maximum contamination level applies to an area of not more than 100 cm^2.

[e]The amount of removable radioactive material per 100 cm2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

[f]The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/hr at 1 cm and 1.0 mrad/hr at 1 cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.

CHAPTER 2
REVISION LOG

Rev.	Effective Date	Affected Pages	Revision Description
0	04/30/2009	ALL	Initial Application Submittal.
1	03/31/2010	ALL	Incorporate RAI responses submitted to the NRC via MFN-09-802 dated 12/28/2009.
2	06/18/2010	8, 15	Added details regarding the transition from design/construction to operations and the concurrent construction and operations phases. Added the word "overseeing" to the designated responsibilities of the NCS function.
3	10/29/2010	19, 20	Incorporate RAI responses from October 5, 2010 RAI letter. Added a description of the Regulatory Affairs General Manager, added review of 70.72 changes to FSRC responsibilities.
4	03/30/2011	18	Added bullet to Fire Safety Manager's responsibilities.

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FIGURES

2. ORGANIZATION AND ADMINISTRATION

This chapter of the GE-Hitachi Global Laser Enrichment LLC (GLE) Commercial Facility License Application (LA) presents the organizations responsible for managing the design, construction, operation, and decommissioning of the GLE Commercial Facility. Key management and supervisory positions and functions are described, including personnel qualifications for each key position. This chapter also describes the management system and administrative procedures for effective implementation of Environmental, Health, and Safety (EHS) functions at the GLE Commercial Facility.

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

2.1 ORGANIZATIONAL STRUCTURE

2.1.1 Corporate Functions, Responsibilities, and Authority

GLE supports the national energy security goal of maintaining a reliable and secure domestic source of enriched uranium. GLE uses the laser-based technology, which represents a cost-effective and efficient technology for the enrichment of uranium for domestic and foreign nuclear power plants.

GLE is a limited liability corporation formed to provide uranium enrichment services for commercial nuclear power plants. The GLE partnership is described in GLE LA Section 1.2, *Institutional Information*. GLE's immediate parent company, GE-Hitachi Nuclear Energy Americas LLC (GEH), is the parent company of U.S. Nuclear Regulatory Commission (NRC) licensees whom are licensed under 10 CFR 50, *Domestic Licensing of Production and Utilization Facilities (Ref. 2-1)*, 10 CFR 70, *Domestic Licensing of Special Nuclear Material (Ref. 2-2)*, and 10 CFR 72, *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor Related Greater Than Class C Waste (Ref. 2-3)*, at facilities in Sunol, California; Wilmington, North Carolina; and Morris, Illinois. The GLE President and CEO receives direction from the GLE parent company GE-Hitachi Nuclear Energy Americas through the GEH Fuel Cycle Senior Vice President.

The GLE President and CEO provides overall direction and management with respect to design, construction, operation, and decommissioning activities. Figure 2-1, *GLE Organizational Structure During Design and Construction*, details the organization of GLE during design and construction. Figure 2-2, *GLE Organizational Structure During Operations*, details the organization of GLE during operations.

2.1.2 GLE Design and Construction Organizational Structure

As the owner and operator, GLE is responsible for the design, construction, operation, maintenance, modification, and testing of the GLE Commercial Facility. The GLE President and CEO is responsible for ensuring the facility complies with applicable regulatory requirements and establishing the basic policies of the Quality Assurance (QA) Program. These polices are described in the Quality Assurance Program Description (QAPD) document, are transmitted to all levels of management, and are implemented through approved written policies, plans, and/or procedures.

The lines of communication of key management positions during design and construction are shown in Figure 2-1. The GLE EHS and QA organizations support the GLE Projects Manager; however, the functions are independent allowing for objective audit, review, and control activities.

The GLE Projects Manager is responsible for managing the design, construction, initial startup, and procurement activities. In addition to managing A/E and construction contracts, the GLE Projects Manager also manages a group of Project Managers and the Project Controls Manager. The Project Managers are responsible for implementing procurement, construction, engineering, project engineering, project controls, and startup.

The Engineering Manager is the design authority and is responsible for developing the conceptual design for the facility, which includes the development of design requirements, design bases, and design criteria for the enrichment process and supporting systems. An architect/engineering (A/E) firm has been contracted to further specify structures and systems, as well as to ensure the design meets applicable U.S. codes and standards. A contractor specializing in site evaluations has been contracted to perform the site evaluation. Nuclear consultants have been contracted to support the Integrated Safety Analysis (ISA) and development of the LA. During the construction phase, preparation of construction documents, in addition to construction itself, is completed utilizing qualified contractors. The GLE QA function reviews and approves contractor QA Programs. Approval of contractor QA Programs shall be obtained prior to commencing work activities.

The reporting lines and qualifications of the principal managers for design and construction of the facility are as follows:

The QA and Infrastructure Program Manager reports directly to the GLE President and CEO. The QA and Infrastructure Program Manager shall have, as a minimum, a bachelor's degree in an engineering or scientific field and four (4) years of supervisory nuclear experience in the implementation of a QA Program. The QA and Infrastructure Program Manager shall have at least two (2) years of experience in a QA organization at a nuclear facility.

The Operations Manager reports directly to the GLE President and CEO. The Operations Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four (4) years of related nuclear experience.

The Engineering Manager reports directly to the GLE President and CEO. The Engineering Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and a minimum of five (5) years of related nuclear experience in implementing and supervising a nuclear engineering program.

The GLE Projects Manager reports directly to the GLE President and CEO. The GLE Projects Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field, five (5) years of nuclear experience, and three (3) years of supervisory or management experience.

The Security Manager reports directly to the GLE President and CEO. The Security Manager shall have, as a minimum, a bachelor's degree (or equivalent) in a related field and five (5) years of related experience; or a high school diploma with eight (8) years of related experience.

The GLE EHS Manager reports directly to the GLE President and CEO. The GLE EHS Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and five (5) years of management experience in assignments involving regulatory activities. The manager of the GLE EHS function shall have experience in the understanding and management of Nuclear Criticality Safety (NCS), Environmental Protection, and Industrial Safety programs.

2.1.3 Operations Organizational Structure

The GLE organizational structure during operations is shown in Figure 2-2. GLE has direct responsibility for preoperational testing, initial startup, operation, and maintenance of the GLE Commercial Facility. The GLE Facility Manager reports to the GLE President and CEO and is responsible for the overall operation, administration, and regulatory compliance of the GLE Commercial Facility. In the discharge of these responsibilities, the GLE Facility Manager directs the activities of the following: QA, Operations, Engineering, Projects, Security and Emergency Preparedness, Infrastructure Programs, EHS, and the Facility Safety Review Committee (FSRC).

The responsibilities, authorities, and lines of communication of key management positions within the Operations Organization are discussed in Section 2.2, *Key Management Positions, Responsibilities, and Qualifications.*

During operations, the QA Manager reports to the GLE Facility Manager; however, the QA Manager has the authority and responsibility to directly contact the GLE President and CEO with any QA concerns during operations. Likewise, the GLE EHS Manager has the authority and responsibility to directly contact the GLE President and CEO with any EHS concerns during operations.

2.1.4 Transition from Design and Construction to Operations

GLE is responsible for the design, QA, construction, testing, initial startup, operation, and decommissioning of the GLE Commercial Facility. When the end of Phase 1 construction (initial construction of 1 MSWU facility) approaches, the focus of the organization will shift from design and construction to initial startup and operation. As Phase 1 facility construction nears completion, GLE will staff the Operations organization to ensure a smooth transition from Phase 1 construction activities (managed by the Projects team) to operation activities (managed by the Operations team). During this transition, the GLE EHS Manager position reports directly to the GLE President and CEO (as shown in Figure 2-1) for EHS matters related to design and construction and reports directly to the GLE Facility Manager (as shown in Figure 2-2) for EHS matters related to operations. This position is intentionally duplicated to provide significant continued focus on the EHS goals during design and construction when the Operating organization is not yet fully developed and implemented. Similarly, the QA Manager position is duplicated during the transition from design and construction to operations to ensure quality is adequately maintained throughout the transition phase. The Projects team will continue to manage the construction that occurs during Phase 2 construction (Construction and Component Installation to Ramp-up to 6 MSWU). Similar transitions from the Projects team to the Operations team will occur during each ramp up period. Likewise, the EHS and Quality functions will have active roles in each ramp up period in order to provide continuous facility oversight.

As the construction of systems is completed, the systems undergo acceptance testing as required by approved written policies, plan, and/or procedures. Following successful completion of acceptance testing, systems are transferred from the Projects organization to the Operations organization by means of a detailed transition plan. The transition plan will describe individual roles and responsibilities, and provide task assignments to ensure that the facility remains in compliance during the transition. The transition plan will be available to the NRC upon request. The turnover includes the physical systems, corresponding design information, and records. Following turnover, the Operations organization is responsible for system maintenance. The design basis for the facility is maintained during the transition from construction to operations through the CM Program described in GLE LA Chapter 11, *Management Measures*.

2.2 KEY MANAGEMENT POSITIONS, RESPONSIBILITIES, AND QUALIFICATIONS

This section describes the key functional positions responsible for managing the safe operation of the GLE Commercial Facility. The responsibilities, authorities, and lines of communication for each key management position are provided in this section. Management responsibilities, supervisory responsibilities, and NCS engineering staff responsibilities related to NCS are in accordance with American National Standards Institute (ANSI)/American Nuclear Society (ANS)-8.19-2005, *Administrative Practices for Nuclear Criticality Safety (Ref. 2-4).*

Responsibilities, authorities, and inter-relationships of the GLE organizational groups with responsibilities important to safety are specified in approved written position descriptions and procedures.

Individuals who do not meet the qualification requirements described in this section are not automatically eliminated from a position if other factors provide sufficient demonstration of their abilities to fulfill the duties of the position. These factors shall be evaluated on a case- by-case basis, and approved and documented by the GLE Facility Manager.

2.2.1 Global Laser Enrichment President and Chief Executive Officer

The GLE President and CEO is responsible for providing overall direction and management of GLE activities. The GLE President and CEO is also responsible for maintaining the basic policies of the QA Program, and ensuring those policies are transmitted to all levels of management and implemented appropriately through approved written procedures.

The GLE President and CEO shall have, as a minimum, a bachelor's degree (or equivalent) and five (5) years of related experience. The GLE President and CEO receives direction from the GLE parent company GE-Hitachi Nuclear Energy Americas through the GEH Fuel Cycle Senior Vice President.

2.2.2 Global Laser Enrichment Facility Manager

The GLE Facility Manager reports to the GLE President and CEO and is the individual with the overall responsibility for safety and activities conducted at the GLE Commercial Facility. The activities of the GLE Facility Manager are performed in accordance with GLE's policies, plans, procedures, and work instructions. The GLE Facility Manager provides for safety, control of operations, and protection of the environment by delegating and assigning responsibility to qualified line management and area managers.

The GLE Facility Manager shall have, as a minimum, a bachelor's degree in an engineering or scientific field and four (4) years of experience in nuclear facility operations. The GLE Facility Manager shall be knowledgeable of the safety program concepts as applied to the overall safety of the facility, and has the authority to enforce the shutdown of any process or facility. The GLE Facility Manager must approve restart of an operation that he/she directs to be shutdown.

2.2.3 Quality Assurance Organization

2.2.3.1 Quality Assurance Manager

The GLE QA Manager reports to the GLE Facility Manager and is responsible for establishing and maintaining the GLE QA Program and the Laboratory Services Program. Line management and their staff, who are responsible for performing quality-affecting work, are responsible for ensuring implementation of and compliance with the GLE QA Program. The QA Manager position is independent from other management positions at the facility to ensure the QA Manager has access to the GLE Facility Manager for matters affecting quality. In addition, the QA Manager has the authority and responsibility to contact the GLE President and CEO with any QA concerns. The QA Manager has the authority to stop work based on quality concerns. This authority to stop work and the process to resume stopped work is documented in approved policies, plans, and/or procedures.

The QA Manager shall have, as a minimum, a bachelor's degree in an engineering or scientific field and four (4) years of supervisory nuclear experience in the implementation of a QA Program. The QA Manager shall have a minimum of two (2) years of experience in a QA organization at a nuclear facility.

2.2.3.2 Laboratory Services Manager

The Laboratory Services Manager reports to the QA Manager and has the responsibility for the implementation of chemistry analysis and laboratory programs and procedures for the GLE Commercial Facility. The Laboratory Services Manager's responsibilities typically include, but are not limited to, chemical analysis of samples and maintaining the laboratories.

The Laboratory Services Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and three (3) years of related nuclear experience associated with implementation of a chemistry program.

2.2.4 Operations Organization

2.2.4.1 Operations Manager

The Operations Manager reports to the GLE Facility Manager and has the responsibility of directing the day-to-day operation of the facility. This includes activities such as ensuring the correct and safe operation of uranium hexafluoride (UF_6) processes, proper handling of UF_6, and the identification and mitigation of any off-normal operating conditions.

The Operations Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four (4) years of related nuclear experience.

2.2.4.2 Maintenance Manager

The Maintenance Manager reports to the Operations Manager and has the responsibility of directing and scheduling maintenance activities to ensure proper operation of the facility. Other Maintenance Manager responsibilities typically include, but are not limited to, activities such as: corrective and preventive maintenance of facility equipment; preparation and implementation of maintenance procedures; and coordinating and maintaining testing programs for the facility, to include testing of systems, structures, and components (SSCs) to ensure the SSCs are functioning as specified in design documents.

The Maintenance Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four (4) years of related nuclear experience.

2.2.4.3 Production Control Manager

The Production Control Manager reports to the Operations Manager and is responsible for developing and maintaining production schedules for enrichment services.

The Production Control Manager shall have, as a minimum, a bachelor's degree (or equivalent) in a technical field and three (3) years of experience in operations; or a high school diploma and five (5) years of operations experience.

2.2.4.4 Integrated Safety Analysis Manager

The ISA Manager reports to the Operations Manager. ISA Manager responsibilities typically include, but are not limited to, maintaining the ISA program; identifying items relied on for safety (IROFS); identifying the management measures and QA elements to be applied to safety controls; and providing advice and counsel to area managers on matters of the ISA program.

The ISA Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four (4) years of related experience. The ISA Manager shall have experience in the understanding and management of the assigned programs.

2.2.4.5 Configuration Management Manager

The CM Manager reports to the Operations Manager and is responsible for establishing and maintaining a CM Program for uranium enrichment equipment and safety controls, including related record retention.

The CM Manager shall have, as a minimum, a bachelor's degree (or equivalent) and two (2) years of related experience; or a high school diploma with eight (8) years of related experience. The CM Manager shall have experience in the understanding and management of the assigned programs.

2.2.4.6 Area Managers

Area managers report to the Operations Manager. Area managers are the designated individuals responsible for ensuring activities necessary for safe operations and protection of the environment are conducted properly, within their assigned area(s) of the facility, in which uranium materials are processed, handled, or stored. Designated area manager responsibilities typically include, but are not limited to, the following:

- Assure safe operation, maintenance, and control of activities;
- Assure safety of the environs as influenced by operations;
- Assure performance of ISA for the assigned facility area, as required;
- Assure application of management measures and QA elements to safety controls, as appropriate;
- Assure configuration control for Items Relied on for Safety (IROFS) in the assigned facility area, as required;
- Ensure use of approved written procedures which incorporate safety controls and limits; and
- Provide adequate operator training.

The area managers shall have, as a minimum, a bachelor's degree (or equivalent) in a technical field, and two (2) years of experience in operations, one of which is in fuel cycle facility operations; or a high school diploma with five (5) years of operations experience, two of which are in fuel cycle facility operations. Area managers shall be knowledgeable of the safety program procedures (including Industrial Safety, Radiation Protection [RP], Fire Safety, NCS, and Environmental Protection) and shall have experience in the application of the program controls and requirements, as related to their assigned area of responsibility. The GLE Facility Manager shall approve the assignment of individuals to the position of area manager. A listing of area managers, by area of responsibility, shall be maintained current at the facility.

2.2.4.7 Shift Supervisors

Shift supervisors report to the Operations Manager and are the interface between management and facility operators. Designated shift supervisor responsibilities typically include, but are not limited to, the following:

- Provide day-to-day work direction to operators and other assigned workers;
- Assure safe operation and control of activities;
- Assure adherence to approved written procedures and controls;
- Provide adequate operator oversight and guidance; and
- Identify and communicate off-normal conditions.

The shift supervisors shall have, as a minimum, a high school diploma and three (3) years of experience in a technical field. Shift supervisors shall be knowledgeable of the applicable safety program procedures (including Industrial Safety, RP, Fire Safety, NCS, and Environmental Protection).

2.2.5 Engineering Organization

2.2.5.1 Engineering Manager

The Engineering Manager reports to the GLE Facility Manager and is the design authority. The Engineering Manager has the responsibility for providing engineering support for the GLE Commercial Facility. The responsibilities of the Engineering Manager include, but are not limited to, ensuring the safe operation of enrichment and support equipment; providing maintenance support for equipment and systems; and supporting the development of operating and maintenance procedures. The Engineering Manager is responsible for the development of design changes to the facility.

The Engineering Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and five (5) years of related nuclear experience in implementing and supervising a nuclear engineering program.

2.2.6 GLE Projects Organization

2.2.6.1 GLE Projects Manager

The GLE Projects Manager reports to the GLE President and CEO and has the responsibility for the implementation of facility modifications, and provides engineering support, as needed, to support operations, maintenance, and performance testing of systems and equipment. The GLE Projects Manager is also responsible for managing remaining design and construction activities. The GLE Projects Manager manages a group of Project Managers and a Project Controls Manager. The GLE Projects Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field, five (5) years of nuclear experience, and three (3) years of supervisory or management experience.

2.2.7 Security and Emergency Preparedness Organization

2.2.7.1 Security and Emergency Preparedness Manager

The Security and Emergency Preparedness functions are administratively independent of Operations. The Security and Emergency Preparedness Manager reports to the GLE President and CEO and has designated responsibilities that typically include, but are not limited to, the following:

- Direct the activities of security personnel to ensure the physical protection of the GLE Commercial Facility and GLE Site;
- Protection of classified matter at the facility and obtaining security clearances for facility personnel and support personnel;
- Establish and maintain the Emergency Preparedness Program, to include training and program evaluations;
- Provide advice and counsel to area managers on matters of security and emergency preparedness; and
- Maintain agreements and preparedness with offsite emergency support groups.

The Security and Emergency Preparedness Manager shall have, as a minimum, a bachelor's degree (or equivalent) in a related field and five (5) years of related experience; or a high school diploma with eight (8) years of related experience.

2.2.8 Infrastructure Programs Organization

2.2.8.1 Infrastructure Programs Manager

The Infrastructure Programs Manager reports to the GLE Facility Manager and has the responsibility of providing business and administrative support to the GLE Commercial Facility. The Infrastructure Programs Manager's responsibilities typically include, but are not limited to, Document Control, Records Management, Training, and Administrative Functions.

The Infrastructure Program Manager shall have, as a minimum, a bachelor's degree (or equivalent) in a related field, and three (3) years of related experience in implementing and supervising administrative responsibilities at a nuclear facility.

2.2.8.2 Document Control and Records Management Manager

The Document Control and Records Management Manager reports to the Infrastructure Programs Manager and has the responsibility for establishing and maintaining a Document Control System for adequately controlling documentation and a Records Management System to adequately control QA Records in accordance with the Quality Assurance Program Description.

The Document Control and Records Management Manager shall have, as a minimum, a bachelor's degree (or equivalent) and three (3) years of related experience in implementing and supervising a document control or records management program.

2.2.8.3 Training Manager

The Training Manager reports to the Infrastructure Programs Manager and is responsible for establishing and maintaining the Training Program as well as maintaining training records for personnel at the facility.

The Training Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four (4) years of related experience.

2.2.9 Global Laser Enrichment Environmental, Health, and Safety Organization

The GLE EHS function is administratively independent of Operations but has the authority to enforce the shutdown of any process or facility in the event that controls for any aspect of safety are not assured.

2.2.9.1 *Global Laser Enrichment Environmental, Health, and Safety Manager*

The GLE EHS Manager reports to the GLE Facility Manager. In addition, the GLE EHS Manager has the authority and responsibility to contact the GLE President and CEO with any EHS concerns. The GLE EHS Manager has designated overall responsibility to establish and manage the NCS, Industrial Safety, Material Control and Accounting (MC&A), RP, Environmental Protection, and Fire Safety Programs to ensure compliance with applicable federal, state, and local regulations and laws. These programs are designed to ensure the health and safety of employees and the public, as well as the protection of the environment. The GLE EHS Manager must approve restart of any operation shutdown by the EHS function.

The GLE EHS Manager works with the other facility managers to ensure consistent interpretations of EHS requirements, performs independent reviews, and supports facility and operations change control reviews. This position is independent from other management positions at the facility to ensure objective EHS audit, review, and control activities. The EHS Manager has the authority to issue stop work orders and must be consulted prior to resumption of stopped work. Changes to the facility or to activities of personnel that require prior NRC approval are reviewed and approved by the EHS Manager or designee.

The GLE EHS Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and five (5) years of management experience in assignments involving regulatory activities. The manager of the GLE EHS function shall have experience in the understanding and management of NCS, Environmental Protection, and Industrial Safety programs.

2.2.9.2 *Nuclear Criticality Safety Function*

The NCS function is administratively independent of Operations and has the authority to shutdown potentially unsafe operations. The NCS Manager reports to the GLE EHS Manager and must approve restart of any operation shutdown by the NCS function. Designated responsibilities of the NCS Manager typically include, but are not limited to, overseeing the following:

- Establish the NCS program, to include design criteria, procedures, and training;
- Provide NCS support for operations including ISAs and configuration control;
- Assess normal and credible abnormal conditions;
- Determine NCS limits for controlled parameters;
- Perform methods development and validation to support NCS analyses;
- Perform neutronics calculations, develop Criticality Safety Analyses (CSAs), and approve proposed changes in process conditions or equipment involving fissionable material;
- Specify NCS control requirements and functionality;
- Provide advice and counsel to area managers on NCS control measures, to include review and approval of operating procedures;
- Support emergency response planning and events; and
- Assess the effectiveness of the NCS program through audit programs.

The NCS Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field, four (4) years of experience in assignments involving regulatory activities, and experience in the understanding, application, and direction of NCS programs.

A Senior Engineer within the NCS function shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field with three (3) years of nuclear-related experience in criticality safety. A senior engineer shall have experience in the assigned safety function, and has the authority and responsibility to conduct activities assigned to the NCS function.

An Engineer within the NCS function shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and experience in the assigned safety function. An NCS Engineer shall have the authority and responsibility to conduct activities assigned to the NCS function with the exception of independent verification of NCS analyses.

2.2.9.3 Material Control and Accounting Manager

The MC&A Manager reports to the GLE EHS Manager and has the responsibility for proper implementation and control of the Fundamental Nuclear Material Control Plan (FNMCP). This position is separate from, and independent of, the Operations and Engineering Organizations to ensure a definite division between the MC&A function and the other organizations.

The MC&A Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and five (5) years of experience in the management of a safeguards program for special nuclear material (SNM), to include responsibilities for material control and accountability. No credit for academic training may be taken toward fulfilling this experience requirement.

2.2.9.4 Industrial Safety Manager

The Industrial Safety Manager is administratively independent of Operations and has the authority to shutdown operations when potentially hazardous health and safety conditions are identified. The Industrial Safety Manager reports to the GLE EHS Manager and must approve restart of any operation shutdown by the Industrial Safety function. Designated responsibilities of the Industrial Safety Manager typically include, but are not limited to, the following:

- Identify industrial safety requirements from federal, state, and local regulations which govern GLE Commercial Facility operations;
- Ensure proper implementation of the GLE Industrial Safety Program;
- Develop practices regarding non-radiation chemical safety affecting nuclear activities;
- Provide advice and counsel to area managers on matters of industrial safety;
- Ensure proper implementation of the Laser Safety Program;
- Provide consultation and review of new, existing, or revised equipment, processes, and procedures regarding industrial safety; and
- Provide industrial safety support for ISAs and configuration control.

The Industrial Safety Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and two (2) years of experience in related assignments; or a high school diploma and eight (8) years of related experience.

2.2.9.5 Environmental Protection Function

The Environmental Protection Manager is administratively independent of Operations and has the authority to shutdown operations with potentially adverse environmental impacts. The Environmental Protection Manager reports to the GLE EHS Manager and must approve restart of any operation shutdown by the Environmental Protection function. Designated responsibilities of the Environmental Protection Manager typically include, but are not limited to, the following:

- Identify Environmental Protection requirements from federal, state, and local regulations which govern the facility operation;
- Establish systems and methods to measure and document adherence to regulatory Environmental Protection requirements and license conditions;
- Provide advice and counsel to area managers on matters of Environmental Protection;
- Evaluate and approve new, existing, or revised equipment, processes, and procedures involving Environmental Protection activities;
- Provide Environmental Protection support for ISAs and configuration control; and
- Assure proper federal and state permits, licenses, and registrations are obtained for non-radiation discharges from the GLE Commercial Facility.

The Environmental Protection Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and two (2) years of experience in assignments involving regulatory activities (or equivalent); or a high school diploma and eight (8) years of experience in assignments involving regulatory activities.

2.2.9.6 Radiation Protection Function

The RP function is administratively independent of Operations and has the authority to shutdown potentially unsafe operations. The RP Manager reports to the GLE EHS Manager and is responsible for overseeing the training program for training personnel in radiation protection policies, plans, and/or procedures. The RP Manager is responsible for establishing the initial training program, and as stated in GLE LA Section 4.5.5, reviews the contents of the training program every two years. The RP Manager must approve restart of any operation shutdown by the RP function. Designated responsibilities for the RP Manager typically include, but are not limited to, the following:

- Establish and maintain the RP Programs, procedures, and training;
- Evaluate radiation exposures of employees and visitors, and ensure the maintenance of related records;
- Conduct radiation and contamination monitoring and control programs;
- Evaluate the integrity and reliability of radiation detection instruments;
- Provide RP support for ISAs and configuration control;

- Provide advice and counsel to area managers on matters of RP;
- Support emergency response planning; and
- Assess the effectiveness of the RP Program through audit programs.

The RP Manager shall have, as a minimum, a bachelor's degree in an engineering or scientific field, three (3) years of experience that includes assignments involving responsibility for RP, and experience in the understanding, application, and direction of RP Programs.

A senior engineer of the RP function shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and two (2) years of nuclear industry experience in the assigned function. Alternate minimum experience qualification for a senior member of the RP function is a professional certification in health physics. A senior member shall have experience in the assigned safety function, and has authority and responsibility to conduct activities assigned to the RP function.

2.2.9.7 Fire Safety Function

The Fire Safety function is administratively independent of Operations and has the authority to shut down operations when imminent hazardous fire safety conditions are identified. The Fire Safety Manager reports directly to the GLE EHS Manager and must approve restart of any operation shutdown by the Fire Safety function. Designated responsibilities of the Fire Safety Manager typically include, but are not limited to, the following:

- Identify fire protection requirements from federal, state, and local regulations which govern GLE Commercial Facility operations;
- Perform an annual review of Consumer Products Safety Commission website for identification of recall data associated with fire protection basic components;
- Ensure proper implementation of the GLE Fire Protection Program and ensure performance of fire protection systems is maintained;
- Manage a staff composed of trained personnel with experience in fire protection;
- Manage the GLE Commercial Facility Fire Brigade;
- Ensure inspection, testing, and maintenance of fire protection systems, features, and equipment is conducted;
- Develop practices regarding fire safety affecting nuclear activities;
- Provide advice and counsel to area managers on matters of fire safety;
- Provide consultation and review new, existing, or revised equipment, processes, and procedures regarding fire safety; and
- Provide fire safety support for ISA and configuration management activities.

The Fire Safety Manager shall have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four (4) years of experience in fire protection related assignments.

The Fire Safety Manager staff shall include a licensed fire protection engineer with a minimum of seven (7) years of fire protection related experience.

Additional available support staff shall include a mechanical engineer, electrical engineer, and structural engineer; all with a minimum of four years of fire protection related experience. Operational support staff performing inspection, observation, and training duties shall have a minimum of two (2) years of fire protection experience. Support staff can be available either through direction employment or under contract.

2.2.10 Licensing and Regulatory Affairs Organization

2.2.10.1 Regulatory Affairs General Manager

The Regulatory Affairs General Manager shall have, as a minimum, a technical bachelor's degree (or equivalent) and five years of related experience. The Regulatory Affairs General Manager is responsible for providing leadership and strategic guidance for all the licensing activities in GEH Nuclear Energy (GLE's immediate parent organization) and reports to the President and Chief Executive Officer of GEH Nuclear Energy.

2.2.10.2 Licensing Manager

The Licensing Manager reports operationally to the GLE Facility Manager and functionally to the Regulatory Affairs General Manager. The Licensing Manager has responsibility for coordinating facility activities to ensure compliance with applicable NRC requirements. The Licensing Manager is also responsible for ensuring abnormal events are reported to the NRC in accordance with NRC regulations.

The Licensing Manager shall have, as a minimum, a bachelor's degree (or equivalent) and five (5) years of related experience in implementing and supervising nuclear activities in compliance with NRC regulations and facility license commitments.

2.2.11 Safety Committees

2.2.11.1 Facility Safety Review Committee

The FSRC provides the GLE Facility Manager with an independent overview of the safety of operations, and provides management with guidance relative to involvement in safety risks. The committee shall provide professional advice and counsel on Environmental Protection, NCS, RP, and Industrial Safety issues affecting nuclear activities.

A review of the ALARA program and projects shall be conducted annually. This ALARA review shall consider:

- Programs and projects undertaken by the RP function and the Radiation Safety Committee (RSC);
- Facility changes made per 70.72 process;
- Performance including, but not limited to, trends in airborne concentrations of radioactivity, personnel exposures, and environmental monitoring results; and
- Programs for improving the effectiveness of equipment used for effluent and exposure control.

The FSRC is responsible to the GLE Facility Manager. The committee's proceedings, findings and recommendations are reported in writing to the GLE Facility Manager, appropriate line management, and appropriate area manager(s) responsible for operations. Such reports shall be retained for a minimum of three years.

The committee shall consist of the Chairman and five (5) members, at a minimum. The committee shall include competence in the applicable scientific and engineering disciplines and shall be staffed with members outside of the GLE Operations Organization. The committee shall hold a minimum of three (3) meetings each calendar year with a maximum interval of 180 days between any two consecutive meetings.

2.2.11.2 Radiation Safety Committee

The objective of the RSC is to maintain occupational radiation exposures ALARA through improvements in operations. The committee meets monthly to maintain a continual awareness of the status of projects, performance measurement and trends, and the current radiological safety conditions of site activities. The maximum interval between meetings shall not exceed 60 days. A written report of each RSC meeting is forwarded to the appropriate line management, area managers, and the GLE EHS Manager. Records of the committee proceedings are maintained for a minimum of three (3) years. The committee consists of managers or representatives from key functions with activities affecting radiological safety. GLE LA Chapter 4, *Radiation Protection*, provides further information regarding the RSC.

2.2.11.3 Chemical Review Committee

Before a new chemical is ordered, the requester must obtain approval from the Chemical Review Committee. The Chemical Review Committee is comprised of a representative of the EHS Organization, an area manager, and others as deemed appropriate by the EHS representative. The EHS representative leads the review and is a qualified chemical safety reviewer. The process for approval includes reviewing the health and safety risks of the chemical, as well as appropriate handling, storage, and disposal information. Every effort is made to limit the amount of hazardous chemicals used, including identifying feasible alternative chemicals or processes. GLE LA Chapter 6, *Chemical Process Safety*, provides further information on the Chemical Review Committee.

2.3 MANAGEMENT MEASURES

Management measures for the conduct and maintenance of GLE's EHS Programs are contained in approved written policies, plans, and/or procedures as described in GLE LA Chapter 11. Such practices are part of a Document Control Program, and appropriately span the organizational structure and major facility activities to control interrelationships and specify program objectives, responsibilities, and requirements. Personnel are appropriately trained to the requirements of these management controls, and compliance is monitored through internal and independent audits and assessments. Management measures for IROFS are defined in the individual IROFS Boundary Control Documents.

2.3.1 Configuration Management

CM is provided for IROFS throughout facility design, construction, testing, and operation. CM provides oversight and control of design information, safety information, and records of modifications (both temporary and permanent) that could impact the ability of IROFS to perform their safety functions when needed. The Operations Manager has responsibility for CM. Selected documentation is controlled under the CM Program 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 approved written policies, plans, and/or procedures. See GLE LA Section 11.1, *Configuration Management*, for additional details on CM.

2.3.2 Maintenance

The GLE Maintenance Program shall be implemented for the operations phase of the GLE Commercial 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 when needed. Maintenance activities include: corrective and preventive maintenance, surveillance/monitoring, and functional testing. These maintenance activities are discussed in further detail in GLE LA Section 11.2, *Maintenance*.

2.3.3 Training and Qualifications

Personnel training is conducted, as necessary, to provide reasonable assurance that individuals are qualified and continue to understand and recognize the importance of safety while performing assigned activities. Training is provided for each individual working at the GLE Commercial Facility, commensurate with assigned duties. Training and qualification requirements are met prior to personnel fully assuming the duties of safety-significant positions, and before assigned tasks are independently performed. The system established for training and retraining is described in GLE LA Section 11.3, *Training and Qualifications*.

2.3.3.1 Nuclear Safety Training

GLE training policy requires that employees complete formal nuclear safety training prior to unescorted access to Radiological Controlled Areas (RCAs). Formal training relative to nuclear safety includes, but is not limited to, the following topics:

- Radiation and radioactive materials,

- Risks involved in receiving low-level radiation exposure in accordance with 10 CFR 19.12, *Instruction to Workers (Ref. 2-5)*,
- Basic criteria and practices for RP,
- Industrial safety,
- Maintaining radiation exposures ALARA,
- Maintaining radioactivity in effluents ALARA, and
- Emergency response; and
- Applicable NCS objectives contained in ANSI/ANS-8.19-2005 and ANSI/ANS-8.20-1991, Nuclear Criticality Safety Training (Ref. 2-6).

2.3.3.2 Operator Training

Operator training is performance-based and incorporates the structured elements of analysis, design, development, implementation, and evaluation. Job-specific training includes applicable procedures, safety provisions, and requirements. Emphasis is placed on safety requirements where human actions are important to safety. Operator training and qualification requirements are met prior to safety-related tasks being independently performed or before startup following significant changes to safety controls.

2.3.4 Procedures

GLE Commercial Facility activities are conducted through the use of approved written policies, plans, and/or procedures (herein referred to as procedures). Applicable procedure and training requirements are satisfied before use of any procedure. Approved written procedures are used to control activities to ensure the activities are carried out in a safe manner.

Procedures are categorized as either operating procedures or management control procedures. Operating procedures provide specific direction for task-based work. Management control procedures describe administrative and general facility practices approved and issued by cognizant management at a level appropriate to the scope of the practice. These procedures direct and control activities across the various process functions and assign functional responsibilities and requirements for these activities.

Additional details on the use of procedures, including the preparation of procedures in accordance with the Document Control Program are provided in GLE LA Section 11.4, *Procedures*.

2.3.5 Audits and Assessments

The GLE QA Program requires periodic audits and assessments to confirm activities affecting quality comply with the QA Program and that the QA Program is being implemented effectively. Additional details on audit and assessments are provided in GLE LA Section 11.5, *Audits and Assessments*.

2.3.5.1 Facility Safety Review Committee

The FSRC provides technical and administrative reviews of facility operations that could affect facility and worker safety. The FSRC shall review audit findings and performance, including external inspections, for adequacy and timeliness of corrective actions and for trends or overall weaknesses as indicated by audit findings.

2.3.5.2 Quality Assurance Organization

The QA Organization conducts periodic audits of activities associated with the GLE Commercial Facility to verify the facility's compliance with established procedures.

2.3.5.3 Audited Organization

Audited organizations shall assure that deficiencies identified are corrected in a timely manner. Audited organizations shall transmit a response to each audit report within the time period specified in the audit report. For each identified deficiency, the response shall identify the corrective action taken or to be taken. For each identified deficiency, the responses shall also address whether or not the deficiency is considered to be indicative of other problems (for example, a specific audit finding may indicate a generic problem) and the corrective action taken or to be taken for any such identified problems. Copies of audit reports and responses are maintained in accordance with the Records Management Program.

2.3.6 Incident Investigations

Incident investigations are performed to assure that the upset condition(s) is understood, and appropriate corrective actions are identified and implemented to prevent recurrence. GLE Management measures include documenting process-upset conditions in Unusual Incident Reports (UIRs). UIRs are documented and the associated corrective actions are tracked to completion. The objectives of the incident investigation and reporting procedure(s) are to: establish the validity of the data related to the incident; develop and implement corrective action plans, as appropriate; document an event which was or could become a danger to persons or property; and ensure that proper levels of GLE management and public agencies are notified. Additional details on Incident Investigations are provided in GLE LA Section 11.6, *Incident Investigations*.

2.3.7 Records Management

Approved written procedures that control the process for submittal, receipt, processing, retention, maintenance, and storage of facility documents or records are established. Details on the Records Management Program are provided in GLE LA Section 11.7, *Records Management*.

2.4 EMPLOYEE CONCERNS

GLE is committed to providing a safe and productive work environment that encourages employees to raise issues or concerns related to the design, construction, or operation of the GLE Commercial Facility. 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 (for example, ESH Manager, GLE Facility Manager, QA Manager),
- The facility safety personnel (that is, any of the safety engineers or managers);
- NRC's requirements under 10 CFR 19, Notices, Instructions, and Reports to Workers: Inspection and Investigations (Ref. 2-7).

In addition to the above, GLE has established an employee concerns program to provide an avenue for employees to obtain an independent evaluation of concerns.

GLE Management is committed to investigating and resolving employee concerns in an effective manner and providing timely resolutions to issues. The employee concerns program provides methods for establishing a work environment in which employees feel free to raise concerns to their management or the NRC without fear of reprisal.

2.5 WRITTEN AGREEMENTS WITH OFFSITE EMERGENCY RESOURCES

The plans for responding to emergencies at the GLE Commercial Facility are presented in detail in the Radiological Contingency and Emergency Plan (RC&EP). The RC&EP includes a description of the facility Emergency Response organization and interfaces with offsite emergency response organizations. The RC&EP includes references to agreements with applicable offsite emergency response organizations.

2.6 REFERENCES

2-1. 10 CFR 50, *Domestic Licensing of Production and Utilization Facilities*, U.S. Nuclear Regulatory Commission, 2008.

2-2. 10 CFR 70, *Domestic Licensing of Special Nuclear Material*, U.S. Nuclear Regulatory Commission, 2008.

2-3. 10 CFR 72, *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor Related Greater Than Class C Waste*, U.S. Nuclear Regulatory Commission, 2008.

2-4. ANSI/ANS-8.19-2005, *Administrative Practices for Nuclear Criticality Safety*, American Nuclear Society, January 2005.

2-5. 10 CFR 19.12, *Instruction to Workers*, U.S. Nuclear Regulatory Commission, 2008.

2-6. ANSI/ANS-8.20-1991, *Nuclear Criticality Safety Training*, American Nuclear Society, January 1991.

2-7. 10 CFR 19, *Notices, Instructions, and Reports to Workers: Inspections and Investigations*, U.S. Nuclear Regulatory Commission, 2008.

Figure 2-1. GLE Organizational Structure During Design and Construction.

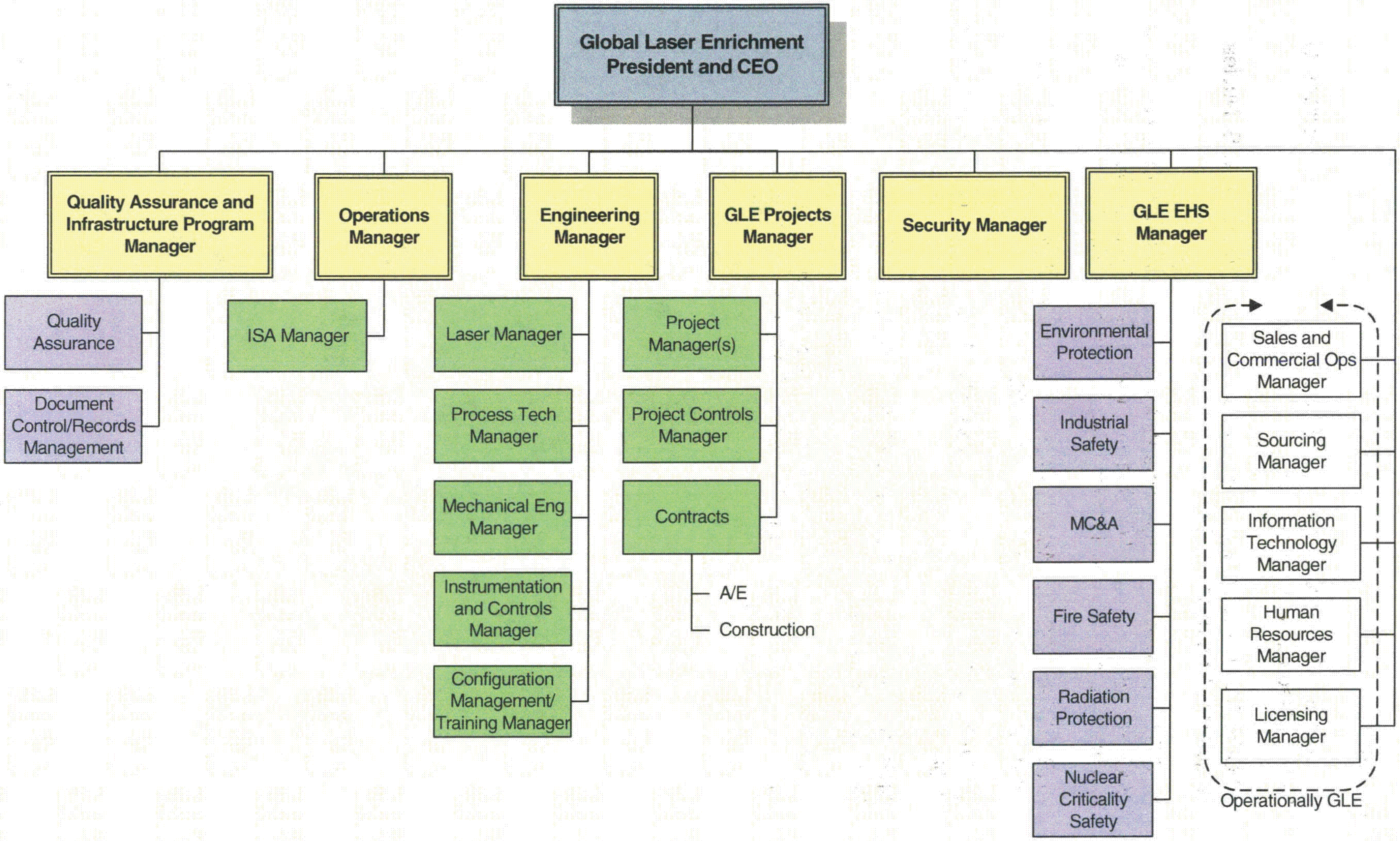
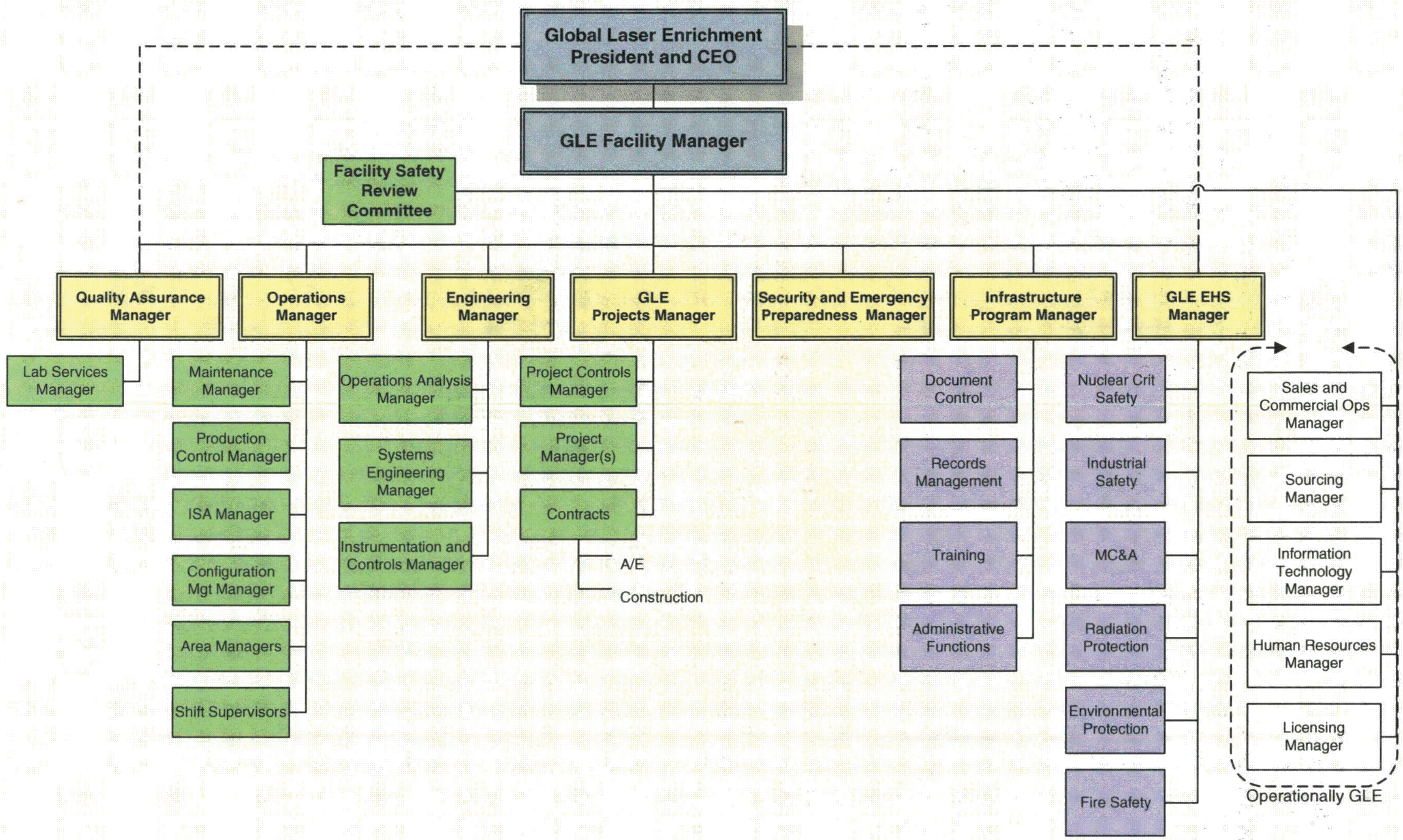


Figure 2-2. GLE Organizational Structure During Operations.



CHAPTER 3
REVISION LOG

Rev.	Effective Date	Affected Pages	Revision Description
0	04/30/2009	ALL	Initial Application Submittal.
1	03/31/2010	8, 9, 16, 23, 25-32	Incorporate RAI responses submitted to the NRC via MFN-09-578 dated 09/04/2009 and MFN-09-802 dated 12/28/2009.
2	06/18/2010	22, 29	Deleted text related to probabilistic risk assessments in QRAs (only done in initial ISA summary, future revisions use both quantitative and qualitative assessments). Added NAVFAC DM 7 to Table 3-1.
3	10/29/2010	27-32	Incorporate RAI responses from NRC letters dated October 5 and October 14, 2010. Updated Table 3-1
4	03/30/2011	23, 27-35	Added discussion on implementation of guidance related to IROFS human factors engineering review. Updated standards and codes listed in Table 3-1.

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TABLES

FIGURES

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3. INTEGRATED SAFETY ANALYSIS (ISA) AND ISA SUMMARY

This chapter presents the GE-Hitachi Global Laser Enrichment LLC (GLE) Integrated Safety Analysis (ISA) commitments and outlines the GLE ISA methodology. The approach used for performing the ISA is based on NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility (Ref. 3-1)*, Chapter 3, Appendix A, Example Procedure for Accident Sequence Evaluation. 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 identifies which accident sequences have consequences that could potentially exceed the performance requirements of 10 CFR 70.61, *Performance Requirements (Ref. 3-2)*; and therefore require a 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 facility and external hazards, credible initiating events, potential accident sequences, the likelihood and consequences of each accident sequence, and the IROFS implemented to prevent or mitigate each credible accident. The ISA Team reviewed the hazard identified for the credible worst-case consequences. Credible high or intermediate consequence accident scenarios were assigned accident sequence identifiers and 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.

The primary scope of the ISA included fires, hazardous material releases, radioactive material releases, credible nuclear criticality accident sequences, and explosions that could result in injuries to workers and/or the public, or significant environmental impacts during routine and non-routine (startup, shutdown, emergency shutdown, etc.) operations.

The accident summary resulting from the ISA identifies which engineered or administrative IROFS must fail to allow the occurrence of consequences that exceed the levels identified in 10 CFR 70.61.

The ISA was used to develop an ISA Summary that has been separated into two documents: (1) an unclassified ISA Summary to be submitted as Security-Related, Export Controlled, and Proprietary Information; and (2) a classified ISA Summary that is submitted separately as Classified, Export Controlled, and Proprietary Information.

3.1 SAFETY PROGRAM AND INTEGRATED SAFETY ANALYSIS COMMITMENTS

3.1.1 Process Safety Information

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

- The hazards of materials used or produced in the process, which includes information on chemical and physical properties included on material safety data sheets (MSDSs) meeting the requirements of 29 CFR 1910.1200(g), *Toxic and Hazardous Substances, (Ref. 3-3).*

- Technology of the process which includes block flow diagrams or simplified process flow diagrams, a brief outline of the process, safe upper and lower limits for controlled parameters (for example, temperature, pressure, flow, and concentration), and evaluation of the health and safety consequences of process deviations.

- 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, IROFS (for example, interlocks, detection, or suppression systems), electrical classification, and relief system design and design basis.

Process safety information is maintained up-to-date by the Configuration Management (CM) Program described in GLE License Application (LA) Section 11.1, *Configuration Management*. Changes to the ISA are conducted in accordance with approved written procedures. This includes implementation of a facility change mechanism that meets the requirements of 10 CFR 70.72, *Facility Changes and Change Process (Ref. 3-4)*. The development and implementation of procedures is described in GLE LA Section 11.4, *Procedures*.

GLE 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 GLE LA Section 2.2, *Key Management Positions, Responsibilities, and Qualifications.*

3.1.2 Integrated Safety Analysis

GLE has conducted an ISA for each process, such that it identifies the following:

- Nuclear criticality hazards,

- Radiological hazards,

- Chemical hazards that could increase radiological risk,

- Facility hazards that could increase radiological risk,
- Credible accident sequences,
- Consequences and likelihood of each accident sequence, and
- IROFS including the assumptions and conditions under which they support compliance with the performance requirements of 10 CFR 70.61.

A summary of the results of the ISA, including the information specified in 10 CFR 70.65(b), *Additional Contents of Application (Ref. 3-5)*, is provided in the ISA Summary.

GLE 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 U.S. Nuclear Regulatory Commission (NRC) in accordance with 10 CFR 70.72(d)(1) and (3). The ISA update process accounts for changes made to the facility or its processes. This update also verifies that initiating event frequencies and IROFS reliability values assumed in the ISA remain valid. Required ISA changes, as a result of the update process, are included in a revision to the ISA. Evaluation of facility changes, or a change in the process safety information, which may alter the parameters of an accident sequence, is performed using the ISA method(s) described in the ISA Summary. For any revisions to the ISA, personnel having qualifications similar to those of ISA Team members who conducted the original ISA are used. Personnel used to update and maintain the ISA and ISA Summary are trained in the ISA method(s) and are suitably qualified.

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, the adequacy of existing IROFS and associated management measures are promptly evaluated and the necessary changes are made, if required. Unacceptable performance deficiencies associated with IROFS are addressed through updates to the ISA.

3.1.3 Management Measures

Management measures are utilized to maintain the IROFS so that they are available and reliable to perform their safety functions when needed. Management measures ensure compliance with the performance requirements assumed in the ISA documentation. The measures are applied to particular structures, systems, components (SSCs), equipment, and activities of personnel; and may be graded commensurate with the reduction of the risk attributable to that IROFS. Management Measures are described in GLE LA Chapter 11, *Management Measures*.

3.1.4 Design Codes and Standards

GLE commits to follow the industry practice to adhere to all "shall" statements in standards applied. Suggestions and recommendations in applied standards (so called "should" statements) are not considered by GLE as binding commitments unless it is specifically stated that GLE's intent is to treat the "should" statements as binding commitments (that is, treat as if they are "shall" statements). GLE may make such commitments as part of the description of the safety program basis. If a definitive commitment to a "should" statement is necessary to provide adequate protection, GLE may provide explanation of this as an issue in response to requests for additional information (RAIs) on specific licensing actions. Suggestions and recommendations in applied standards may or may not be used by GLE, at its discretion if not otherwise identified as binding commitments. Shown in Table 3.1, *Code of Record*, is an inclusive listing of codes and standards that are planned to be used in the safe design of the facility.

3.2 INTEGRATED SAFETY ANALYSIS SUMMARY AND DOCUMENTATION

3.2.1 Site Description

The ISA Summary provides a description of the GLE Site and the surrounding Owner Controlled Area (herein referred to as the Wilmington Site). A summary description of the GLE Site and the Wilmington Site is contained in GLE LA Chapter 1, *General Information*.

3.2.2 Facility Description

The ISA Summary provides a description of the GLE Commercial Facility. A summary description of the GLE Commercial Facility is provided in GLE LA Chapter 1.

3.2.3 Process, Hazards, and Accident Sequences

The ISA Summary provides a description of the GLE Commercial Facility processes and associated SSCs, the process hazards, and a general description of the accident sequences evaluated in the ISA. A summary of the enrichment process is provided in GLE LA Chapter 1.

3.2.4 Compliance with the Performance Requirements of 10 CFR 70.61

The ISA Summary provides information that demonstrates GLE's compliance with the performance requirements of 10 CFR 70.61.

3.2.4.1 Accident Sequence Evaluation and IROFS Designation

The ISA Summary provides information that demonstrates compliance with the performance criteria of 10 CFR 70.61. The ISA Summary provides sufficient information to demonstrate that credible high consequence events are controlled to the extent needed to reduce the likelihood of occurrence to "Highly Unlikely" and credible intermediate consequence events are controlled to the extent needed to reduce the likelihood of occurrence to "Unlikely."

3.2.4.2 Management Measures

The ISA Summary provides a description of the management measures to be applied to IROFS for each accident sequence for which the consequences could exceed the performance requirements of 10 CFR 70.61.

3.2.4.3 Criticality Monitoring

The GLE Commercial Facility has a Criticality Accident Alarm System (CAAS) as required by 10 CFR 70.24, *Criticality Accident Requirements (Ref. 3-6)*. CAAS coverage shall be provided in each process area where special nuclear material (SNM) is handled, used, or stored, with the exception of those areas exempted as described in Section 1.2.5.7 of this License Application. Areas where special nuclear material (SNM) is handled, used, or stored in amounts at or above the 10 CFR 70.24 mass limits have CAAS coverage. The CAAS is designed, installed, and maintained in accordance with ANSI/ANS 8.3-1997, *Criticality Accident Alarm System (Ref. 3-7)*, as modified by Regulatory Guide 3.71, *Nuclear Criticality Safety Standards Fuels and Material Facilities (Ref. 3-8)*. The CAAS is described in GLE LA Chapter 5, *Nuclear Criticality Safety*.

3.2.4.4 New Facilities or New Processes at Existing Facilities

Baseline design criteria (BDC) that must be used for new facilities, is specified in 10 CFR 70.64, *Requirements for New Facilities or New Processes at Existing Facilities (Ref. 3-9)*. The ISA accident sequences for the credible high and intermediate consequence events for the GLE Commercial Facility have defined the design basis events. The IROFS for these events and safety parameter limits ensure that the associated BDC are satisfied. IROFS safety parameter limits are available in the ISA documentation. The BDC in 10 CFR 70.64 have been used as bases for the design of the GLE Commercial Facility as described below.

3.2.4.4.1 Quality Standards and Records

SSCs that are determined by the ISA to be IROFS are designed, fabricated, erected, and tested in accordance with the applicable quality assurance (QA) criteria described in GLE LA Section 11.8, *Other Quality Assurance Elements*. Appropriate records of the design, fabrication, erection, procurement, and testing of SSCs that are IROFS are maintained throughout the life of the facility. Management Measures applicable to IROFS are discussed in GLE LA Chapter 11 and in the ISA Summary.

3.2.4.4.2 Natural Phenomena Hazards

SSCs that are determined to be IROFS are designed to withstand the effects of, and be compatible with, the environmental conditions associated with operation, maintenance, shutdown, testing, and accidents for which the IROFS are required to function.

3.2.4.4.3 Fire Protection

SSCs that are IROFS are designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Non-combustible and heat resistant materials are used wherever practical throughout the facility, particularly in locations vital to the control of hazardous materials and to the maintenance of safety control functions. Fire detection, alarm, and suppression systems are designed and provided with sufficient capacity and capability to minimize the adverse effects of fires and explosion on IROFS. The design includes provisions to protect against adverse effects that may result from either the operation or the failure of the fire suppression system.

3.2.4.4.4 Environmental and Dynamic Effects

SSCs that are IROFS are protected against dynamic effects, including effects of missiles and discharging fluids, which may result from natural phenomena; accidents at nearby industrial, military, or transportation facilities; equipment failure; and other similar events and conditions both inside and outside the facility.

3.2.4.4.5 Chemical Protection

The design provides adequate protection against chemical risks produced from licensed material, facility conditions that affect the safety of licensed material, and hazardous chemicals produced from licensed material.

3.2.4.4.6 Emergency Capability

SSCs that are required to support the GLE Radiological Contingency and Emergency Plan (RC&EP) are designed for emergencies. The design provides accessibility to the equipment of onsite and available offsite emergency facilities and services such as hospitals, fire and police departments, ambulance service, and other emergency agencies.

3.2.4.4.7 Utility Services

Onsite utility service systems required to support IROFS are provided. Each utility service system required to support IROFS are designed to perform their function under normal and abnormal conditions. Utility systems are described in the ISA Summary.

3.2.4.4.8 Inspection, Testing, and Maintenance

SSCs that are determined to be IROFS are designed to permit inspection, maintenance, and testing.

3.2.4.4.9 Criticality Control

The design of process and storage systems shall include demonstrable margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the process and storage conditions, in the data and methods used in calculations, and in the nature of the immediate environment under accident conditions. Process and storage systems are designed and maintained with sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible. The Nuclear Criticality Safety (NCS) Program and NCS methodologies and technical practices are described in GLE LA Chapter 5.

3.2.4.4.10 Instrumentation and Controls

Instrumentation and control systems are provided to monitor variables and operating systems that are significant to safety over anticipated ranges for normal operation, abnormal operation, accident conditions, and safe shutdown. These systems ensure adequate safety of process and utility service operations in connection with their safety function.

The variables and systems that require surveillance and control include process systems having safety significance, the overall confinement system, confinement barriers and their associated systems, and other systems that affect the overall safety of the facility. Controls shall be provided to maintain these variables and systems within the prescribed operating ranges under normal conditions. Instrumentation and control systems are designed to fail into a safe state or to assume a state demonstrated to be acceptable on some other basis if conditions such as disconnection, loss of energy or motive power, or adverse environments are experienced.

3.2.4.4.11 Defense-in-Depth Practices

The facility and system designs are based on defense-in-depth practices. The design incorporates a preference for engineered controls over administrative controls to increase overall system reliability. For criticality safety, the engineered controls preference is for use of passive engineered controls over active engineered controls. The design also incorporates features that enhance safety by reducing challenges to IROFS. Facility and system IROFS are identified in the ISA Summary.

The enrichment process systems and support systems are described in the ISA Summary. In addition to identifying the IROFS associated with each system, the ISA Summary identifies the additional design and safety features (considerations) that provide defense-in-depth.

3.2.5 Integrated Safety Analysis Methodology

GLE utilized methodologies identified in NUREG-1520, Chapter 3, Appendix A, to identify hazards and evaluate accident scenarios. This approach employs a semi-quantitative risk index method for categorizing accident sequences in terms of their consequences of concern and their likelihood of occurrence. The risk index method framework identifies which accident sequences have consequences that could exceed the performance requirements of 10 CFR 70.61 and; therefore, require designation of 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 facility and external hazards, potential accidents, accident descriptions, the likelihood and consequences of the accidents, and the IROFS.

The ISA uses a hazard analysis method, the What-If/Checklist Method, to identify the hazards relevant to each node or the facility in general. The ISA Team reviewed the hazards identified for the "credible worst-case" consequences. The credible high or intermediate severity consequence accident scenarios were assigned accident description identifiers, accident descriptions, frequency or probability, and then a risk index determination was performed. The risk index was used to evaluate unmitigated risk as unacceptable or acceptable.

For each accident scenario having an unacceptable unmitigated risk index, IROFS were defined and the mitigated likelihood determined for each accident scenario. Using the unmitigated initiating event frequency and the failure probability of each IROFS, the mitigated likelihood and mitigated risk was determined. 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. The credible accidents that potentially exceed the levels identified in 10 CFR 70.61 are evaluated using a Quantitative Risk Analysis (QRA) approach. The determination of the mitigated likelihood for an accident scenario is documented in a QRA report.

The intent of the QRA reports is to evaluate unacceptable risk identified during a formal What-If analysis. The ISA provides sufficient background and operational information to understand and examine accident scenarios that result in undesired outcomes for each initiating event. Each QRA report provides details concerning an accident scenario's quantification, including the method used; initiating event frequency determination; enabling or conditional event probabilities; the IROFS credited to prevent or mitigate the initiating event(s) being analyzed; the failure probabilities for the credited IROFS; and the overall likelihood estimates. Initiating event frequencies of occurrence presented in the QRAs were conservatively selected with the maximum event frequency bounded by a frequency of once per year. The QRA reports are controlled documents and maintained up-to-date by the CM Program described in GLE LA Chapter 11.

Figure 3-1, *Integrated Safety Analysis Process Flow Diagram,* describes the ISA process steps. The following sub-sections correspond to each block in the flow diagram.

3.2.5.1 Define Nodes to be Evaluated

The first step of the ISA is for the ISA Team to systematically break down the process system, subsystem, facility area, or operation being studied into well-defined nodes. The ISA nodes establish the study area boundaries in which the various process systems and supporting systems entering or exiting the node, or activities occurring in the area, can be defined in order to allow interactions to be studied.

Operations were treated in this manner so that the entire facility was evaluated in a logical process flow approach. This approach is also used to evaluate the hazards associated with each process or operation, and to identify any new hazards resulting from modifications made to an existing process or operation. The GLE Commercial Facility defined nodes are listed in Table 3-2, *Integrated Safety Analysis Nodes*. Information used to define the nodes and to perform the process hazard analysis (PHA) includes, but are not limited to, the following:

- System descriptions,
- Process flow diagrams,
- Plot plans,
- Topographic maps,
- Equipment arrangement drawings with general equipment layout and elevations,
- Design temperatures and pressures for major process equipment and interconnected piping,
- Materials of construction for major process equipment and interconnected piping,
- MSDSs for any chemicals involved in the process (including any intermediate chemical reaction products) and other pertinent data for the chemicals or process chemistry (such as, chemical reactivity hazards),
- Utility system drawings, and
- Criticality safety analyses (CSAs) / radiological safety assessments (RSAs).

3.2.5.2 *Hazard Identification*

What-If analysis and Checklist Methods were used for identifying the hazards for the GLE process. Event Tree Analysis was employed to assist in determining credible or non-credible events and in identifying IROFS. These methods are consistent with the guidance provided in NUREG-1520 and NUREG-1513, *Integrated Safety Analysis Document (Ref. 3-10)*. The hazard identification process documents materials that are:

- Radioactive,
- Fissile,
- Flammable,
- Explosive,
- Toxic, and
- Reactive.

The hazards identification process results in identification of radiological or chemical characteristics that have the potential for causing harm to workers, the public, or to the environment. The hazards of concern for the GLE Commercial Facility are related to either a release of uranium hexafluoride (UF_6) (loss of confinement) or a criticality. In general, the loss of confinement would initially result in moisture in the air reacting with the UF_6, forming uranyl fluoride (UO_2F_2) and hydrogen fluoride (HF) as by-products. The HF, which would be in a gaseous form, could be transported through the facility and ultimately beyond the site boundary. HF is a toxic chemical with the potential to cause harm to the workers or the public. For licensed material or hazardous chemicals produced from licensed materials, chemicals of concern are those that, in the event of release, have the potential to exceed concentrations defined in 10 CFR 70, *Domestic Licensing of Special Nuclear Material (Ref. 3-11)*. Criteria for evaluating potential releases and characterizing their consequence as either "High" or "Intermediate" for members of the public and facility workers are presented in Table 3-3, *Consequence Severity Categories Based on 10 CFR 70.61*, and Table 3-4, *AEGL Thresholds from the EPA for Uranium Hexafluoride, Soluble Uranium, and Hydrogen Fluoride*.

An HF release would cause a visible cloud and a pungent odor. The odor threshold for HF is less than 1 part per million (ppm) and the irritating effects of HF are intolerable at concentrations well below those that could cause permanent injury or which produce escape-impairing symptoms. Employees are trained in proper actions to take in response to a release and it can be confidently predicted that workers will take immediate self-protective action to escape a release area upon detecting any significant HF odor. Sufficient time is available for the worker to reliably detect and evacuate the area of concern. Public exposures were estimated to last for duration of 30 minutes. This is consistent with self-protective criteria for UF_6/HF plumes listed in NUREG-1140, *A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees (Ref. 3-12)*. The AEGL-1, -2, and -3 values were used as the threshold concentration levels for establishing a low, intermediate, or high severity consequence as shown in Table 3-3. AEGL values for other time periods may be utilized if more appropriate for the accident scenarios in question.

10 CFR 70.61(b)(3) states, *An intake of 30 mg or greater of uranium in soluble form by any individual located outside the controlled area identified pursuant to Paragraph (f) of this section*. The UF_6 concentration in air is not directly equivalent to soluble uranium intake. GLE uses an accepted intake value of 75 mg or greater, corresponding to the threshold for permanent renal damage consistent with a high consequence event to a worker as defined in 10 CFR 70.61(b)(4).

Dermal exposures to HF have been evaluated in the ISA Summary. Although HF is not used directly in the enrichment process, limited quantities of dilute HF (< 4%) are generated in the Laboratory and Decontamination and Maintenance Areas. The criteria for assessing the consequence severity for HF dermal exposures are provided in Table 3-3.

The What-If/Checklist Analysis method was used for identifying process hazards for the UF_6 process systems at GLE Commercial Facility. This PHA technique combines the What-If Analysis with Checklist Analysis, which is used to identify and document items identified in the hazard analysis meetings. The hybrid method lends a more systematic nature to the "Brainstorming" character of the What-If method. For identified single-failure events (that is, those accidents that result from the failure of a single control), the What-If method is the recommended approach. Previously performed What-If analyses developed for similar or identical processes at the Wilmington Site were used as a checklist to ensure completeness of the GLE Commercial Facility What-If analyses. The primary sources were What-If analyses developed for onsite facilities. Implementation of the What-If/Checklist method was accomplished using the GLE Commercial Facility design and performing a What-If for each system.

The results of the ISA Team meetings are summarized in the ISA What-If/Checklist tables, which forms the basis of the hazards portion of the Hazard and Risk Determination Analysis. The What-If/Checklist 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 3-5, *What-If/Checklist Example*. The What-If Checklist is divided into ten (10) columns, as follows:

1. <u>Item</u> – This is a unique number assigned to each What-If.

2. <u>What-If</u> – This column provides a description of the What-If question to be analyzed.

3. <u>Scenarios Initiator</u> – This column provides a description of the initiating event required to cause the accident.

4. <u>Consequence</u> – This column provides a description of the design basis event (for example, the potential and worst case consequences from fire, potential criticality event, etc.)

5. <u>Category</u> – This column provides the risk category affecting workers, the public, and the environment.

6. <u>Severity</u> – This column identifies the estimated severity category as unmitigated hazard.

7. <u>Likelihood</u> – This column identifies the frequency category of the event as unmitigated hazard.

8. Risk – This column identifies whether the unmitigated risk is acceptable or unacceptable based on the estimated severity, likelihood, and the results of the risk index.

9. IROFS – This column identifies the IROFS which identifies the engineered and/or administrative protection designed to prevent the hazard from occurring.

10. References – This column provides reference to documents used by the ISA Team that provided support to the determinations made during the hazard review.

This approach was used for the process system hazard identification. The results of the unmitigated What-If/Checklists are used directly as input to the risk matrix and risk index development. In addition, the hazard identification identifies potentially hazardous process conditions. Most hazards were assessed individually for the potential impact on the discrete components of the process systems. However, hazards were assessed on a facility-wide basis for credible hazards from fires (such as, external to the process system) and external events (such as, seismic, severe weather, etc.).

As stated earlier, the hazards of concern are related to either a release of UF_6 or a postulated criticality event as a potential source of damaging energy and would result in the release of prompt radiation and airborne fission products. The radiation and airborne fission products could result in direct radiation exposure and chemical/radiological inhalation exposure to workers and the public. Each SSC that may possibly contain enriched uranium is designed with criticality safety as an objective.

For the design of new facilities, like the GLE Commercial Facility, or significant additions or changes in existing facilities, the proposed design is reviewed by the NCS function to identify potential criticality hazards. The NCS function evaluates each fissile material process to identify the normal and credible abnormal conditions, and establishes the controls required to meet the double contingency design criteria. Use of the double contingency design criteria assures that nuclear processes remain subcritical under normal and credible abnormal conditions. The NCS evaluations that provide the criticality safety basis are documented in CSAs, which describe the facility criticality hazards and the identification of criticality accident scenarios. The CSAs are an integrated part of the ISA, which document the criticality hazards and credible criticality accident scenarios. The ISA input information is included in the ISA documentation.

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* combustible material from an unidentified ignition source (such as, electrical shorting, or other source);
- Postulated the development of a fire occurring in transient combustible material from an unidentified ignition source (such as, electrical shorting, or other source); and
- Evaluated the uranic content in the space and its configuration (for example, UF_6 solid/gas in cylinders, UF_6 gas in piping, UF_6 and/or byproducts bound on chemical traps, 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 important systems, a Fire Hazards Analysis (FHA) 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 estimated. Therefore, in order to place reasonable and conservative bounds on the fire scenarios analyzed, the ISA Team estimated *in-situ* combustible loadings based on the FHA information of the *in-situ* combustible loading for the GLE Commercial Facility. This information indicates that *in-situ* combustible loads are expected to be very low.

External events were considered at the site and facility level. The external event ISA considered both natural phenomena and man-made hazards. During the external event ISA Team meeting, each area of the GLE Commercial Facility 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 identified to be a low consequence, then a specific design basis with a likelihood of "Highly Unlikely" would be selected. Each external event was assessed for both the unmitigated case and then for the mitigated case. The mitigated cases could be a specific design basis for that external event, IROFS, or a combination of both.

Natural phenomena hazards (NPH) considered for evaluation included:

- Earthquakes,
- Hurricanes (including topical storms),
- Tornados (including tornado missiles and extreme straight wind),
- Volcanoes,
- Flooding,
- Tsunamis,
- Snow and ice, and
- Local precipitation.

External man-made hazards considered for evaluation included:

- Transportation hazards onsite/offsite,
- Onsite facility hazards,
- Aircraft crashes,
- Wildland fires (range fires),
- Pipelines,
- Roadways and highways,
- Nearby industrial facilities,
- Nearby military installations,
- Railways,

- Waterways,
- Underground utilities (onsite use of natural gas and electrical services),
- Internal flooding from onsite above ground liquid storage tanks, and
- Land use impacts.

3.2.5.3 Identify Accident Scenarios

The goal is to identify credible accident scenarios or sequences by analyzing single initiating events. Using approved methods, the ISA Team identified potential accident scenarios associated with a process or operation, including possible worse-case consequences, causes (events that can initiate the accident), and safeguards or controls that are available to prevent the cause of the event or mitigate the consequences. Safeguards are design features or administrative programs that provide defense-in-depth, but are not credited as IROFS. Consequences of interest include nuclear criticality accidents, radiological material releases, radiation exposures, chemical/toxic exposures from licensed material or hazardous chemicals produced from licensed material, and fires and explosions. Hazards are defined to be materials, equipment, or energy sources with the potential to cause injury or illness to humans.

An important product of an ISA consists of a description of accident scenarios identified and recorded during the analysis process. An accident scenario involves an initiating event, any factors that allow the accident to propagate (enablers), and any factors that reduce the risk (likelihood or consequence) of the accident (controls). The accident scenario is a scenario of specific real events.

When analyzing accident scenarios, the ISA Team considered process deviations, human errors, internal facility events, and credible external events, including natural phenomena. Natural phenomenon events, such as hurricanes, tornadoes/high winds, seismic events, and external events (such as aircraft crashes) are addressed separately in Chapter 2 of the ISA Summary. FCSS ISG-08, *Natural Phenomena Hazards (Ref. 3-13)*, was used as guidance when evaluating natural phenomena hazards as initiating events. The team evaluated common mode failures and systems interactions where preventive actions and/or control measures are required to prevent and/or mitigate accident scenarios. The team-listed scenarios considered not credible. In addition to normal conditions, the team considered abnormal conditions including startup, shutdown, maintenance, and process upsets.

For each accident scenario, enabling conditions, and conditional events that affect the outcome of the accident scenario (for example, conditions that affect the likelihood of the scenario or could mitigate the consequences to either workers or the public) were identified where appropriate.

An enabling condition does not directly cause the scenario but must be present for the initiating event to proceed to the consequences described. Enabling conditions are expressed as probabilities and can reflect such things as the mode of operation (for example, percent of operational online availability).

Conditional events that affect the probability of the undesired outcome were also identified. These include probabilistic consideration of individual or administrative actions that would not be considered IROFS but would affect the overall likelihood of the accident. For example, if a scenario involves personal injury hazards, at least one worker must be present in the affected area at the time of the event for the injury to occur. Thus, the presence of workers in the affected area is a conditional modifier for a consequence involving personal injury. Another example of a conditional event is the probability that a worker can successfully evacuate from an area given that a hazard is present.

In considering accident scenarios at the GLE Commercial Facility, it is necessary to determine which scenarios are considered not credible and which are credible. When conducting the PHA, the ISA Team considered each accident scenario as credible, unless the scenario could be determined to be not credible. See Section REF _Ref231182005 \r \h 3.2.5.5, *Determine Unmitigated Likelihood*, for the criteria GLE used to determine if an accident scenario is credible.

3.2.5.4 Determine Consequence Severity

Table 3-3 presents the radiological and chemical consequences severity limits of 10 CFR 70.61 for each of the three accident consequences categories. Table 3-4 provides information on the chemical dose limits specific to the GLE Commercial Facility.

For each credible accident scenario identified, the ISA Team assigned a severity ranking for the consequences using the consequence severity rankings provided in Table 3-3. Assigning a severity ranking allowed each accident scenario to be categorized in terms of the performance requirements outlined in 10 CFR 70.61(b), (c), and (d). The Severity Ranking System is outlined below:

- A severity ranking of 3 corresponds to high consequences,
- A severity ranking of 2 corresponds to intermediate consequences, and
- A severity ranking of 1 corresponds to low consequences.

When estimating the possible "worst-case" consequences of an accident scenario, the ISA Team members used experience, guidance from NUREG/CR-6410, *Nuclear Fuel Cycle Facility Accident Analysis Handbook (Ref. 3-13)*, and best judgment.

10 CFR 70.61 specifies two categories for a credible accident description consequence: "Credible High Consequence" and "Intermediate Consequence." Implicitly there is a third category for accidents that produce consequence less than "Intermediate." These are referred to as "Low Consequence" accident descriptions. The primary purpose of PHA is to identify the uncontrolled and unmitigated accident descriptions. These accident descriptions 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 consequence, calculations were performed using the methodology described in the ISA documentation. The consequence of concern is the chemo-toxic exposure to HF and UO_2F_2. The dose consequence for each of the accident descriptions were evaluated and compared to the criteria for "High" and "Intermediate" consequences.

The inventory or uranic material for each accident considered was dependent on the specific accident description. For potential criticality accidents, the consequence was conservatively assumed to the high for the worker, the public, and the environment. Scenarios that resulted in a severity rank of 2 or 3 are: criticality, large UF_6/HF release (such as a multiple cylinder failure or cascade failure), and a heated cylinder release. A solid or gas release of a cold trap, low-temperature takeoff station (LTTS), or single cylinder that is not heated does not exceed intermediate consequence requirements. For a severity level of 1, there is "No Safety Consequence of Concern." There is no further action and the What-If checklist is updated.

3.2.5.5 Determine Unmitigated Likelihood

The likelihood of an accident scenario occurring was determined for the unmitigated case (unmitigated likelihood). Unmitigated likelihood is the likelihood or frequency that the initiating event or cause of the accident sequence occurs. This likelihood/frequency estimate assumes that none of the available safeguards or IROFS are available to perform their intended safety function. Table 3-6, *Unmitigated Likelihood Categories,* shows the likelihood of occurrence limits of 10 CFR 70.61 for each of the three likelihood categories. The team assigned a likelihood level for each accident scenario using the defined categories in Table 3-7, *Event Likelihood Categories*, and Table 3-8, *Determination of Likelihood Category.* When assigning a likelihood category, the team made use of process knowledge, accident scenario information, operating history, and manufacturers/product information to determine which category of likelihood was appropriate. For accident scenarios where multiple initiating events have been identified, the team estimated the likelihood for the most credible initiating event. This helped ensure that the accident scenario was screened using the most conservative estimate of risk.

The definitions of likelihood terms are presented in the following sections.

3.2.5.5.1 Highly Unlikely

The guideline for acceptance of the definition of "Highly Unlikely" has been derived as the highest acceptable frequency that is consistent with a goal of having no inadvertent nuclear criticality accidents and no accidents of similar consequences in the industry. To within an order of magnitude, this is taken to mean a frequency limit of less than one such accident in the industry every 100 years. This has been translated into a guideline limiting the frequency of individual accidents to 10^{-5} per-event per-year. As the goal is to have no such accidents, accident frequencies should be reduced substantially below this guideline when feasible.

3.2.5.5.2 Unlikely

Intermediate consequence events include significant radiation exposures to workers (those exceeding 0.25 Sieverts or 25 rem). No increase in the rate of such significant exposures is the NRC's goal. This has been translated into a guideline of 4.0×10^{-5} per-event per-year. This guideline may be more generally considered as a range between 10^{-4} and 10^{-5} per-event per-year since exact frequencies at such levels cannot accurately be determined.

3.2.5.5.3 *Not Credible*

The definition of "Not Credible" is also taken from NUREG-1520. 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 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:"

- An external event for which the frequency of occurrence can conservatively be estimated as less than once in a million years.
- A process deviation that consists of a description of many unlikely human actions or errors for which there is no reason or motive. In determining that there is no reason for such actions, a wide range of possible motives, short of intent to cause harm, must be considered. Necessarily, no such description of events can ever have actually happened in any fuel cycle facility.
- Process deviations for which there is a convincing argument, given physical laws, that they are not possible, or are unquestionably extremely unlikely.

3.2.5.5.4 *Credible*

A "Credible" accident is any event that does not meet the definition of "Not Credible" as defined above.

3.2.5.6 *Determine Unmitigated Risk*

Credible accident scenarios identified for the facility, which have the capability of producing conditions that fail to meet the performance requirements of 10 CFR 70.61(b), (c) or (d), are included in the scope of the ISA Summary. For each credible accident scenario, the ISA Team used the severity category ranking and unmitigated likelihood level to assign an unmitigated risk level. (The unmitigated risk is determined from the product of the severity category and the unmitigated-likelihood category.) The ISA Team used the risk matrix in Table 3-9, *Unmitigated Risk Assignment Matrix*, to determine the unmitigated risk. The unmitigated risk associated with each accident scenario indicates the relative importance of the associated controls. Accident scenarios of which the consequences and likelihoods yield an unacceptable risk index require further evaluation to determine IROFS and mitigated risk, as described in Section 3.2.5.8, *Develop IROFS and Frequency Determination*.

If the unmitigated risk is less than or equal to 4, the unmitigated risk is acceptable and no further action is required. The What-If table is updated to reflect this conclusion of no further action and the Qualitative Risk Analysis is performed.

3.2.5.7 *Perform Quantitative Risk Analysis*

The QRA identifies the GLE Commercial Facility nodes to which it applies, describes the node operations and operational areas, presents the QRA layout including the PHA reference nodes, accident description, initiating events evaluated, potential preventive and mitigative features, and describes management measures. An event tree analysis is provided and the overall likelihood of the accident is given.

3.2.5.8 *Develop IROFS and Frequency Determination*

For each accident scenario having an unacceptable unmitigated risk index, IROFS must be defined and the mitigated likelihood determined for each accident scenario. Using the unmitigated initiating event frequency and the failure probability of each IROFS, the mitigated likelihood is determined.

The QRAs present an accident evaluation including a detailed discussion concerning the selection of initiating events, IROFS, and the quantification of the accident sequences through the use of event trees. Determination of the mitigated likelihood for an accident scenario is documented in a QRA Report. The intent of the QRA reports is to provide sufficient background and operational information to understand and examine accident scenarios that result in undesired outcomes for each initiating event. Each QRA report provides details concerning an accident scenario's quantification, including method used, initiating-event frequency determination, the IROFS credited to prevent or mitigate the initiating event(s) being analyzed, the failure probabilities for the credited IROFS, and the overall likelihood estimates. The QRA reports are controlled documents and are maintained up-to-date by the CM Program described in GLE LA Section 11.1. The quantification results from each QRA are summarized in this ISA Summary.

The mitigated likelihood of the accident scenario occurring with the preventive or mitigating IROFS in-place must meet the requirements in 10 CFR 70.61, which requires that unacceptable consequences be limited. This is accomplished using index values, which are defined as the logarithm of the frequency (or probability) associated with the initiating event and subsequent IROFS failures for the accident scenario. The values of the index numbers for an accident scenario, depending on the number of events involved, are added to obtain a total likelihood index, "T." The likelihood index is therefore the logarithm of the overall likelihood (that is, $\log_{10}(L_T)$). Accident scenarios are then assigned to one of the three likelihood categories of the risk matrix, depending on the value of the likelihood index in accordance with Table 3-7.

The reliability and availability of an IROFS to perform is a function of the management measures applied to each IROFS. The management measures provide the overall management oversight and assurance that the GLE safety program is maintained and functions properly. These management measures are described in GLE LA Chapter 11. ISA Summary, Appendix C, provides a consolidated list of IROFS.

For IROFS, 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 (Ref. 3-16)*; and NUREG-0711, *Human Factors Engineering Program Review Model (Ref. 3-17)*. This guidance will be implemented in a Human Factors Engineering Plan and integrated into the Design Process, Training Program, Procedures Program, and Quality Assurance Program implementing policies, plans, and procedures, as applicable.

In this document, safety controls and IROFS are synonymous. Safeguards are design features or administrative programs that provide defense-in-depth, but are not IROFS and are not credited with preventing or mitigating accident scenarios. 10 CFR 70.64 states that the design process must be founded on defense-in-depth principles, and incorporate, to the extent practicable, preference for engineered controls over administrative controls, and reduction of challenges to the IROFS that are frequently or continuously challenged. Safety controls used at the facility can be characterized as either administrative or engineered. Administrative controls are generally not considered to be as reliable as engineered controls since human errors usually occur more frequently than equipment failures. Engineered controls may be categorized as being "Passive" or "Active." Passive controls include pipes or vessels that provide containment. Active controls include equipment such as pumps or valves that perform a specific function related to safety. In general, passive controls are considered to be less prone to failure than active controls.

IROFS are those engineered or administrative controls, or control systems, which comprise the SSCs that form the preventive and/or mitigating barriers identified by the ISA. The IROFS selected for each accident scenario may be a control that helps reduce the likelihood that the initiating event occurs, detects or mitigates the consequences, or helps reduce the amount of hazardous material released. IROFS are the barriers that prevent and/or mitigate the unacceptable consequences identified by the performance requirements of 10 CFR 70.61(b), (c) and (d). When selecting IROFS, the IROFS must be independent of the initiating event (for example, occurrence of the initiating event does not cause failure of the IROFS) and other credited IROFS (for example, failure of one IROFS does not cause failure of another IROFS).

GLE commits to identify IROFS as a part of the ISA process and include the identification of the IROFS in the ISA Summary prepared and maintained for the GLE Commercial Facility. The IROFS are defined in such a way as to delineate their boundaries, to describe the characteristics of the preventive/mitigating function, and to identify the assumptions and conditions under which the item is relied on.

3.2.5.9 Update What-If/Checklist, Risk Index, and ISA Summary

The QRA document results in the development of IROFS and the overall accident sequence frequency determination based on the event tree evaluation of the potential accident. This information was then used to update the What-If/Checklist table, including the unmitigated likelihood and the unmitigated risk.

Based on the updated What-If/Checklist and the QRA, the Accident Sequence Summary and Risk Index (Table 3-10) is completed. For accident sequences that are of low consequence, or that have a risk index of 4 or less, the risk is acceptable and Table 3-10 requires no entries (that is, "N/A") for the initiating event frequency, IROFS and their failure probabilities, or likelihood index.

The ISA process is an iterative process. The ISA Summary provides an overview of the ISA based upon the existing design level of detail. The ISA Summary that supports the License Application is based on the level of design necessary to establish the safety basis for the GLE Commercial Facility and support the licensing effort.

The final step of the ISA process (see Figure 3-1) is to update supporting ISA documentation and then develop the ISA Summary. As the design of the GLE Commercial Facility progresses, the ISA and supporting documents will be revised, or new supporting documents developed.

3.2.5.10 ISA Integration

The ISA is intended to give assurance that the potential failures, hazards, accident descriptions, 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 simultaneously be 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 mitigative function of another IROFS. The ISA Team performed an integrated review during the process hazard review and an overall integration review after the nodes were completed. Some items that warrant special consideration during the integration process evaluation 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; that is, 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 descriptions with multiple impacts, that is, impacts on chemical, fire, criticality, and/or radiation safety. For example, a flood might cause both a loss of confinement 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 that tend to be common cause situations leading to interactions between processes, systems, buildings, etc.

3.2.6 Integrated Safety Analysis Team

The ISA was performed, and is maintained, 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 the following:

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

The ISA Team leader is trained and knowledgeable in the ISA method(s) chosen for the hazard and accidents evaluations. A qualified NCS engineer is included on each ISA Team. Collectively, the team had an understanding of the process operations and hazards under evaluation. The ISA Manager is responsible for the overall direction of the ISA. Additional information on the ISA Team is provided in ISA Summary Chapter 1, *General ISA Information*.

3.2.7 Descriptive List of IROFS

The ISA Summary provides a list of IROFS in the identified high and intermediate accident sequences.

3.2.8 Sole Items Relied On For Safety

Sole IROFS are not used for the GLE Commercial Facility. Instead, a minimum of two (2) independent IROFS are typically selected.

3.3 REFERENCES

3-1. NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility*, U.S. Nuclear Regulatory Commission, March 2002.

3-2. 10 CFR 70.61, *Performance Requirements*, U.S. Nuclear Regulatory Commission, 2008.

3-3. 29 CFR 1910.1200, *Toxic and Hazardous Substances*, Occupational Safety and Health Administration, 2008.

3-4. 10 CFR 70.72, *Facility Changes and Change Process*, U.S. Nuclear Regulatory Commission, 2008.

3-5. 10 CFR 70.65, *Additional Content of Application*, U.S. Nuclear Regulatory Commission, 2008.

3-6. 10 CFR 70.24, *Criticality Accident Requirements*, U.S. Nuclear Regulatory Commission, 2008.

3-7. ANSI/ANS 8.3-1997 (R2003), *Criticality Accident Alarm System*, American Nuclear Society, January 1997.

3-8. Regulatory Guide 3.71, *Nuclear Criticality Safety Standards for Fuels and Material Facilities*, U.S. Nuclear Regulatory Commission, Revision 1, October 2005.

3-9. 10 CFR 70.64, *Requirements for New Facilities or New Processes at Existing Facilities*, U.S. Nuclear Regulatory Commission, 2008.

3-10. NUREG-1513, *Integrated Safety Analysis Guidance Document*, U.S. Nuclear Regulatory Commission, May 2001.

3-11. 10 CFR 70, *Domestic Licensing of Special Nuclear Material,* U.S. Nuclear Regulatory Commission, 2008.

3-12. NUREG-1140, *A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees*, U.S. Nuclear Regulatory Commission, January 1988.

3-13. FCSS ISG-08, *Natural Phenomena Hazards,* U.S. Nuclear Regulatory Commission, Revision 0, October 2005.

3-14. NUREG/CR-6410, *Nuclear Fuel Cycle Facility Accident Analysis Handbook*, U.S. Nuclear Regulatory Commission, March 1998.

3-15. 10 CFR 20, *Standards for Protection Against Radiation,* U.S. Nuclear Regulatory Commission, 2008.

3-16. NUREG-0700, *Human-System Interface Design Review Guidelines,* U.S. Nuclear Regulatory Commission, Revision 2, May 2002.

3-17. NUREG-0711, *Human Factors Engineering Program Review Model,* U.S. Nuclear Regulatory Commission, Revision 2, February 2004.

Table 3-1. GLE Commercial Facility Design Codes and Standards.[1]

Code Group / Reference	Code Number	Year or Edition	Title
ACGIH	2090	2001	Industrial Ventilation: A Manual of Recommended Practice
ACI	117	2006	Specifications for Tolerances for Concrete Construction
ACI	318	2008	Building Code Requirements for Structural Concrete
ACI	349	2007	Code Requirements for Nuclear Safety Related Concrete Structures
AISC	325-05 13th Edition	2006	Manual of Steel Construction
AISC	341	2005	Seismic Provision for Structural Steel Buildings
AISC	360	2005	Specification for Structural Steel Building
AISC	N-690 (S327)	2006	Nuclear Facilities, Steel Safety-Related Structures for Design and Fabrication
ANSI	N13.2	1982	Administrative Practices in Radiation Monitoring (A Guide for Management)
ANSI	N14.1	2001	Nuclear Materials - Uranium Hexafluoride – Packaging for Transport
ANSI/AIHA	Z9.5	2003	Laboratory Ventilation
ANSI/ANS	2.26	2004	Categorization of Nuclear Facility Structures, Systems, and Components for Seismic Design
ANSI/ANS	3.1	1993	Selection, Qualification, and Training of Personnel for Nuclear Power Plants
ANSI/ANS	8.1	2007	Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactor
ANSI/ANS	8.3	1997	Criticality Accident Alarm System as modified by Regulatory Guide 3.71, Nuclear Criticality Safety Standards Fuels and Material Facilities
ANSI/ANS	8.19	2005	Administrative Practices for Nuclear Criticality Safety
ANSI/ANS	8.20	1991	Nuclear Criticality Safety Training

[1] In citing industry consensus codes and standards the applicant has not delineated specific commitments in the standards that will be adopted. These industry consensus codes and standards may not be adopted in their entirety, but form the initial baseline of applicable codes and standards that are evaluated during the design of the GLE CF. Actual codes and standards are established in design documents and the design criteria manual. These documents provide the level of compliance or non-compliance necessary to understand the design criteria used for the design and construction of the GLE Facilities.

Code Group / Reference	Code Number	Year or Edition	Title
ANSI/ANS	8.21	1995	Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors
ANSI/ANS	8.22	1997	Nuclear Criticality Safety Based on Limiting and Controlling Moderators
ANSI/ANS	8.23	1997	Nuclear Criticality Accident Emergency Planning and Response
ANSI/ANS	8.24	2007	Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations
ANSI/ANS	8.26	2007	Criticality Safety Engineer Training and Qualification Program
ANSI/ASME	AG-1	2009	Code on Nuclear Air and Gas Treatment, Section FC-5160.
ANSI/ASME	B16.5	1996	Pipe Flanges and Flanged Fittings
ANSI/ASME	B30.2	2005	Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trough Hoist)
ANSI/ASME	B31.3	2008	Process Piping
ANSI/ASME	B31.9	2008	Building Services Piping
ANSI/ASME	NOG-1	2004	Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)
ANSI/ASSE	Z117.1	2009	Safety Requirements for Confined Spaces
ANSI/IEEE	C2	2007	National Electric Safety Code
ANSI/IEEE	C37.04	2006	Rating Structure for AC High-Voltage Circuit Breakers Rated on a Symmetrical Current Basis
ANSI/IEEE	C37.06	2000	Switchgear – AC High-voltage Circuit Breakers Rated on a Symmetrical Current Basis - Preferred Ratings and Related Required Capabilities
ANSI/IEEE	C37.11	2003	AC High-Voltage Circuit Breaker Control Requirements
ANSI/IEEE	C37.20.2	2005	Metal-Clad Switchgear
ANSI/IEEE	C37.90	2005	Standard for Relays and Relay Systems Associated with Electric Power Apparatus
ANSI/IEEE	C37.90.1	2002	IEEE Standard for Surge Withstand Capability (SWC) Tests for Relays and Relay Systems Associated with Electric Power Apparatus
ANSI/IEEE	C37.100	2001	Definitions for Power Switchgear
ANSI/IEEE	C57.12.80	2002	Standard Terminology for Power and Distribution Transformers
ANSI/IEEE	C57.12.90	2006	Standard Test Code for Liquid-Immersed Distribution, Power, and Regulating Transformers

Code Group / Reference	Code Number	Year or Edition	Title
ANSI/IEEE	C57.12.91	2001	Standard Test Code for Dry-Type Distribution and Power Transformers
ANSI/ISA	67.04.01	2006	Setpoints for Nuclear Safety-Related Instrumentation
ASCE	7-05	2006	Minimum Design Loads for Buildings and Other Structures
ASHRAE	62.1	2007	Ventilation for Acceptable Indoor Air Quality
ASHRAE	90.1	2007	Energy Standard for Buildings Except Low-Rise Residential Buildings
ASME	AG-1	2009	Code on Nuclear Air and Gas Treatment
ASME	N510	2007	Testing of Nuclear Air Treatment Systems
ASME	NQA-1	1994	Quality Assurance Requirements for Nuclear Facility Applications, w/Addenda Part I: Basic Requirements and Supplementary Requirements for Nuclear Facilities, Part II: Quality Assurance Requirements for Nuclear Facility Application, Part III: Non-Mandatory Appendices
ASME	Section VIII	2007	Boiler and Pressure Vessel Code
ASTM	C761-04	2004	Standard Test Methods for Chemical, Mass Spectrometric, Spectrochemical, Nuclear, and Radiochemical Analysis of Uranium Hexafluoride
ASTM	C787-06	2006	Standard Specification for Uranium Hexafluoride for Enrichment
ASTM	C996-04	2004	Standard Specifications for Uranium Hexafluoride Enriched to Less than 5% ^{235}U
ASTM	D6646-03	2003	Standard Test Method for Determination of the Accelerated Hydrogen Sulfide Breakthrough Capacity of Granular and Pelletized Activated Carbon
ASTM	E84	2008	Standard Test Method for Surface Burning Characteristics of Building Materials
ASTM	E814	2008	Standard Test Method for Fire Tests of Penetration Firestop Systems
ASTM	E1168-95	2008	Standard Guide for Radiological Protection Training for Nuclear Facility Workers
CGA	G-5	2005	Hydrogen
CGA	H-5	2008	Installation Standards for Bulk Hydrogen Supply Systems
CGA	P-1	2008	Safe Handling of Compressed Gas in Cylinders
CGA	SB-2	2007	Safety Bulletin, Oxygen-Deficient Atmospheres, 4th Edition
IAEA	TS-R-1	2009	Regulations for the Safe Transport of Radioactive Material
IBC	2006	2006	2006 International Building Code,

Code Group / Reference	Code Number	Year or Edition	Title
ICC	NCBC	2009	2006 ICC International Plumbing Code, IPC w/2009 NC Amendments
ICC	NCBC	2009	2006 ICC International Mechanical Code, IMC w/2009 NC Amendments
ICC	NCBC	2009	North Carolina State Building Codes, Version 1.0, 2009 2006 ICC International Building Code w/2009 NC Amendments
ICC	NCFC	2009	North Carolina Fire Code, IFC - 2006 w/2009 NC Amendments
ICRP	68	1995	Dose Coefficients for Intakes of Radionuclides by Workers
IEEE	80	2000	Guide for Safety in AC Substation Grounding
IEEE	81	1983	Guide for Measuring Earth Resistivity, Ground Impendence and Earth Surface Potential of a Ground System
IEEE	142	2007	Grounding of Industrial and Commercial Power Stations
IEEE	323	2008	IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generation Stations
IEEE	344	2004	IEEE Recommended Practice for Seismic Qualification of 1E Equipment for Nuclear Power Generation Stations
IEEE	383	2003	IEEE Standard for Qualifying Electric Cables and Field Splices for Nuclear Generating Systems
IEEE	384	2008	IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits
IEEE	450	2002	IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications
IEEE	484	2002	IEEE Recommended Practice for Installation Design and Installation of Vented Lead-Acid Batteries for Stationary Applications
IEEE	485	2008	IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications
IEEE	519	1992	Recommended Practices and Requirements for Harmonic Control in Electrical Power Systems
IEEE	946	2004	IEEE Recommended Practice for the Design of DC Auxiliary Power Systems for Generating Stations
IEEE	1100	2005	Recommended Practice for Powering and Grounding Sensitive Electronic Equipment
IEEE	1202	2006	IEEE Standard for Flame Testing of Cables For Use in Cable Tray in Industrial and Commercial Occupancies

Code Group / Reference	Code Number	Year or Edition	Title
IEEE	N323	1978	American National Standard Radiation Protection Instrumentation Test and Calibration
NAVFAC	DM 7	1983	*Naval Facilities Engineering Command Design Manual,* Naval Facilities Engineering Command
NEMA	SG 4	2005	Alternating-Current High-Voltage Circuit Breaker
NEPA	--	1969	National Environmental Policy Act
NFPA	1	2009	Fire Code
NFPA	10	2002	Standard for Portable Fire Extinguishers
NFPA	13	2007	Installation of Sprinkler Systems
NFPA	14	2007	Standard for the Installation of Standpipes and Hose Systems
NFPA	20	2007	Standard for the Installation of Stationary Fire Pumps for Fire Protection
NFPA	22	2008	Standard for Water Tanks for Private Fire Protection
NFPA	24	2007	Standard for the Installation of Private Fire Service Mains and Their Appurtenances
NFPA	25	2008	Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems
NFPA	30	2008	Flammable and Combustible Liquids Code
NFPA	45	2004	Standard on Fire Protection for Laboratories Using Chemicals
NFPA	51	2007	Design and Installation of Oxygen-Fuel Gas Systems for Welding, Cutting, and Allied Processes
NFPA	51B	2009	Fire Prevention During Welding, Cutting, and Other Hot Work
NFPA	54	2009	National Fuel Gas Code
NFPA	55	2005	Storage, Use, and Handling of Compressed Gases and Cryogenic Fluids in Portable and Stationary Containers, Cylinders, and Tanks, with ERRATA 1 2006
NFPA	58	2008	Liquefied Petroleum Gas Code
NFPA	69	2008	Standard on Explosion Prevention Systems
NFPA	70	2008	National Electrical Code®
NFPA	70E	2009	Standard for Electrical Safety in the Workplace
NFPA	72	2007	National Fire Alarm Code®
NFPA	75	2009	Protection of Information Technology Equipment
NFPA	80	2007	Standard for Fire Doors and Other Opening Protectives

Code Group / Reference	Code Number	Year or Edition	Title
NFPA	80A	2007	Recommended Practice for Protection of Buildings from Exterior Fire Exposures
NFPA	90A	2009	Standard for the Installation of Air-Conditioning and Ventilating Systems
NFPA	90B	2009	Standard for the Installation of Warm Air Heating and Air-Conditioning Systems
NFPA	91	2004	Standard for Exhaust Systems for Air Conveying of Vapors, Gases, Mists and Noncombustible Particulate Solids
NFPA	92A	2006	Standard for Smoke-Control Systems Utilizing Barriers and Pressure Differences
NFPA	92B	2005	Standard for Smoke Management Systems in Malls, Atria, and Large Spaces
NFPA	101®	2009	Life Safety Code®
NFPA	105	2007	Standard for the Installation of Smoke Door Assemblies and Other Opening Protectives
NFPA	110	2005	Standard for Emergency and Standby Power Systems
NFPA	111	2005	Standard on Stored Electrical Energy Emergency and Standby Power Systems
NFPA	115	2008	Standard for Laser Fire Protection
NFPA	220	2009	Standard on Types of Building Construction
NFPA	221	2009	Standard for High Challenge Fire Walls, Fire Walls, and Fire Barrier Walls
NFPA	241	2009	Standard for Safeguarding Construction, Alteration, and Demolition Operations
NFPA	253	2006	Standard Method of test for Critical Radiant Flux for Floor Covering Systems Using a Radiant Heat Energy Source
NFPA	255	2006	Standard Method of Test of Surface Burning Characteristics of Building Materials
NFPA	497	2008	Recommended Practice for the Classification of Flammable Liquids, Gases, or Vapors and of Hazardous (Classified) Locations for Electrical Installations in Chemical Process Areas
NFPA	600	2005	Standard on Industrial Fire Brigades
NFPA	601	2005	Standard for Security Services in Fire Loss Prevention
NFPA	704	2007	Standard System for the Identification of the Hazards of Materials for Emergency Response
NFPA	780	2008	Standard for the Installation of Lightning Protection Systems

Code Group / Reference	Code Number	Year or Edition	Title
NFPA	801	2008	Standard for Fire Protection for Facilities Handling Radioactive Materials
NFPA	901	2006	Standard Classifications for Incident Reporting and Fire Protection Data
NFPA	1143	2009	Standard for Wildland Fire Management
NFPA	1144	2008	Standard for Reducing Structure Ignition Hazards from Wildfire
NFPA	1500	2007	Fire Department Occupational Safety and Health Program
NFPA	1620	2003	Recommended Practice for Pre-Incident Planning
NFPA	2001	2008	Standard on Clean Agent Fire Extinguishing Systems
NRC		2007	Environmental Assessment for Renewal of Special Nuclear Material License No. SNM-1097 General Electric Company Nuclear Energy Product Facility
NRC	Inspection Manual 0609	2005	Appendix F, Fire Protection Significance Determination Process
NRC	FCSS-ISG-08	Rev. 0	Natural Phenomena Hazards, Interim Staff Guidance Document for Fuel Cycle Facilities
NRC Reg. Guide	1.59	Rev. 2	Design Basis Floods for Nuclear Power Plants
NRC Reg. Guide	1.76	Rev. 1	Design-Basis Tornado and Tornado Missles for Nuclear Power Plants
NRC Reg. Guide	1.132	Rev. 2	Site Investigations for Foundations of Nuclear Power Plants
NRC Reg. Guide	1.180	Rev. 1	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems
NRC Reg. Guide	1.198	Rev. 0	Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites
NRC Reg. Guide	1.75	Rev. 3	Physical Independence of Electric Systems
NRC Reg. Guide	3.12	1973	General Design Guide for Ventilations Systems of Plutonium and Fuel Fabrication Plants
NRC Reg. Guide	3.67	Rev. 0	Standard Format and Content of Emergency Plans for Fuel Cycle and Materials Facilities
NRC Reg. Guide	3.71	2005, Rev. 1	Nuclear Criticality Safety Standards Fuels and Material Facilities

Code Group / Reference	Code Number	Year or Edition	Title
NRC Reg. Guide	4.16	1985	Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexaflouride Production Plants
NRC Reg. Guide	4.20	1996	Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees Other than Power Reactors
NRC Reg. Guide	4.21	2008	Minimization of Contamination and Radioactive Waste Generation: Life Cycle Planning
NRC Reg. Guide	8.2	Rev. 0	Guide for Administrative Practices in Radiation Monitoring
NRC Reg. Guide	8.7	Rev. 2	Instructions for Recording and Reporting Occupational Radiation Dose Data
NRC Reg. Guide	8.9	Rev. 1	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program
NRC Reg. Guide	8.10	Rev. 1-R	Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable
NRC Reg. Guide	8.13	Rev. 3	Instruction Concerning Prenatal Radiation Exposure
NRC Reg. Guide	8.15	Rev. 1	Acceptable Programs for Respiratory Protection
NRC Reg. Guide	8.24	1979, Rev 1	Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabriication
NRC Reg. Guide	8.25	Rev. 1	Air Sampling in the Workplace
NRC Reg. Guide	8.29	Rev. 1	Instruction Concerning Risks from Occupational Radiation Exposure
NRC Reg. Guide	8.34	Rev. 0	Monitoring Criteria and Methods to Calculate Occupational Radiation Doses
NRC Reg. Guide	8.37	Rev. 0	ALARA Levels for Effluents From Materials Facilities
NUREG	0700	2002, Rev. 2	Human-System Interface Design Review Guidelines
NUREG	0711	2004, Rev. 2	Human Factors Engineering Program Review Model
NUREG	1140	1988	A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees
NUREG	1278	1983	Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications
NUREG	1391	1991	Chemical Toxicity of Uranium Hexafluoride Compared to Acute Effects of Radiation

Code Group / Reference	Code Number	Year or Edition	Title
NUREG	1505	1998	A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys
NUREG	1513	2001	Integrated Safety Analysis Guidance Document
NUREG	1520	2002	Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility
NUREG	1575	2000	Multi-Agency Radiation Survey and Site Investigation Manual
NUREG	1748	2003	Environmental Review Guidance for Licensing Actions Associated with NMSS Programs
NUREG	1757	2006	Consolidated NMSS Decommissioning Guidance
NUREG	1887	2007	RASCAL 3.0.5: Description of Model and Methods,
NUREG/CR	4461	2007, Rev. 2	Tornado Climatology of the Contiguous United States
NUREG/CR	6410	1998	Nuclear Fuel Facility Cycle Accident Analysis Handbook
NUREG/CR	6928	2007	Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants
PCI	MNL-120 6th Edition	2004	Precast Concrete Institute Design Handbook: Precast and Pre-Stressed Concrete
SMACNA	006	2005	HVAC Duct Construction Standards - Metal and Flexible
SMACNA	1922	2004	Rectangular Industrial Duct Construction Standards
SMACNA	1520	1999	Rounded Industrial Duct Construction Standards
SMACNA	1143	2003	HVAC Air Duct Leakage Test Manual, First Edition
SMACNA	1780 3rd Edition	2002	HVAC Systems Testing, Adjusting, and Balancing
SMACNA	1958 4th Edition	2006	HVAC Systems Duct Design
UL	555	2010	Standard for Safety Fire Dampers
UL	555S	2010	Standard for Safety Smoke Dampers
UL	586	2009	Standard for Safety High-Efficiency, Particulate, Air Filter Units
UL	900	2007	Standard for Safety Air Filter Units
UL	1277	2001	Electrical Power and Control Tray Cables with Optional Fiber Members

Table 3-2. Integrated Safety Analysis Nodes.

Node Number / Designation	Node Description/Name
4100-00	Cylinder Storage and Handling
4200-00	Feed/Vaporization
4300-00	Product Withdrawal
4400-00	Tails Withdrawal
4500-00	*Intentionally Left Blank*
4600-00	Cascade / Gas Handling
4700-00	Blending
4800-00	Sampling
4900-00	Radioactive Waste (Liquid/Solid)
5000-00	HVAC/MCES
5100-00	Utilities
5200-00	Decontamination/Maintenance
5300-00	*Intentionally Left Blank*
5400-00	Laboratory Operations
5500-00	Laser System
5600-00	External Events
5700-00	Balance of Plant

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

Severity Ranking	Consequence Description		
	Workers	**Offsite Public**	**Environment**
3	Radiological dose greater than 1 Sv (100 rem)	Radiological dose greater than 0.25 Sv (25 rem)	N/A
	75 mg soluble uranium intake	30 mg soluble uranium intake	
	Chemical exposure greater than AEGL-3 (10 minute exposure)	Chemical exposure greater than AEGL-2 (30 minute exposure)	
	A criticality accident occurs	A criticality accident occurs	
	Dermal exposure from an HF solution that endangers the life of the worker	Dermal exposure to HF solution resulting in irreversible or other serious long-lasting effects	
2	Radiological dose greater than 0.25 Sv (25 rem) but less than or equal to 1 Sv (100 rem)	Radiological dose greater than 0.05 Sv (5 rem) but less than or equal to 0.25 Sv (25 rem)	Radioactive release greater than 5,000 times 10 CFR 20, Appendix B, Table 2
	Chemical exposure greater than AEGL-2 but less than or equal to AEGL-3 (10 minute exposure)	Chemical exposure greater than AEGL-1 but less than or equal to AEGL-2 (30 minute exposure)	
	Dermal exposure to HF solution resulting in irreversible or other serious long-lasting health effects	Dermal exposure from HF solution resulting in mild transient health effects	
	Direct eye contact with any HF solution (leads to irreversible or other serious long-lasting health effects)		
1	Accidents with radiological and/or chemical exposures to workers less than those above	Accidents with radiological and/or chemical exposures to the public less than those above	Radioactive releases to the environment producing effects less than those specified above

Sv = Sieverts

AEGL = Acute Exposure Guideline Level

The MSDS for chemicals used in the GLE process were reviewed for hazards to the workers. HF solution was determined to present a potential serious or long-lasting health hazard and is therefore included in above table. No other chemicals were identified as presenting potential serious or long-lasting health hazards as used in the GLE process.

Table 3-4. AEGL Thresholds from the EPA for Uranium Hexafluoride, Soluble Uranium, and Hydrogen Fluoride.

Uranium Hexafluoride [mg/m^3]					
	10 min	30 min	60 min	4 hr	8 hr
AEGL 1	3.6	3.6	3.6	NR	NR
AEGL 2	28	19	9.6	2.4	1.2
AEGL 3	216	72	36	9	4.5
Soluble Uranium [mg/m^3]					
	10 min	30 min	60 min	4 hr	8 hr
AEGL 1	2.4	2.4	2.4	NR	NR
AEGL 2	19	13	6.5	1.6	0.8
AEGL 3	145	48	24	6	3.0

Soluble Uranium = UF_6 x Uranium fraction [0.67]

Hydrogen Fluoride [mg/m^3]					
	10 min	30 min	60 min	4 hr	8 hr
AEGL 1	0.8	0.8	0.8	0.8	0.8
AEGL 2	78	28	20	10	10
AEGL 3	139	51	37	18	18

Table 3-5. What-If/Checklist Example.

GLE Commercial Facility		Site: Wilmington, North Carolina	Unit: TR-XXXX.XX	System:
Method: What-If/Checklist		Design Intent		
No: XX		Description:		

Item	What-If..?	Scenarios Initiators	Consequences	Cat	S	UL	UR	Safeguards	References

Table 3-6. Unmitigated Likelihood Categories.

Likelihood Category	Qualitative Description
1	Consequence Category 3 accidents must be “Highly Unlikely”
2	Consequence Category 2 accidents must be “Unlikely”
3	“Not Unlikely”

Table 3-7. Event Likelihood Categories.

	Likelihood Category	Frequency or Probability of Occurrence*
Not Unlikely (Credible)	**3**	More than or equal to 10^{-4} per-event per-year
Unlikely (Credible)	**2**	Between 10^{-4} and 10^{-5} per-event per-year
Highly Unlikely	**1**	Less than or equal to 10^{-5} per-event per-year

Note: Based on approximate order-of-magnitude ranges.

Table 3-8. Determination of Likelihood Category.

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

*The likelihood category is determined by calculating the likelihood index, T, then using this table. The term T is calculated as the sum of the indices for the events in the accident sequence.

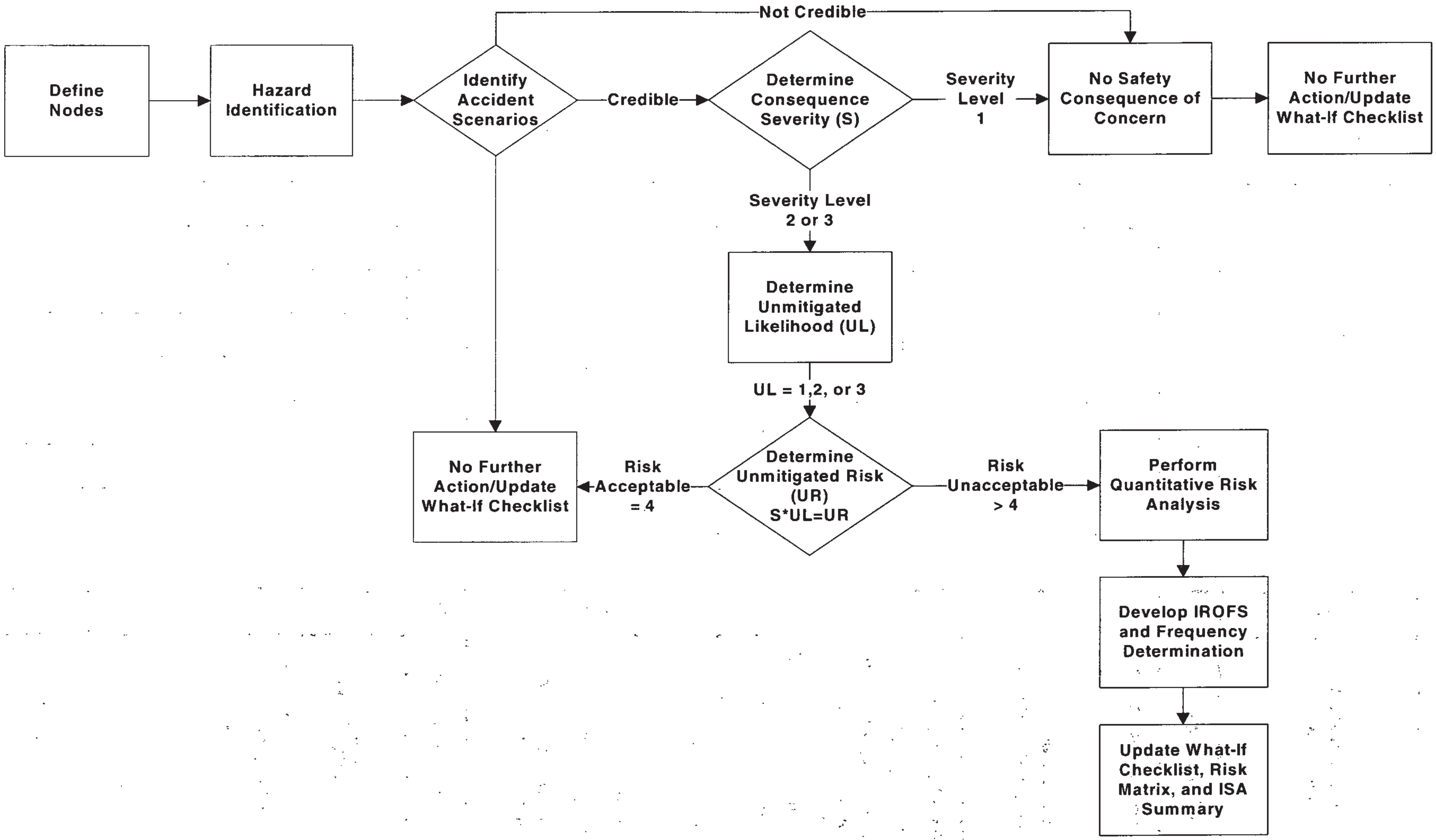
Table 3-9. Unmitigated Risk Assignment Matrix.

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 3-10. Accident Sequence Summary and Risk Index Evaluation.

Accident Identifier	Initiating Event	Initiating Event	Safety Parameter 1 or IROFS 1	Failure Probability Index 1	Preventive Safety Parameter 2 or IROFS 2	Failure Probability Index 2	Preventive Safety Parameter 3 or IROFS 3	Failure Probability Index 3	Likelihood Index T Uncontrolled / Controlled (c+e+g+i)	Likelihood Category	Consequence Evaluation Reference	Consequence Category	Risk Index (I=iXk)	Comments and Recommendations
(a)	(b)	(c)	(d)	(e)	(f)	(g)	(h)	(i)	(j)	(k)	(l)	(m)	(n)	(o)

Figure 3-1. Integrated Safety Analysis Process Flow Diagram.



CHAPTER 5
REVISION LOG

Rev.	Effective Date	Affected Pages	Revision Description
0	04/30/2009	ALL	Initial Application Submittal.
1	03/31/2010	7, 11-14, 18-20, 24, 27, 28, 31	Incorporate RAI responses submitted to the NRC via MFN-09-577 dated 09/04/2009 and MFN-09-801 dated 12/28/2009.
2	06/25/2010	19, 25, 26	Section 5.4.1.3.2 revised to include nonparametric method for determination of bias uncertainty, Section 5.4.4.5 revised to remove reference to MRA
3	12/17/2010	25	Section 5.4.4.5 revised to clarify terminology
4	03/10/2011	5, 12, 17, 20, 23, 24,	Added applicable revision number to ANS 8.1 reference. Section 5.3.5 revised to clarify how the CAAS is maintained and to provide more detail with regard to the scope and timing of compensatory measures upon loss of CAAS coverage. Replaced "MMS" with MoS".

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TABLES

NONE

FIGURES

NONE

5. NUCLEAR CRITICALITY SAFETY

5.1 MANAGEMENT OF THE NUCLEAR CRITICALITY SAFETY PROGRAM

5.1.1 Nuclear Criticality Safety Design Philosophy

In accordance with baseline design criterion (9) contained in 10 CFR 70.64(a), *Requirements for New Facilities or New Processes at Existing Facilities (Ref. 5-1)*, the design of fissile material processes must "provide for criticality control including adherence to the double contingency principle." The double contingency principle, as identified in American National Standard Institute (ANSI)/American Nuclear Society (ANS) 8.1-1998 (R2007), *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors (Ref. 5-2)*, is the fundamental technical basis for design and operation of fissile material processes within the GE-Hitachi Global Laser Enrichment LLC (GLE) Commercial Facility. As such, process designs shall incorporate sufficient margins of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible. As used in the double contingency principle, the term "concurrent" means: if the effect of the first process change persists until a second change occurs, an inadvertent nuclear criticality could result. It does not mean the two initiating events must occur simultaneously. The possibility of an inadvertent nuclear criticality can be markedly reduced if failures of nuclear criticality safety (NCS) controls are rapidly detected and processes rendered safe.

The established NCS design criteria and NCS reviews are applicable to: (1) new and existing processes, facilities, or equipment which process, store, transfer, or otherwise handle fissile materials; and (2) any change in existing processes, facilities, or equipment which may have an impact on the established basis for NCS. For fissile material operations, double contingency protection may be provided by either control of at least two independent parameters, or control of a single parameter using a system of multiple independent controls. The defense of one or more system parameters provided by at least two independent controls is documented in the GLE Criticality Safety Analyses (CSAs).

In accordance with the requirements contained in 10 CFR 70.61(d), *Performance Requirements (Ref. 5-3)*, "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions all nuclear processes are subcritical." The NCS Program evaluates each fissile material process to identify the normal and credible abnormal conditions, and establish the controls required to meet the double contingency design criteria. Use of the double contingency design criteria assures that all nuclear processes remain subcritical under credible conditions. As required in 10 CFR 70.62, *Safety Program and Integrated Safety Analysis (Ref. 5-4)*, the Integrated Safety Analysis (ISA) documents the credible accident sequences that could lead to an inadvertent nuclear criticality, and identifies the likelihood of occurrence for each potential accident sequence. For these credible accident sequences, the engineered and administrative NCS controls required to prevent an inadvertent nuclear criticality and meet the overall likelihood requirements specified in GLE LA Chapter 3, *Integrated Safety Analysis,* are designated as Items Relied on for Safety (IROFS). For each IROFS identified, appropriate management measures are applied to assure the control is available and reliable to perform its function when needed. The ISA methodology is described in GLE LA Chapter 3, and the ISA Summary.

5.1.2 Nuclear Criticality Safety Program Objectives

The NCS Program establishes and maintains NCS safety limits and operating limits for controlled parameters in nuclear processes. Qualified NCS personnel evaluate operations involving fissile material to determine the basis for safety of operation based on the assessment of both normal and credible abnormal conditions. Functional requirements for criticality safety controls are specified commensurate with the NCS design criteria, and management measures are applied to ensure the availability and reliability of the controls. The GLE NCS Program management commits to the following objectives:

- Develop, implement, and maintain an NCS Program that meets the regulatory requirements of 10 CFR 70, *Domestic Licensing of Special Nuclear Material (Ref. 5-5)*;
- Provide sufficient IROFS and defense-in-depth, and demonstrate an adequate margin of safety to prevent an inadvertent nuclear criticality in operations in which fissile material is present;
- Protect against the occurrence of accident sequences identified in the ISA Summary, which could result in an inadvertent nuclear criticality;
- Comply with NCS performance requirements in 10 CFR 70.61;
- Establish and maintain NCS controlled parameters and procedures;
- Establish and maintain NCS subcritical limits and operating limits for identified IROFS;
- Conduct NCS evaluations, herein referred to as CSAs, to assure under normal and credible abnormal conditions, fissile material processes remain subcritical and maintain an adequate margin of safety;
- Establish and maintain NCS postings, training, and emergency procedure training;
- Establish and maintain NCS IROFS, based on current NCS determinations;
- Adhere to NCS baseline design criteria requirements in 10 CFR 70.64(a), for new facilities and new processes at existing facilities requiring a license amendment under 10 CFR 70.72, *Facility Changes and Change Process (Ref. 5-6)*;
- Comply with NCS ISA Summary requirements in 10 CFR 70.65(b), *Additional Content of Applications (Ref. 5-7)*;
- Comply with NCS ISA Summary configuration management (CM) requirements in 10 CFR 70.72.

5.1.3 Evaluation of Nuclear Criticality Safety

As part of the design of new facilities, or significant additions or changes in existing facilities, the proposed design is reviewed and approved by the NCS function. Prior to operation of a new or modified facility/process, an evaluation is performed to demonstrate that the entire process will remain subcritical under both normal and credible abnormal conditions. When NCS considerations are impacted by a change, the NCS function recommends changes to the process parameter necessary to maintain safe operation of the facility, and specifies appropriate controls and management measures required for safety. The approval by the NCS function is required prior to operation of a new or modified facility/process. This NCS approval is documented in accordance with established practices and conforms to the CM Program described in GLE LA Section 11.1, *Configuration Management*.

GLE personnel initiate proposed changes to the facility (such as, design changes, changes to processes, operating and maintenance procedures, IROFS, and management measures) through use of a change request. Change requests are processed in accordance with approved written procedures. Change requests, which establish or involve a change in existing criticality safety parameters, require a Senior NCS Engineer to disposition the proposed change with respect to impacts to the safety basis and the need for a CSA. If a new analysis or a revision to an existing analysis is required, the change is not placed into operation until the CSA is complete and preoperational requirements specified by the NCS function are fulfilled. This assures that the documented safety basis is applicable to the current configuration of the facility.

The purpose of the CSA is to demonstrate compliance with 10 CFR 70.64(a)(9), the double contingency principle, through control of one or more parameters important to criticality safety. The parameters to be controlled and the controls on specified parameters are determined and evaluated in the CSA. The controls specified in the CSA may be passive engineered, active engineered, or administrative. Additional requirements for management measures such as postings, periodic inspections, and maintenance requirements are also specified in the CSA to assure the NCS controls are available and reliable. Application of the double contingency principle assures that the process will remain subcritical under normal and credible abnormal conditions.

5.2 ORGANIZATION AND ADMINISTRATION

5.2.1 General Organization and Administrative Methods

The GLE organizational structure and administrative practices have been established consistent with the guidance in ANSI/ANS 8.1-1998 (R2007) and ANSI/ANS 8.19-2005, *Administrative Practice for Nuclear Criticality Safety (Ref. 5-8)*. Organizational positions, experience, and qualification requirements of personnel and functional responsibilities are described in GLE LA Chapter 2, *Organization and Administration*, which includes an outline of the organizational relationships. The GLE Operations Organization shall be provided adequate resources to ensure an effective NCS Program is implemented.

5.2.2 Nuclear Criticality Safety Organization

The NCS function is administratively independent of the Operations Organization and has the authority to shutdown potentially unsafe operations. The NCS function consists of an NCS Manager responsible for implementation of the NCS Program, and at least one Senior NCS Engineer to allow independent reviews of NCS evaluations. Specific details of the responsibilities and qualification requirements for the NCS Manager, Senior NCS Engineer, and NCS Engineer are described in GLE LA Chapter 2.

NCS personnel are trained in the interpretation of data pertinent to NCS and are familiar with the operation of the GLE Commercial Facility prior to being qualified as a member of the NCS function. Training and qualification of NCS personnel is described in Section 5.3.1, *Training and Qualification of the Nuclear Criticality Staff.*

5.2.3 Operating Procedures

Fissile material operations are performed in accordance with approved written operating procedures. If personnel encounter a condition not covered by the operating procedure, the individual is required to safely stop the operation and report the defective condition to the NCS function, either directly or through Operations management. The operation may not be restarted until the NCS function has evaluated the situation and the necessary procedure instructions are provided. Operations personnel are trained in this procedural compliance policy.

Procedures that govern the handling of enriched uranium are reviewed and approved by the NCS function. The Operations Organization is responsible for developing and maintaining operating procedures that incorporate limits and controls established by the NCS function. GLE management assures operators and other affected personnel review and understand these procedures through postings, training programs, and/or other written, electronic, or verbal notifications.

Documentation associated with the review and approval of operating procedures, and operator training or orientation is maintained within the CM Program and further described in GLE LA Chapter 11, *Management Measures*.

5.2.4 Postings and Labeling

NCS requirements defined by the NCS function are made available at workstations in the form of approved written or electronic operating procedures, and/or clear visible postings. Postings may include the placement of signs and/or marking on walls, floors, or process equipment to summarize key NCS requirements and limits, to designate approved work and storage areas, or to provide instructions or specific precautions to personnel. Information that may be displayed on postings include: limits on material types and forms, allowable quantities by weight or number, required spacing between units, critical control steps in the operation, and control limits (when applicable) on quantities such as moderation, density, or enrichment. Storage postings are located in conspicuous places and include, as appropriate: material type, container identification, number of items allowed, and mass, volume, moderation, and/or spacing limits. In addition, when administrative controls or specific actions/decisions by operators are involved, postings include pertinent requirements identified within the CSA.

Where practical, fissile material containers are labeled such that the material type, ^{235}U enrichment, and gross and/or net weight can be clearly identified or determined. Exceptions to this labeling process include the following:

- Large process vessels in which the content is continuously changing;
- Shipping containers which are labeled as required for shipment;
- Uranium hexafluoride (UF_6) cylinders containing heels in which the net weight is known but the exact fissile content is not quantified;
- Containers of one liter volume or less, or where labeling is not practical;
- In limited circumstances, where the exact enrichment of the material contained is not known (for example, equipment cleanout material or sludge removed from sumps); and
- Waste boxes/drums and contaminated items in which the exact fissile content is very small and not quantified.

Where labeling does not indicate the exact material type, enrichment, and gross and/or net weight, other methods are used to identify the presence of fissile material such as postings, procedures, and training.

5.3 NUCLEAR CRITICALITY SAFETY MANAGEMENT MEASURES

5.3.1 Training and Qualifications of the Nuclear Criticality Safety Staff

Training and qualification of NCS staff is conducted consistent with the guidance in ANSI/ANS 8.26-2007, *Criticality Safety Engineer Training and Qualification Program* (Ref. 5-9). As such, GLE has established a formalized NCS Engineer Training and Qualification Program that is periodically reviewed and maintained by the qualified NCS engineers. This program includes on-the-job training (OJT), demonstration of proficiency, periodic required technical classes or seminars, and participation in offsite professional development activities.

The NCS Engineer Training and Qualification Program content emphasizes on-the-job experience to fully understand the processes, procedures, and personnel required to assure that NCS controls on identified NCS parameters are properly implemented and maintained.

5.3.2 Auditing, Assessing, and Upgrading the Nuclear Criticality Safety Program

NCS audits and assessments are performed consistent with the guidance in ANSI/ANS 8.19-2005. Details of the GLE NCS Audit and Assessment Program are described in GLE LA Section 11.5, *Audits and Assessments*.

NCS audits are conducted by approved NCS personnel and documented in accordance with approved written procedures. Findings, recommendations, and observations are reviewed with the GLE Environmental, Health, and Safety (EHS) Manager to determine if other safety impacts exist. NCS audit findings are transmitted to applicable line managers and area managers for appropriate action and are tracked to completion.

NCS professionals, independent of GLE NCS personnel, conduct periodic NCS Program reviews. The program review provides a means to independently assess the effectiveness of GLE NCS Program components. The audit team is composed of individuals recommended by the NCS Manager, and the team's audit qualifications are approved by the GLE Facility Manager or GLE EHS Manager. Audit results are reported in writing to the NCS Manager, who disseminates the report to line management. Results in the form of corrective action requests are tracked to completion.

5.3.3 Integrated Safety Analysis Summary Revisions and the Nuclear Criticality Safety Program

In accordance with ANSI/ANS 8.19-2005, the CSA is a collection of information that "provides sufficient detail, clarity, and lack of ambiguity to allow independent judgment of the results." The CSA documents the safety basis for the defined fissile process, establishes the subcritical limits on associated controlled parameters, and establishes controls on said parameters to satisfy the double contingency principle.

Documented CSAs are controlled elements of the ISA methodology described in GLE LA Chapter 3 and the ISA Summary. The CSA establishes the NCS bases for a particular system under normal and credible abnormal conditions. CSAs are prepared or updated for new or significantly modified fissile units, processes, or facilities within the GLE Commercial Facility in accordance with the established CM Program described in GLE LA Chapter 11. When a facility change requires a CSA to be re-evaluated or modified, the modifications are carefully evaluated for effects on the ISA Process Hazards Analysis (PHA) and ISA Summary. Likewise, when changes are made to the PHA or ISA Summary, the changes are evaluated for effects on the documented CSAs. Documentation of the ISA Team review and approval of changes made to the PHA or ISA Summary is maintained in accordance with the CM Program.

5.3.4 Modifications to Operating and Maintenance Procedures

Operating and maintenance procedures are maintained consistent with the guidance in ANSI/ANS 8.19-2005. The Operations Organization is responsible for developing and maintaining operating procedures that incorporate limits and controls established by the NCS function. GLE management assures that appropriate GLE personnel and contractors review and understand these procedures through processes such as postings, training programs, and/or other written, electronic, or verbal notifications.

Procedures that govern the operation and maintenance of equipment involved in fissile material processes are reviewed and approved by the NCS function. Based on the review, the NCS function verifies that the required limits and controls have been incorporated into the procedure. In addition, the NCS function assures no single, inadvertent departure from a procedure could cause an inadvertent nuclear criticality and recommends modifications to the procedures to reduce the likelihood of occurrence of an inadvertent nuclear criticality. Documentation of the procedure review and approval process is maintained as described in GLE LA Sections 11.1 and 11.4.

5.3.5 Nuclear Criticality Accident Alarm System

The Criticality Accident Alarm System (CAAS) is designed and maintained to ensure compliance with requirements in 10 CFR 70.24, *Criticality Accident Requirements (Ref. 5-10)*, and ANSI/ANS 8.3-1997, *Criticality Accident Alarm System (Ref. 5-11)* as modified by Regulatory Guide 3.71, *Nuclear Criticality Safety Standards for Fuels and Material Facilities (Ref. 5-12)*. An evaluation that demonstrates compliance with the CAAS requirements of 10 CFR 70.24 is documented and maintained under CM. The location and spacing of the detectors are selected taking into account shielding by massive equipment or materials. Spacing between detectors is reduced where high-density building materials such as brick, concrete, or grout-filled cinder block shield a potential accident area from the detector. Low-density materials of construction, such as wooden stud construction walls, plaster, or metal corrugated panels, doors, non-load walls, and steel office partitions, are accounted for with conservative modeling approximations in determining detector placement.

The CAAS initiates immediate evacuation of the facility to ensure radiation exposure to workers is minimized. Employees are trained to recognize the evacuation signal and to evacuate promptly to a designated safe location. This system and proper response protocol is described in the GLE Radiological Contingency and Emergency Plan (RC&EP). Emergency response planning, procedures, and training to address an inadvertent criticality are consistent with the guidance in ANSI/ANS 8.23-1997, *Nuclear Criticality Accident Emergency Planning and Response (Ref. 5-13).*

GLE commits to having a CAAS that:

- Has components that are located or protected to minimize damage in case of fire, explosion, corrosive atmosphere, or other credible extreme conditions;

- Is designed to minimize the potential failure, including false alarms, due to human error and has major system components labeled;

- Is designed to remain operational in the event of seismic shock equivalent to the requirements of the International Building Code;

- Is uniform throughout the facility for the type of radiation detected, mode of detection, alarm signal, and system dependability;

- Provides coverage in each area that needs CAAS coverage by a minimum of two detectors; and

- Is clearly audible in areas that must be evacuated, or provides alternate visual notification methods documented to be effective in notifying personnel of a necessary evacuation.

The CAAS is maintained through routine response checks, and scheduled functional tests, and periodic instrument calibrations conducted in accordance with approved written procedures. In the event of loss of normal power, emergency power is automatically supplied to the CAAS. In the event that CAAS coverage is lost and not restored to an area, affected operations are promptly rendered safe. In this context, promptly means that the actions are initiated within one hour and completed consistent with the evaluations associated with the activity and completion times. The situation is initially rendered safe by shifting modes to the Standby Mode in the affected area or by suspension of activities with the potential to result in an inadvertent nuclear criticality within four hours, unless longer time periods have been determined and justified in advance. Continuing protection for the duration of the CAAS coverage loss is accomplished by implementing compensatory measures (e.g., self-alarming dosimetry, personnel access restriction to affected area, etc.) or by restoring equivalent coverage with a portable CAAS unit(s) for continued operation in any mode. Selection of compensatory measures is to be consistent with the extent and cause of the outage. In the Standby Mode, fissile material process systems are idled (no significant movement or enrichment of process gas) and manual movement, handling, or processing of fissile materials outside of process systems is secured.

5.3.5.1 CAAS Exemption Basis

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

10 CFR 70.17, *Specific Exemptions (Ref. 5-14),* allows the U.S. Nuclear Regulatory Commission (NRC), upon application of any interested person or upon its own initiative, to grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. The requested exemption is authorized by law because there is no statutory provision prohibiting the grant of the exemption. The requested exemption will not endanger life or property or the common defense and security and is otherwise in the public interest for the reasons discussed below. Exemption from CAAS coverage is requested for each of the following locations based on the discussion presented.

5.3.5.1.1 UF_6 Cylinder Storage Pads

The Tails and In-Process Pads are used for storage of source material only (not SNM) and therefore would not require CAAS coverage according to the regulations. Although a potential exists for storing UF_6 cylinders containing SNM on these pads (a wrong cylinder event), the 30B and 48GLE model cylinders are sufficiently different due to size, in the case of the 30B, and in color, in the case of the 48GLE, that such upsets will be immediately identifiable and correctable. Controls exist prior to material being stored on the cylinder pads to prevent such a mishap. 30B model cylinders are stored on the product pad and contain 5 wt% ^{235}U, or less, enriched material. 48GLE model cylinders are stored under CAAS coverage in the Cylinder Shipping and Receiving Area. Transport, handling, and storage of the 30B model cylinder, only involves solid UF_6 and is doubly contingent based on the robust nature of the container, routine certification of the cylinders, and on post-handling inspections that verify the integrity of the cylinder.

- UF_6 cylinder vessel is engineered to be "leak-tight" containers that prevent moderating materials from entering the cylinder. The packaging shall consist of bare metal cylinders (no protective overpacks required), which are designed, fabricated, inspected and marked in accordance with ANSI N14.1, *Nuclear Materials – Uranium Hexafluoride – Packaging for Transport (Ref. 5-15),* standard in effect at the time of manufacture.
- Cylinder integrity is verified through routine operational and periodic inspections and testing pursuant to ANSI N14.1 standard in effect at the time of action.
- To prevent cylinder breach (loss of cylinder integrity), only approved overhead crane rigging, forklift, or transport carrier is used for handling UF_6 cylinders in accordance with approved procedures and authorized trained personnel.
- The robust design of the 30B model cylinders are established as defense-in-depth criticality safety controls to ensure the health and safety of the public and workers and are maintained by the GLE Quality Program to applicable ANSI standards.

Evaluation of historical data associated with 30B model cylinder handling also concludes that the cylinders have not been damaged as frequently as 48-inch cylinders of any make (due in part because fewer 30B model cylinders are handled, 30B model cylinders are stacked only one-high, 30B model cylinders have a shorter storage period, and 30B model cylinders are smaller and lighter than 48-inch cylinders). Further, most 48-inch cylinder failures have been small and healed with uranium tetrafluoride (UF_4), hydrated uranyl fluoride ($UO_2F_2 \cdot x H_2O$), and corrosion product "patches" that significantly slowed further intrusion from water (liquid or vapor) *(Ref. 5-16)*.

Due to the high turn-around rate of 30B model cylinders in use, failure to identify corrosion type cylinder wall failures is judged highly unlikely. Evaluation of puncture events to these cylinders have concluded that under maximum rainfall rates for the region, the time to accumulate enough water in a 30B model cylinder to support criticality ranges from 2 to 8 days for very conservatively postulated 12- to 6-inch diameter holes that are difficult to miss during post-handling inspections. Further, these evaluations were conservatively based on an enrichment of 8 wt% ^{235}U and not the approved 5 wt% ^{235}U, or less, approved for 30B model cylinders stored on the Product Pad.

Administrative controls require damage to be remediated within eight hours of identification of the post-handling inspection. Lastly, the Product Pad is not a continuously occupied area. Personnel enter the area only to move 30B cylinders to and from the pad and to inspect cylinders and the cylinder yards to satisfy the requirements for various programs (Material Control and Accounting [MC&A], Quality Assurance [QA], and Fire Protection [FP]).

5.3.5.1.2 Trailer Storage Area

UF_6 cylinders temporarily stored in this area are packaged according to U.S. Department of Transportation (DOT) requirements in over-packs (for SNM containing 30B cylinders) and, as such, have undergone substantial evaluation to evaluate the accidental criticality hazard and assure that the packaging system provides conservative protection against accidental criticality to preclude the need for CAAS in transit.

5.3.5.1.3 UF_6 Cylinder Staging Area

UF_6 cylinders handled in this area are in the process of being packaged in an over-pack (for SNM containing 30B cylinders). The cylinders are either in the DOT packaged state or continuously monitored until the packaging is complete or the cylinder is removed to the Cylinder Shipping and Receiving Area (a CAAS covered area). Any mishaps that occur are immediately identified by the DOT packaging inspection process and corrective actions taken to remediate any hazard identified. Packaging activities are not performed in this area during rain (this requirement is driven by the need to perform radiological surveys on "dry" surfaces of the cylinders, shipping packages, and truck) where moderation control failure can occur during a cylinder mishap. Once packaged according to DOT requirements in an approved over-pack, the staging area is a location for temporary storage until the trailer is moved to Over Road Truck Trailer Storage Area for shipment.

In addition to the above features for safe storage of the cylinders to preclude accidental criticality, the increased vehicular and pedestrian traffic in support of CAAS maintenance and calibration requirements in these areas would cause a subsequent increased likelihood for impact events involving cylinders. CAAS detection clusters are required to be mounted high over the storage areas and the calibration and maintenance activity causes additional vehicular traffic in the area and introduces new drop hazards (bucket truck or man-lift collapse) that do not otherwise exist. This equipment and traffic increases the likelihood for fire and impact events on the UF6 Cylinder Pads and this places workers at a higher risk for injury and exposure relative to the mitigative value provided by the activation of the CAAS.

5.3.6 Corrective Action Program

A regulatory compliance tracking system is used to track planned corrective or preventive actions in regard to procedural, operational, regulatory, or safety-related deficiencies. NCS Program management assures that unacceptable performance deficiencies, which could result in an inadvertent nuclear criticality, are addressed using the Corrective Action Program. The Corrective Action Program is described in GLE LA Section 11.6, *Incident Investigations*.

5.3.7 Nuclear Criticality Safety Records Retention

Records of CSAs are maintained in sufficient detail and form to permit independent review and audit of the calculation method and results. Such records are retained during the conduct of activities and in accordance with approved written procedures following cessation of such activities. Records of employee nuclear safety training and NCS related documents under configuration control are maintained as described in GLE LA Section 11.7, *Records Management*.

5.4 NUCLEAR CRITICALITY SAFETY METHODOLOGIES AND TECHNICAL PRACTICES

5.4.1 Nuclear Criticality Safety Analysis Methods

5.4.1.1 K_{eff} Limits

Validated analytical methods may be used to evaluate individual process operations or potential system interaction. When analytical methods are used, the effective neutron multiplication factor (k_{eff}) of the system, plus three times the standard deviation of the analytical method, must be less than or equal to the established upper subcritical limit (USL) for both normal and credible process upset (accident) conditions; that is:

$$k_{eff} + 3\sigma \leq USL$$

Normal operating conditions assume the optimum credible conditions (that is, most reactive) expected to be encountered when the criticality control systems function properly. Credible process upsets assume optimum credible conditions anticipated for each off-normal or credible accident condition, and must be demonstrated critically safe in accordance with Section 5.1.1, *Nuclear Criticality Safety Design Philosophy*. The NCS function derives safety limits and operating limits by using these criteria to ensure processes remain subcritical under both normal and credible abnormal conditions. Safety and operating limits are established with sufficient margin of safety taking into consideration variability and uncertainty in process parameters under control to protect against a limit being accidentally exceeded. The sensitivity of key controlled parameters are evaluated with respect to the effect on k_{eff} for each system to assure adequate criticality safety controls are defined for the analyzed system. These studies are performed to correlate the change in k_{eff} that occurs as a result of a change to a controlled parameter.

5.4.1.2 Analytical Methods

Methodologies currently employed by the NCS function include hand calculations utilizing published experimental data (such as, ARH-600, Criticality Handbook [*Ref. 5-17*]), and Monte Carlo codes (specifically, Geometry Enhanced Merit [GEMER]) that utilize stochastic methods to approximate a solution to the three-dimensional neutron transport equation. Additional Monte Carlo code packages (such as, SCALE, MCNP) or S_n Discrete Ordinates codes (such as, ANISN, DORT, TORT, or the DANTSYS code package) may be used after validation has been performed as described in Section 5.4.1.3, *Validation Techniques*, and Section 5.4.1.4, *Validation Reports*.

The primary analytical method used for GLE criticality calculations is the GEMER Monte Carlo Program. GEMER is a multi-group Monte Carlo Program that approximates a solution to the neutron transport equation in three-dimensional space. The GEMER Criticality Program is based on 190-energy group structure to represent the neutron energy spectrum. In addition, GEMER treats resolved resonances explicitly by tracking the neutron energy and solving the single-level Breit-Wigner Equation at each collision in the resolved resonance range in regions containing materials whose resolved resonances are explicitly represented. The cross-section treatment in GEMER is especially important for heterogeneous systems since the multi-group treatment does not accurately account for resonance self-shielding.

5.4.1.3 *Validation Techniques*

The validity of the calculational method (computer code and nuclear cross-section data) used for the evaluation of NCS must be demonstrated and documented in validation reports according to approved written procedures. The validation of the computer code must determine its calculational bias, bias uncertainty, and the minimum margin of subcriticality safety (MoMS) using well-characterized and adequately documented critical experiments. The following definitions apply to the documented validation report(s):

Bias – The systematic difference between calculated results and experimentally measured values of k_{eff} for a fissile system.

Bias Uncertainty – The integrated uncertainty in experimental data, calculational methods, and models estimated by a valid statistical analysis of calculated k_{eff} values for critical experiments.

Minimum Margin of Subcriticality Safety (MMoS) – An allowance for any unknown (or difficult to identify or quantify) errors or uncertainties in the method of calculating k_{eff}, that may exist beyond those which have been accounted for explicitly in calculating bias and bias uncertainty.

GLE validation methodologies are consistent with the guidance in ANSI/ANS 8.1-1998 (R2007) and ANSI/ANS 8.24-2007, *Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations (Ref. 5-18)*. In accordance with the requirements of these national consensus standards, the GLE criteria to establish subcriticality requires the calculated k_{eff} to be less than or equal to an established USL, as presented in the validation report, for a system or process to be considered subcritical. The validation of the calculational method and cross-sections considers a diverse set of parameters that include, but are not limited to:

- Fuel enrichment, composition, and form of associated uranium materials,
- Homogeneity or heterogeneity of the system,
- Presence of neutron absorbing materials,
- Characterization of the neutron energy spectra,
- Types of neutron moderating materials,
- Types of neutron reflecting materials,
- Degree of neutron moderation in the system (such as, H/fissile atom ratio), and
- Geometry configuration of the system (such as, shape, size, spacing, reflector).

Within the validation, various areas of applicability are established based on parameters having a significant effect on the calculation of k_{eff}, bias, and bias uncertainty. The areas of applicability are established by grouping experiments with common parameters of importance to determine bias and bias uncertainty. Parameters with a significant effect on the calculation include: (1) neutron energy spectrum; (2) neutron absorbing materials; and (3) heterogeneity (for low-enriched uranium [LEU] systems). Based on these known parameters of importance, a typical grouping of areas of applicability for a validation may be as follows:

- Homogeneous LEU systems (thermal spectrum),
- Heterogeneous LEU systems (thermal spectrum),
- Common absorber systems (such as, boron, cadmium, gadolinium).

In performing CSA, the appropriate area of applicability shall be applied based on a comparison of parameters being evaluated to parameters covered by the area of applicability. For GLE Commercial Facility Operations, the most common area of applicability is homogeneous LEU systems based on the fact that materials evaluated are typically: (1) homogeneous (uranium hexafluoride and uranyl fluoride); (2) low-enriched (≤ 10 wt% ^{235}U); and (3) slightly to optimally moderated (thermal spectrum). When applying the validation outside an area of applicability, justification must be provided in the CSA. The selection of critical experiments, for each identified area of applicability of the NCS computer code validation, incorporates the following considerations:

- Experimental data for validation is assessed for completeness, accuracy, and applicability to operations prior to selection and use as a critical benchmark.
- Selection of experiments must encompass appropriate parameters spanning the range of normal and credible abnormal conditions that are anticipated to be evaluated using the calculational method.
- To minimize systematic error, benchmark data selected for validation are drawn from multiple, independent series, and sources of critical experiments. The range of parameters characterized by selected critical experiments is used to define the area of applicability for the code.
- The calculational method used to analyze the set of critical benchmarks incorporates the same analytic techniques used to analyze systems or processes to which the validation is applied.
- Data outliers in results obtained for the critical experiments selected for the validation may only be rejected based upon inconsistency of the data with known physical behavior.

The calculational bias, bias uncertainty, and USL over each defined area of applicability are determined by statistical methods as described in the following sections.

5.4.1.3.1 Calculational Bias

The bias is determined either as a constant, if no trends exist, or as a smooth and well-behaved function of a selected characteristic parameter (for example, hydrogen-to-fissile ratio) by regression analysis. Regression analysis may be used when trends exist with parameters statistically significant over the area of applicability.

Bias is determined from the calculated benchmark k_{eff} data, which are weighted using the overall uncertainty of each calculated data point. The overall uncertainty accounts for calculation uncertainty and benchmark uncertainty. Bias is applied over its negative range and assigned a value of zero over its positive range.

5.4.1.3.2 Bias Uncertainty

The bias uncertainty may be estimated using one of the following statistical methods. The details of each statistical method are documented in the validation report.

<u>Single-Sided Lower Confidence Band (SSLCB)</u>: Estimates bias uncertainty to ensure, at a 95% level of confidence, a future calculation of k_{eff} for a critical system or process is actually above the lower confidence limit. The SSLCB may be used when there is a clear trend in the calculated critical benchmark results.

<u>Single-Sided Lower Tolerance Band (SSLTB)</u>: Estimates the bias uncertainty to ensure, at a 95% level of confidence, at least 95% of future calculations of k_{eff} for critical systems or processes are actually above the lower tolerance limit. The SSLTB may be used when there is a clear trend in the calculated critical benchmark results.

<u>Single-Sided Lower Tolerance Limit (SSLTL)</u>: Estimates the bias uncertainty to ensure, at a 95% level of confidence, at least 95% of future calculations of k_{eff} for critical systems or processes are actually above the lower tolerance limit. The SSLTL is used when there are no trends apparent in the calculated critical benchmark results.

<u>Non-Parametric Method</u>: Estimates the bias uncertainty to ensure, at a 95% level of confidence, that future calculations of k_{eff} for critical systems or processes are actually above the lower tolerance limit. This statistical technique is based on a rank order analysis of the data. When the sample size is insufficient to obtain a 95% confidence level using the statistical method, additional non-parametric margin is applied to assure the desired degree of confidence is achieved. The non-parametric technique is applied in cases where the calculated critical benchmark results (non-trending data) or the residuals of bias regression (trending data) fail the normality test.

5.4.1.3.3 Data Normality

Where no trends to a characteristic parameter exist (SSLTL), the normality of calculated k_{eff} values for the set of critical experiments must be verified prior to estimation of bias and bias uncertainty. Where trends to a characteristic parameter do exist (SSLCB and SSLTB), normality of the regression analysis residuals must be verified prior to estimation of the bias and bias uncertainty.

5.4.1.3.4 Upper Subcritical Limit (USL)

The USL is established based on calculated bias, bias uncertainty, and MMS MoS for the area of applicability as follows:

$$USL = 1 + bias - bias\ uncertainty - MMoS$$

At GLE, a minimum *MoMS* = 0.03 is used to establish acceptance criteria for criticality calculations, which compared to the uncertainty in calculated k_{eff} values, is large.

The following acceptance criteria, considering worst-case credible accident conditions, must be satisfied, when using k_{eff} calculations by Monte Carlo methods, to establish subcritical limits for the GLE Commercial Facility:

$$k_{eff} + 3\sigma \leq USL$$

where σ is the standard deviation of the k_{eff} value obtained from the calculational method.

5.4.1.4 Validation Reports

Validation reports are documented, reviewed, and approved for each analytical method used to derive NCS limits. Validation reports are created, revised, reviewed, and approved by the NCS function and are controlled under the CM Program. The following requirements apply to Validation reports documented by the NCS function:

- Describe the NCS analytical method to which the validation applies.
- Clearly describe the theory of the validation methodology in sufficient detail to allow understanding of the methodology and independent duplication of results.
- Describe the mathematical and statistical operations used in the validation methodology to determine bias and bias uncertainty, including statistical testing performed to verify the acceptability of results.
- Provide a description or summary of the benchmark experiments or critical experiments selected for the validation, which indicate experiment characteristics important to the area of applicability and a reference to reliable experimental data.
- Identify the bias, uncertainty in the bias, uncertainty in calculated data, uncertainty in the benchmark experiments, and margin of subcriticality. If the derived bias is positive, it must be assigned a value of zero.
- Summarize the range in (or values of) NCS parameters describing the area of applicability. The area of applicability should be consistent with the values of parameters used in selected benchmark experiments. Any extrapolation beyond the area of applicability should be supported by an established mathematical methodology or sound engineering judgment. The mathematical method used to determine the acceptable extrapolation limit for a regression model is the leverage statistic. The leverage statistic is a measure of the distance between the extrapolation point for a predication and the

mean of trending parameter values in the critical benchmark data set. For a predication by extrapolation to be considered reliable with the predefined confidence level, its leverage value should not exceed the largest leverage value in the benchmark data set.

- Provide a description of the analytical method verification process and assurance that only verified software and hardware are used in the validation process.

5.4.1.5 Computer Software and Hardware Configuration Control

The software and hardware used within the criticality safety calculational system is configured and controlled in accordance with CM approved written procedures. Software changes are conducted in accordance with CM Program described in GLE LA Section 11.1.

Software, designated for use in NCS, are compiled into working code versions with executable files traceable by length, time, date, and version. Working code versions of compiled software are validated against critical experiments using an established methodology with differences in experiment and analytical methods being used to calculate bias and uncertainty values to be applied to the calculational results.

Each individual workstation is verified to produce results equivalent to the development workstation prior to use of the software for criticality safety calculation demonstrations on the production workstation. The verification results are documented for each individual workstation. Modifications to software and nuclear data affecting the calculational logic require re-validation of the software. Modifications to hardware or software that do not affect calculational logic are followed by code operability verification; in which case, selected calculations are performed to verify equivalent results from previous verifications. Deviations noted in code verification that may alter the bias or uncertainty requires re-qualification of the code prior to release for production use.

5.4.2 Control Practices

CSAs identify specific independent controls necessary to provide safe double contingent protection of a process. As discussed in Section 5.1.1, controls identified in the CSA are selected to assure no single credible event or failure can result in a criticality accident. As such, it is demonstrated that the process will remain subcritical under both normal and credible abnormal conditions. Prior to use in any enriched uranium process, NCS controls are verified against CSA criteria. The ISA methodology described in GLE LA Chapter 3 implements performance based management of process requirements and specifications important to NCS.

5.4.2.1 Verification and Maintenance of Controls

Reliable methods and instruments are used when NCS parameters are controlled by measurement. To assure continued reliability, required periodic verification and maintenance of controls are performed as described in GLE LA Section 11.2, *Maintenance*. The purpose of the verification program is to ensure the controls selected and installed fulfill the requirements identified in the CSA.

Processes are examined in the "as-built" condition to validate safety design and to verify the installation conforms to control specifications identified in the CSA. NCS personnel observe or monitor the performance of initial functional tests, and conduct preoperational audits to verify the controls function as intended, and the installed configuration agrees with the control specifications identified in the CSA. Operations personnel are responsible for subsequent verification of controls through the use of periodic functional testing or verification. When necessary, control calibration and routine maintenance are normally provided by the Instrument and Calibration and/or Maintenance functions. The purpose of the Maintenance Program is to ensure that the effectiveness of NCS controls designated for a specific process are maintained at the original level of intent and functionality. This requires a combination of routine maintenance, functional testing, and verification of design specifications on a periodic basis.

Verification and maintenance activities are performed per established practices documented through the use of forms and/or computer tracking systems. NCS personnel randomly review control verifications and maintenance activities to assure controls remain effective. Details of the Maintenance Program are described in GLE LA Section 11.2.

5.4.2.2 Consideration of Material Composition (Heterogeneity)

The CSA for each process determines the effects of material composition (for example, type, chemical form, physical form) within the process being analyzed, and identifies the basis for selection of compositions used in subsequent system modeling activities. In considering material composition, it is especially important to distinguish between homogeneous and heterogeneous system conditions. Heterogeneous effects are particularly relevant for LEU processes where all other parameters being equal; heterogeneous systems are typically more reactive than homogeneous systems. Systems involving uranium hexafluoride and uranyl fluoride are typically homogeneous; however, solid forms of uranium oxides may be heterogeneous. Evaluation of systems where the particle size varies must take into consideration effects of heterogeneity, as appropriate, for the process being analyzed.

5.4.3 Means of Control

The relative effectiveness and reliability of controls are considered during the CSA process. Passive engineered controls are preferred over other system controls and are utilized when practical and appropriate. Active engineered controls are the next preferred method of control. Administrative controls are the least preferred; however, augmented administrative controls are preferred over simple administrative controls. A criticality safety control must be capable of preventing a criticality accident independent of operation or failure of any other criticality control for a given credible initiating event.

5.4.3.1 Passive Engineered Controls

A device using only fixed physical design features to maintain safe process conditions without any required human action. Assurance is maintained through specific periodic inspections or verification measurement(s), as appropriate.

5.4.3.2 Active Engineered Controls

A physical device using active instrumentation, electrical components, or moving parts to maintain safe process conditions without any required human action. Assurance is maintained through specific periodic functional testing, as appropriate. Active engineered controls are designed to be fail-safe (that is, failure of the control results in a safe condition).

5.4.3.3 Administrative Controls

Either an augmented administrative control or a simple administrative control as defined herein:

Augmented Administrative Control – A procedurally required or prevented human action, combined with a physical device, which alerts an operator when action is needed to maintain safe process conditions or otherwise adds substantial assurance of the required human performance.

Simple Administrative Control – A procedural human action prohibited or required to maintain safe process conditions.

Use of administrative controls is limited to situations where passive and active engineered controls are not practical. Administrative controls may be proactive (requiring action prior to proceeding) or reactive (proceeding unless action occurs). Proactive administrative controls are preferred. Assurance is maintained through periodic verification, audit, and training.

5.4.4 Control of Parameters

NCS is achieved by controlling one or more parameter(s) of a system within established subcritical limits. The CM Program may require NCS staff review of proposed new or modified processes, equipment, or facilities to ascertain impact on controlled parameters associated with the particular system. Assumptions relating to processes, equipment, or facility operations, including material composition, function, operation, and credible upset conditions, are justified and documented in the CSA and independently reviewed.

Identified below are specific controlled parameters, which include mass, geometry, enrichment, reflection, moderation, concentration, interaction, neutron absorption, and process characteristics that may be considered during the NCS review process.

5.4.4.1 Mass

Mass control may be used for NCS control alone or in combination with other control methods. Mass control may be utilized to limit the quantity of uranium within specific process operations or vessels and within storage, transportation, or disposal containers. Mass may be controlled by direct measurement (for example, use of certified scales) through the use of fixed geometric dimensions and the assumption of a conservative fissile material density, or by using analytical or non-destructive methods.

Establishment of mass limits involves consideration of enrichment, potential moderation, reflection, geometry, spacing, and material composition. The CSA considers normal operations and credible process upsets in determining actual mass limits for the system and for defining additional controls. When only administrative controls are used for mass-controlled systems, double batching is considered to ensure adequate safety margin.

Where mass is the only parameter being controlled, and double batching is considered credible, the mass of any single accumulation shall not exceed either: (1) a safe batch, which is defined to be 45 percent of the minimum critical mass; or (2) 50 percent of the safe mass limit derived using validated analytical methods and an approved MoMS.

Where mass is one of two parameters being controlled, or where engineered controls prevent over batching, the mass of any single accumulations shall not exceed either: (1) 75 percent of the minimum critical mass; or (2) the safe mass limit derived using validated analytical methods and an approved MMSMoS.

When experimental data from published handbooks are used for mass limits, the following assumptions are applicable to the minimum critical mass: (1) spherical geometry; (2) full water reflection; (3) optimal moderation content; and (4) maximum credible enrichment. In addition, the chemical and physical form specified in the handbook must be at consistent with, or more restrictive than, that which may be present in the actual system to which the limit will be applied.

5.4.4.2 Geometry

Geometry may be used for NCS control alone or in combination with other control methods. Favorable geometry is based on limiting dimensions of defined geometrical shapes to established subcritical limits. Structure and/or neutron absorbers that are not removable constitute a form of geometry control. At the GLE Commercial Facility, favorable geometry is developed conservatively assuming full water or concrete equivalent reflection, optimal hydrogenous moderation, worst credible heterogeneity, and maximum credible enrichment. Examples of parameters used for engineered geometry controls include cylinder diameters, annulus inner and outer radii, slab thickness, and/or fixed volumes.

Subcritical limits for geometry controls may be derived using either validated analytical methods and an approved MMS MoS or experimental data. Where experimental data are used, the margins of safety are 90 percent of the minimum critical cylinder diameter, 85 percent of the minimum critical slab thickness, and 75 percent of the minimum critical sphere volume.

Geometry control systems are analyzed and evaluated allowing for fabrication tolerances and dimensional changes that may likely occur through corrosion, wear, or mechanical distortion. Before beginning operations, dimensions and nuclear properties applicable to the geometry control are verified using appropriate instrumentation. The CM Program is used to maintain these dimensions and nuclear properties within acceptable limits. Provisions are also made for periodic inspection, if credible conditions exist in which changes in the dimensions or nuclear properties of the equipment could occur, resulting in the inability to meet established NCS limits.

5.4.4.3 Enrichment

Enrichment control may be utilized to limit the weight percent ^{235}U within a process, vessel, or container, thus providing a method for NCS control. Enrichment controls may be used to segregate materials of different enrichment or to prevent material from being enriched above an NCS limit. Where enrichment is controlled, active engineered or administrative controls are required to measure or verify the enrichment, or to prevent the introduction of uranium at unacceptable enrichment levels within a defined subsystem. In cases where enrichment control is not utilized, the maximum credible enrichment for the particular process or subsystem is utilized in the CSA.

5.4.4.4 Reflection

Most systems are designed and operated with the assumption of 12-inch water or optimum reflection surrounding the system. In such cases, controls limiting reflection are not required since optimum reflection has been demonstrated safe. However, subject to approved controls limiting reflection, certain system designs may be analyzed, approved, and operated in situations where the analyzed reflection is less than optimum. In the CSA, the neutron reflection properties of the credible process environment are also considered. For example, reflectors more effective than water (such as, concrete) and adjacent structural materials are considered when appropriate.

5.4.4.5 Moderation

Moderation control may be used for NCS control alone or in combination with other control methods. Moderation controls are used to limit the amount of moderation present within fissile material. Where moderation is used as an NCS controlled parameter, moderation controls are implemented consistent with the guidance in ANSI/ANS 8.22-1997, *Nuclear Criticality Safety Based on Limiting and Controlling Moderators (Ref. 5-19)*. When moderation control is used, the area is posted as a Moderation Controlled Area (MCA) and specific moderation controls are delineated. Operations in MCAs must be demonstrated safe under normal and abnormal conditions such that the double contingency principle is satisfied.

In evaluating systems where a controlled parameter is moderation, the following requirements apply:

- Identify credible sources of moderation intrusion and control the ingress of moderation in accordance with the double contingency principle;
- Design physical structures, barriers, and/or equipment involved in the system to limit or control the ingress of moderation;
- Use qualified instrumentation where moderation control requires the moderation content or other system parameters to be measured or monitored;
- Use redundant independent sampling methods where moderation control is the only controlled parameter; and
- Control combustible materials, document fire-fighting methods in approved written procedures, and provide for approved sprinkler systems, manual means, or non-hydrogenous chemicals for fire fighting as specified by the process analysis.

Where moderation control is the only controlled parameter, the minimum protection is never less than two independent controls on moderation for each credible accident sequence, which must fail before a criticality accident is possible. Additional defense in depth protection may also be specified in process evaluations. The basis for selection of moderation controls shall be documented in CSAs and evaluated in accordance with the ISA Process described in GLE LA Chapter 3. The introduction and use of moderating materials (such as, cleaning agents, oils, or lubricants) within designated MCAs are subject to controls/limits that are approved by the NCS function.

5.4.4.6 *Concentration (or Density)*

Concentration control may be used for NCS control alone or in combination with other control methods. Concentration controls are established to ensure the concentration level is maintained within defined limits for the system. Each process relying on concentration control has engineered controls in place to detect and/or mitigate the effects of high concentration within the system; otherwise, the most reactive credible concentration (density) is assumed.

Concentration control is typically used in processes containing solution with low uranium concentrations such as a liquid effluent system. In evaluating systems containing concentration-controlled solution, the following requirements apply:

- Preclude a high concentration of uranium in a process unless the process is demonstrated safe at any credible concentration (for example, a favorable geometry tank);
- Equip the tank/vessel with backflow prevention controls (for example, air break, siphon breaks, overflow lines) where appropriate and inspect periodically for buildup; and
- Take precautions where precipitating agents are added to ensure agents are not inadvertently introduced.

When concentration is the only parameter controlled to prevent criticality, concentration may be controlled by two independent combinations of measurement and physical control, with each physical control capable of preventing the concentration limit from being exceeded in an unsafe location. The preferred method of attaining independence is to ensure that at least one of the two combinations is an active engineered control.

5.4.4.7 *Interaction (or Unit Spacing)*

Interaction/spacing control may be used for NCS control alone or in combination with other control methods. Interaction controls are based on either neutronic isolation or spacing of interacting units to control neutron leakage. Physical separation between process operations, vessels, or containers may be provided by either engineered or augmented administrative controls depending on the application. Where engineered spacing controls are required the structural integrity of the engineered feature must be sufficient for normal and credible abnormal conditions.

Units may be considered effectively non-interacting (isolated) if they are: (1) separated by 12-inches of full density water equivalent; (2) separated by the larger of 12-foot air distance or the greatest distance across an orthographic projection of the largest fissile accumulation on a plane perpendicular to the line joining their centers; or (3) shown to be non-interacting based on comparison of the calculated effective multiplication factor for the unit and that of the entire system.

5.4.4.8 *Neutron Absorbers*

Neutron absorbing materials may be utilized to provide a method for NCS control for a process, vessel, or container. Stable compounds such as boron carbide fixed in a matrix (such as, aluminum or polyester resin, elemental cadmium clad in appropriate material, elemental boron alloyed stainless steel, or other solid neutron absorbing materials) with an established dimensional relationship to the fissionable material are recommended. The use of neutron absorbers in this manner is defined as part of a passive engineered control. When evaluating the absorber effectiveness for an application, the neutron spectrum is considered in the CSA.

Where neutron absorbers are used as an NCS controlled parameter, fixed neutron absorbers controls are implemented consistent with the guidance in ANSI/ANS 8.21-1995, *Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors (Ref. 5-20).*

Only fixed absorbers may be used as NCS controls on neutron absorption. Soluble neutron absorbers (for example, boric acid) and removable neutron absorbers (for example, Raschig Rings) are not used as NCS controls.

5.4.4.9 Process Characteristics

Within certain fissile material operations, credit may be taken for physical, chemical, and nuclear properties of the process and/or materials as NCS controls. Use of process characteristics is based upon the following requirements:

- Identify the bounding conditions and operational limits in the CSA and communicate, through training and procedures, to appropriate Operations personnel.
- Base bounding conditions for such process and/or material characteristics on established physical, chemical, or nuclear reactions, known scientific principles, and/or facility-specific experimental data supported by operational history.
- The devices and/or procedures, which maintain the limiting conditions, must have the reliability, independence, and other characteristics required of a criticality safety control.

5.4.5 Criticality Safety Analyses

The scope and content of any particular CSA reflects the needs and characteristics of the system being analyzed and typically includes the applicable information requirements listed below.

<u>Scope</u> – Defines the stated purpose of the analysis.

<u>General Discussion</u> – Presents an overview of the process affected by the proposed change. This section includes, as appropriate: process description, flow diagrams, normal operating conditions, system interfaces, and other important to design considerations.

<u>Criticality Safety Controls/Bounding Assumptions</u> – Defines the controlled parameter(s) and summarizes the criticality safety controls on each identified parameter that are imposed as a result of the evaluation. This section also clearly presents a summary of the bounding assumptions used in the analysis. Bounding assumptions include: worst credible contents (for example, material composition, density, enrichment, and moderation), boundary conditions, inter-unit water, and a statement on assumed structure. In addition, this section may include a statement summarizing interface considerations with other units, subareas, and/or areas.

Model Description – Presents a narrative description of the actual model used in the analysis. An identification of both normal and credible upset (accident) conditions and model file naming convention is provided. Key input listings and corresponding geometry plot(s) for both normal and credible upset cases are also provided.

Calculational Results – Identifies how the calculations were performed, what tools or reference documents were used, and when appropriate, presents a tabular listing of the calculational result and associated uncertainty (for example, $K_{eff} + 3\sigma$) results as a function of the key parameter(s) (for example, wt. fraction H_2O). When applicable, the assigned bias of the calculation is also clearly stated and incorporated into both normal and/or accident limit comparisons.

Safety During Upset Conditions – Presents a concise summary of the upset conditions considered credible for the defined unit or process system. This section includes a discussion as to how established NCS limits and controls address each credible process upset (accident) condition to maintain subcriticality.

Specifications and Requirements for Safety – When applicable, presents both design specifications and criticality safety requirements for correct implementation of established controls. These requirements are incorporated into operating procedures, training, maintenance, and QA as appropriate to implement the specifications and requirements.

Compliance – Concludes the analysis with pertinent summary statements and includes a statement regarding license compliance.

Verification – A qualified Senior NCS Engineer, who was not involved in the analysis, verifies each CSA in accordance with GLE LA Section 5.4.5.1, *Technical Reviews.*

Appendices – Where necessary, include a summary of information ancillary to calculations such as parametric sensitivity studies, references, key inputs, model geometry plots, equipment sketches, useful data, etc., for each defined system.

5.4.5.1 Technical Reviews

Independent technical reviews of proposed criticality safety control limits specified in the CSA are performed. A Senior NCS Engineer is required to perform the independent technical review. The independent technical review consists of a verification that the neutronics geometry model and configuration used adequately represent the system being analyzed. In addition, the reviewer verifies that the proposed material characterizations such as density, concentration, etc., adequately represent the system. The reviewer also verifies that the proposed criticality safety controls are adequate. The independent technical review of the specific calculations and computer models is performed using one of the following methods:

- Verify the calculations with an alternate computational method;
- Verify methods with an independent analytic approach based on fundamental laws of nuclear physics;
- Verify the calculations by performing a comparison to results from a similar design or to similar previously performed calculations; or

- Verify the calculations by performing specific checks of the computer codes used, and by performing evaluations of code input and output.

Based on one of these prescribed methods, the independent technical review provides a reasonable measure of assurance that the chosen analysis methodology and results are correct.

5.5 REPORTING REQUIREMENTS

A program for evaluating the criticality significance of NCS events is established for making the required notification to the NRC Operations Center. Qualified individuals make the determination of the significance of NCS events. The determination of loss or degradation of double contingency protection is made against the documented CSA, the License, and 10 CFR 70, Appendix A. GLE commits to the following NCS reporting requirements:

- The reporting criteria of 10 CFR 70, Appendix A and the report content requirements of 10 CFR 70.50, *Reporting Requirements (Ref. 5-21)*, are incorporated into approved written procedures.
- If it cannot be ascertained within one hour of the discovery of an event, whether the criteria of 10 CFR 70, Appendix A, Paragraph (a) applies, the event should be treated as a one-hour reportable event.
- If it cannot be ascertained within 24 hours of discovery of an event, whether the criteria of 10 CFR 70, Appendix A, Paragraph (b) applies, the event should be treated as a 24-hour reportable event.
- The required report is issued when the IROFS credited is lost, irrespective of whether the safety limits of the associated parameters are actually exceeded.

5.6 REFERENCES

5-1. 10 CFR 70.64, *Requirements for New Facilities or New Processes at Existing Facilities*, U.S. Nuclear Regulatory Commission, 2008.

5-2. ANSI/ANS 8.1-1998 (R2007), *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, American Nuclear Society, January 1998.

5-3. 10 CFR 70.61, *Performance Requirements,* U.S. Nuclear Regulatory Commission, 2008.

5-4. 10 CFR 70.62, *Safety Program and Integrated Safety Analysis,* U.S. Nuclear Regulatory Commission, 2008.

5-5. 10 CFR 70, *Domestic Licensing of Special Nuclear Material*, U.S. Nuclear Regulatory Commission, 2008.

5-6. 10 CFR 70.72, *Facility Changes and Change Process*, U.S. Nuclear Regulatory Commission, 2008.

5-7. 10 CFR 70.65, *Additional Content of Applications,* U.S. Nuclear Regulatory Commission, 2008.

5-8. ANSI/ANS 8.19-2005, *Administrative Practice for Nuclear Criticality Safety*, American Nuclear Society, January 2005.

5-9. ANSI/ANS 8.26-2007, *Criticality Safety Engineer Training and Qualification Program*, American Nuclear Society, June 2007.

5-10. 10 CFR 70.24, *Criticality Accident Requirements*, U.S. Nuclear Regulatory Commission, 2008.

5-11. ANSI/ANS 8.3-1997 (R2003), *Criticality Accident Alarm System*, American Nuclear Society, January 1997.

5-12. Regulatory Guide 3.71, *Nuclear Criticality Safety Standards for Fuels and Material Facilities*, U.S. Nuclear Regulatory Commission, Revision 1, October 2005.

5-13. ANSI/ANS 8.23-1997, *Nuclear Criticality Accident Emergency Planning and Response*, American Nuclear Society, January 1997.

5-14. 10 CFR 70.17, *Specific Exemptions*, U.S. Nuclear Regulatory Commission, 2008.

5-15. ANSI N14.1-2001, *Nuclear Materials – Uranium Hexafluoride – Packaging for Transport,* American National Standards Institute, January 2001.

5-16. POEF-2086, ORNL/TM-11988, *Investigation of Breached Depleted UF_6 Cylinders*, Barber, E.J., et. al., September 1991.

5-17. ARH-600, *Criticality Handbook*, R. D. Carter, G. R. Kiel, and K. R. Ridgway, Atlantic Richfield Hanford Co. Report, 1968.

5-18. ANSI/ANS 8.24-2007, *Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations*, American Nuclear Society, 2007.

5-19. ANSI/ANS 8.22-1997 (R2006), *Nuclear Criticality Safety Based on Limiting and Controlling Moderators*, American Nuclear Society, January 1997.

5-20. ANSI/ANS 8.21-1995 (R2001), *Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors*, American Nuclear Society, January 1995.

5-21. 10 CFR 70.50, *Reporting Requirements*, U.S. Nuclear Regulatory Commission, 2008.

CHAPTER 8

REVISION LOG

Rev.	Effective Date	Affected Pages	Revision Description
0	04/30/2009	ALL	Initial Application Submittal.
1	03/10/2011	8-4	Revised to include commitment to 10 CFR 70.32 for changes to the RC&EP.

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8. EMERGENCY RESPONSE

Plans for handling emergencies at the GE-Hitachi Global Laser Enrichment LLC (GLE) Commercial Facility are presented in the Radiological Contingency and Emergency Plan (RC&EP). The RC&EP has been developed for the entire Wilmington Site and includes the GLE Commercial Facility and the Global Nuclear Fuel – Americas, LLC (GNF-A) Fuel Manufacturing Facility.

The RC&EP was developed in accordance with 10 CFR 70.22(i)(3), *Contents of Applications (Ref. 8-1)* and 10 CFR 40.31(j), *Applications for Specific Licenses (Ref. 8-2)*. The RC&EP is consistent with the guidance presented in Regulatory Guide 3.67, *Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities (Ref. 8-3)*. The RC&EP also addresses the specific acceptance criteria in NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility (Ref. 8-4)*, Chapter 8, *Emergency Management*. The RC&EP is maintained under configuration control, and changes to the RC&EP are evaluated to determine if there is a reduction in effectiveness such that prior NRC approval is required in accordance with the regulations in 70.32(i).

GLE maintains Memorandums of Understanding (MOUs) with offsite support organizations identified in the RC&EP. These organizations, in addition to the State of North Carolina Division of Emergency Management and the State of North Carolina Division of Environment and Natural Resources Radioactive Materials Section, reviewed the RC&EP pursuant to the requirement in 10 CFR 70.22(i)(4) and 10 CFR 40.31(j)(4). Review comments from these organizations were included with the RC&EP submittal to the NRC.

8.1 REFERENCES

8-1. 10 CFR 70.22, *Contents of Applications*, U.S. Nuclear Regulatory Commission, 2008.

8-2. 10 CFR 40.31, *Application for Specific Licenses,* U.S. Nuclear Regulatory Commission, 2008.

8-3. Regulatory Guide 3.67, *Standard Format and Content of Emergency Plans for Fuel Cycle and Materials Facilities*, U.S. Nuclear Regulatory Commission, Revision 0, January 1992.

8-4. NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility*, U.S. Nuclear Regulatory Commission, March 2002.

CHAPTER 11
REVISION LOG

Rev.	Effective Date	Affected Pages	Revision Description
0	04/30/2009	ALL	Initial Application Submittal.
1	03/31/2010	6, 8, 11, 12, 19, 21, 33, 38-40, 42, 46, 50, 51	Incorporate RAI responses submitted to the NRC via MFN-09-578 dated 09/04/2009; MFN-09-801 dated 12/28/2009; and MFN-10-056 dated 02/10/2010.
2	06/18/2010	39, 40	Revised Section 11.8.2.1 to clarify the graded approach to applying management measures.
3	12/17/2010	4, 15, 38	Incorporate RAI responses from NRC letters dated October 5, 2010, November 12. 2010 and November 19, 2010 Revised Section 11.1.1 to clarify applicability of the CM program Revised Section 11.3.2.3 to add laser safety training Revised Section 11.8 to remove duplicate text and refer to the NRC-approved GLE Quality Assurance Plan Description
4	3/30/2011	4, 5, 6, 7, 8	Added language to clarify the application of the graded approach to applying management measures to IROFS. Replaced reference to deleted section of this chapter with GLE QAPD. Revised Section 11.1.4 to indicate that procedures control changes during design, construction and operations. Replaced the term "as built" with "as constructed".

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11. MANAGEMENT MEASURES

This chapter describes the management measures established by GE-Hitachi Global Laser Enrichment LLC (GLE) that are applied to Items Relied on for Safety (IROFS). GLE commits to apply management measures to IROFS on a continuing basis to provide reasonable assurance that IROFS are available and able to perform their intended functions when needed. Implementation of the management measures ensures the GLE Commercial Facility can be operated safely, and provides adequate protection of the workers, the public, and the environment from credible hazards presented in the Integrated Safety Analysis (ISA).

The GLE management measures provide oversight and assurance that the GLE Safety Program is maintained and functions properly. GLE applies management measures in a graded approach based on unmitigated risks as described in the ISA Summary. According to criteria defined in approved written procedures, the relative importance of an IROFS is determined using both the severity of consequence and unmitigated likelihood of an initiating event. Based on the assigned importance, the appropriate type and number of management measures are assigned to assure the IROFS are functional when needed.

The extent that attributes of management measures and QA program elements are applied to IROFS will be determined by evaluating the factors that contribute to reliability of each IROFS. The management measure and QA element attributes for those aspects of the activity that influence reliability of the IROFS will be determined by evaluating the design, function, and task analyses associated with operating and maintaining the IROFS and by assigning the characteristic to the attribute taking into consideration the following:

- Risk significance,
- Applicable regulations, industry codes, and standards,
- Complexity or uniqueness of an item/activity and the environment in which it has to function,
- Quality history of the item in service or activity,
- Degree to which functional compliance can be demonstrated or assessed by test, inspection, or maintenance methods,
- Anticipated life span,
- Degree of standardization,
- Importance of data generated, and
- Reproducibility of results.

The management measure and QA elements attributes assigned to each IROFS will be approved through the configuration management process associated with ISA Baseline Documents and specifically through approval of the IROFS Boundary Definition Packages as the design matures, procedures and training are developed, and pre-operational readiness reviews are conducted.

11.1 CONFIGURATION MANAGEMENT

The objective of the Configuration Management (CM) Program is to ensure the information used to design, construct, operate, and maintain IROFS is current. Safety controls (IROFS) are structures, systems, and components (SSCs) and procedures that prevent or mitigate the risk of credible accidents. The elements of the CM Program provide consistency among the GLE Commercial Facility design and operations, physical configuration, and documentation.

11.1.1 Configuration Management Policy

GLE commits to maintain a formal CM Program in accordance with 10 CFR 70.72, *Facility Changes and Change Process (Ref. 11-1)*. The CM process is implemented by approved written procedures so that each change to the GLE Commercial Facility (the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel) is evaluated, implemented, and tracked. Prior to implementing a change to the GLE Commercial Facility it must be evaluated to determine if an amendment to the license is required to be submitted and approved by the NRC before being implemented. The CM Program includes the following activities:

- Maintenance of facility design information,
- Identification of IROFS,
- Control of information used to operate and maintain the facility,
- Documentation of changes,
- Assurance of adequate safety reviews for changes, and
- Periodic performance assessment of specific safety controls to ensure conformance to design basis documentation.

The level of CM applied to the SSCs, processes, equipment, software, and personnel activities is based on the associated quality level (QL) designation. QLs are defined in the GLE Quality Assurance Program Description, NEDE-33451, Section 3.1, *Quality Levels*.

The CM Program is managed by the CM Manager. During design and construction, the CM Manager reports to the Engineering Manager. During the operational phase, the CM Manager reports to the Operations Manager. See GLE LA Chapter 2, *Organization and Administration*, for additional information on the GLE organization.

During the design phase, CM is based on the design control, and associated procedural controls, to establish and maintain the Technical Design Baseline. Design documents, including the ISA, provide design input, analysis, and/or results specifically for IROFS. Design documents undergo interdisciplinary review prior to initial issue and during each subsequent revision. During the construction phase of the project, changes to drawings and specifications issued for construction, procurement, or fabrication are systematically reviewed, verified, evaluated for impact (including impact to the ISA), and approved prior to implementation. Proper implementation is verified and reflected in the design basis documentation.

In order to provide continued safe and reliable operation of GLE Commercial Facility SSCs, controls are implemented to ensure the quality of the SSCs is not compromised by planned changes (modifications). The following items are addressed prior to implementing a facility change:

- Technical basis for the change,
- Impact on safety, health, and control of licensed material,
- Required modifications to existing procedures, to include any necessary training prior to operation,
- Authorization requirements for the change,
- For temporary changes, the approved duration (expiration date) of the change, and
- Impacts or modifications to the ISA, ISA Summary, and any other component of the overall safety program.

11.1.2 Design Requirements

Procedures define the development, application, and maintenance of the design specifications and requirements. Design requirements are developed to support safety functions, environmental impact-oriented functions, and mission-based functions. IROFS identified in the ISA Summary and design documents are identified in more detail during the final design. Design requirements for IROFS and other SSCs are developed with the baseline design criteria defined in 10 CFR 70.64, *Requirements for New Facilities or New Processes at Existing Facilities (Ref. 11-2)*. The design requirements to support the IROFS and other SSCs are developed by the Engineering Organization and documented in design documents. Prior to approval, the design documents are reviewed to determine adequacy, accuracy, and completeness. After approval, the design documents and the ISA Summary provide the Technical Design Baseline for the facility. Design documents and the ISA are controlled documents. Changes to design documents or the ISA are subject to the Change Control Process. See GLE the GLE Quality Assurance Program Description, NEDE-33451, for additional information on the Design Control Process.

11.1.3 Document Control

Document Control, as defined in approved written procedures, includes creation, revision, storage, tracking, distribution, and retrieval of applicable information, to include, but not limited to, manuals, instructions, drawings, procedures, design documents, specifications, plans, and other documents that pertain to the CM function. Procedures are established to control the life-cycle of documents. Appropriate measures have been established to ensure documents are adequately reviewed, approved, and released for use by authorized personnel.

Document control is implemented in accordance with approved written procedures. An electronic document management system (EDMS) is used to file project records and to make available the latest revision (that is, the controlled copy) of controlled documents. Indices of controlled documents, which are uniquely numbered (including revision number), are maintained and available to affected personnel. Controlled documents are maintained in the EDMS until cancelled or superseded. A cancelled or superseded controlled document continues to be maintained as a record. Hardcopy distribution of controlled documents is provided when needed in accordance with approved written procedures (for example, when the EDMS is not available or the complexity of a task requires that the procedure be in-hand).

11.1.4 Change Control

GLE maintains approved written procedures describing the CM process for controlling design/construction/operation changes, including approval to install or make modifications to facilities, processes, procedures, or equipment. Per approved written procedures, a trained safety reviewer is required to review and approve changes to controlled documents to determine if the ISA is impacted by the proposed change. If there is an impact to the ISA, the change is flagged for review and approval by an ISA Team in accordance with the process described in the ISA Summary. Approved written procedures also detail the controls and define the distinction between types of changes, ranging from an equipment replacement with an identical design authorized as part of normal maintenance, to new or different facility designs which require specified review and approval.

During the design phase the method of ensuring consistency between documents, including consistency between design changes and the ISA, is the interdisciplinary review process. When the project enters the construction phase, changes to documents issued for construction, fabrication, and procurement are documented, reviewed, approved, and posted against each affected design document. Vendor drawings and data also undergo an interdisciplinary review to ensure compliance with procurement specifications and drawings, and to incorporate interface requirements into controlled documents.

During the operations phase, changes to design are documented, reviewed, and approved prior to implementation. GLE's change process fully implements the provisions of 10 CFR 70.72. Measures are provided to ensure responsible facility personnel are made aware of design changes and modifications that may affect the performance of their duties. After completion of a modification to a SSC, the appropriate area manager, or designee, shall ensure that applicable testing has been completed to ensure correct operation of the system(s) affected by the modification and documentation regarding the modification is complete. In order to ensure operators are able to operate a modified system safely, when a modification is complete, necessary documents (such as, the revised process description, checklists for operation and flow sheets) are made available to the Operations and Maintenance Organizations once the modified system becomes "operational." Appropriate training on the modification is completed prior to the system being placed in operation. A formal notice of a modification being completed is distributed to appropriate managers. As-constructed drawings incorporating the modification are completed promptly. These records shall be identifiable and retained for the duration of the facility license.

11.1.5 Assessments

Planned internal and independent assessments are performed to evaluate the application and effectiveness of management measures and implementation of programs related to facility safety. Periodic assessments of the CM Program are conducted to determine the program's effectiveness and correct any identified deficiencies. These assessments include review of documentation and system walk downs of the as-constructed facility. CM assessments are performed, at a minimum, on an annual basis. Individuals not involved in the area being assessed will conduct independent assessments.

11.2 MAINTENANCE

The purpose of planned and scheduled maintenance of IROFS is to assure systems are kept in a condition of readiness to perform designed functions when required. Area managers are responsible for assuring the operational readiness of safety controls in assigned areas of the GLE Commercial Facility.

The Maintenance function utilizes a systems-based program to plan, schedule, track, and maintain records for maintenance activities. Maintenance procedures and instructions are an integral part of the Maintenance Program. Maintenance procedures are described in GLE LA Section 11.4, *Procedures*. Key maintenance requirements for safety controls, such as calibration, functional testing, and replacement of specified components, are derived from the analyses described in the ISA Summary.

The selection and qualification of Maintenance personnel is documented and implemented through approved written procedures. Contractors working on or performing activities that could affect IROFS are required to follow the same procedures as Maintenance personnel. Maintenance activities generally fall into one of the four (4) categories described below.

11.2.1 Corrective Maintenance

Corrective maintenance refers to situations where repairs, replacements, or major adjustments such as recalibration occur. GLE commits to promptly perform corrective actions to remediate unacceptable performance deficiencies in IROFS. The Maintenance Planning and Control System provides documentation and records of SSCs that have been repaired or replaced. When a component of a specified safety control is repaired or replaced, the component is functionally verified via post-maintenance testing to ensure it has the capability to perform the planned and designed function when called upon to do so. If the performance of a repaired or replaced safety control could be different from that of the original component, the change to the safety control is specifically approved under the CM Program and pre-operationally tested to ensure it will perform its desired function when called upon to do so.

11.2.2 Preventive Maintenance

Preventive maintenance (PM) is performed on a periodic basis to prevent failures, facilitate performance, and maintain or extend the life of equipment. PMs help ensure IROFS are available and reliable. The bases for PM tasks are developed through a review of manufacturer recommendations, available industry standards, and historical operating information, where available. PMs are included in the work control process to facilitate planning, scheduling, and execution of these tasks.

Establishment of a PM task is coordinated by the Maintenance Organization and requires input from various disciplines within the Engineering and Operations Organizations. The formal documented bases for the tasks are developed, evaluated, and approved by the Engineering Organization. PM tasks may be changed, new tasks added or deleted, and recommendations made by Operations, Maintenance, or Engineering personnel. Feedback from PM, corrective maintenance, and incident investigations is used, as appropriate, to modify the frequency or scope of a PM activity. Specifically, preventive measures to alleviate premature failure may be added to the PM activity, or a reduction in frequency of a particular PM due to as-found conditions indicating that the PM is occurring more often than necessary.

After conducting PM on IROFS, and prior to returning an IROFS to operational status, functional testing of the SSC, if necessary, is performed to ensure the IROFS performs its intended safety function. Records pertaining to PM are maintained in accordance with the Records Management (RM) System.

11.2.3 Surveillance and Monitoring

The ISA Summary identifies the IROFS that are to be available and reliable to perform their design function for the prevention or mitigation of credible events. The Surveillance and Monitoring Program provides a periodic check of the ability of IROFS to perform their design safety function when called upon to do so. Surveillances are in the form of performance checks, calibrations, tests, and inspections.

GLE utilizes active engineered controls that are integrated into routine operations to the degree practical. The IROFS are monitored as a routine part of the operating process. IROFS associated with passive engineered systems are typically fixed physical design features to maintain safe process conditions. Availability and reliability of IROFS is maintained through preoperational audits and periodic verifications as prescribed in the ISA, and includes consideration of the importance of the IROFS as well as available quality and reliability information. IROFS relying on geometry-based controls, where the geometry is subject to undetected change in routine operation, are periodically verified on a schedule commensurate with the potential for change in the parameters of interest.

Surveillances are included in the work control process to permit timely planning, scheduling, establishment of system or facility conditions, execution of the activity, and creation of documentation that identifies the results of the surveillance. The established frequencies are determined by the IROFS degree of safety importance. The results of surveillance activities are trended to support the determination of performance trends for IROFS. When potential performance degradation is identified, PM frequencies are adjusted or other corrective actions are taken as appropriate.

Incident investigations may identify the root cause of a failure that is related to the type or frequency of maintenance. The lessons learned from such investigations are factored into the Surveillance and Monitoring Program and the PM Program, as appropriate. Maintenance procedures prescribe compensatory measures, if appropriate, for surveillance tests of IROFS that can only be performed while equipment is out of service.

11.2.4 Functional Testing

Functional testing of IROFS is performed as appropriate, following initial installation as part of periodic surveillance testing and after corrective maintenance, PM, or calibration to ensure that the item is capable of performing the designed safety function when required. GLE commits to perform functional tests in accordance with approved written procedures that define the method for the test and the required acceptable results. The results of the tests are recorded and maintained.

Administrative controls that are identified as IROFS are documented in approved written procedures. Administrative controls are assured to be available and reliable during operations by applying the applicable management measures described in this LA Chapter, including the use of procedures and the employee training programs. See GLE LA Section 11.3, *Training and Qualifications*, and Section 11.4 for additional information on how these management measures are applied to administrative controls.

11.2.4.1 Preoperational Testing

Preoperational testing at the facility consists of testing conducted to initially determine various facility parameters and to initially verify the capability of SSCs to meet performance requirements. The major objective of preoperational testing is to verify that IROFS, essential to the safe operation of the facility, are capable of performing their intended function. Initial startup testing is performed beginning with the introduction of uranium hexafluoride (UF_6) and ending with the startup. The purpose of initial startup testing is to ensure safe and orderly UF_6 feeding, and to verify parameters assumed in the ISA. Records of the preoperational and startup tests required prior to operation are maintained. These records include testing schedules and results for IROFS.

11.2.4.2 Post-Maintenance Testing

Post-maintenance testing (PMT) is established to provide assurance that IROFS will perform their intended function following maintenance activities. This test confirms the maintenance performed was satisfactory, the identified deficiency has been corrected, and the maintenance activity did not adversely affect the reliability of the item. This test is performed, with acceptable results, prior to returning the equipment to service.

PMT requirements are developed and included in work packages during the work planning process. The Engineering Organization may provide support to the Operations and Maintenance Organizations in identifying PMT requirements. The PMT meets applicable codes and technical requirements and specifies acceptance criteria. The results of the PMT are documented and retained in the work package with other documentation generated during the maintenance evolution.

11.2.5 Calibration

To assure that IROFS are available and reliable to perform their design function, those components that require calibration to provide a measurement used for safety-related purposed will be calibrated according to approved procedures developed utilizing manufacturer's recommended procedures or, lacking such guidance, procedures developed by knowledgeable professionals following applicable codes and standards. The calibration processes utilizes calibration standards traceable to the National Institute of Standards and Technology (NIST). If no nationally recognized standard exists, the basis for calibration is documented. Calibration setpoints for devices performing safety functions are developed to assure that the device provides the necessary activation of the safety function consistent with the parameter limit and time requirements for initiation of the action. The parameter and activation time limits are established during development of the IROFS description in the Quantitative Risk Analysis (QRA) and are often based on calculation limits provided in the Criticality Safety Analysis (which are generally absolute outside bounds on the parameter) or on other consensus standards (for example, AEGL exposure limits). Given the parameter limit, the activation time requirements, and the context for which the parameter is utilized, the device setpoints are developed using methodology found in appropriate standards (for example, ANSI/ANS 67.04.01-2000, *Setpoints for Nuclear Safety-Related Instrumentation)*, and implemented through approved engineering procedures.

Procedures for the setpoint determination address determination of the calibration ranges of test devices, measuring and test instrumentation for use in the calibration, calibration standard requirements, and the acceptable response of the devices in response to the calibration standard. The functional tests that provide checks on the instruments are provided acceptable tolerance ranges for satisfactory operation. Devices that fail to satisfy the function test tolerances are recalibrated. Setpoint calculations and functional test tolerances are documented in design calculations that are referenced in the IROFS Boundary Definition Packages and available as the basis for development of calibration and functional testing procedures development and training. Calibration and function testing procedures require the documentation of the as-found and as-left condition or the trip point of the device to allow evaluation of the instrument drift characteristics to be used for evaluating/modifying the calibration periodicity or setpoint requirements based on historical device performance.

11.3 TRAINING AND QUALIFICATIONS

The Training Program is designed to ensure personnel who perform activities relied on for safety have the applicable knowledge and skills necessary to design, operate, and maintain the GLE Commercial Facility in a safe manner. Performance-based training is used for analyzing, designing, developing, conducting, and evaluating training. Personnel are trained and tested as necessary to ensure they are qualified on practices important to public and worker safety, safeguarding licensed material, and protection of the environment. Exceptions from training requirements may be granted when justified and documented in accordance with approved written procedures and approved by the appropriate level of management.

11.3.1 Organization and Management of the Training Function

Training Programs for personnel who perform activities relied on for safety, are provided through shared responsibility between the Environmental, Health, and Safety (EHS) disciplines and line management. Line managers are responsible for the content and effective conduct of training for assigned personnel. Training responsibilities for line managers are included in position descriptions, and line managers are given the authority to implement training for assigned personnel. The GLE Training function provides support to line management. Performance-based training is used as the primary management tool for analyzing, designing, developing, conducting, and evaluating training. Area managers are responsible for the content and effective conduct of training for Operations personnel.

Approved written procedures establish the requirements for indoctrination and training of personnel performing activities relied on for safety and ensures the Training Program is conducted in a reliable and consistent manner. Lesson plans or training guides are used for classroom and on-the-job training (OJT) to provide a consistent subject matter. When design changes or facility modifications are implemented, updates of applicable lesson plans are included in the change control process of the CM Program. Personnel may be exempt from training if an individual's prior training, qualifications, and job performance history provides information demonstrating that the individual has achieved the necessary required skills. Exemptions from training shall be documented and approved by management.

Training records are maintained to support management information needs associated with personnel training, job performance, and qualifications. Training records are retained in accordance with RM approved written procedures.

11.3.2 Types of Required Training

Training is provided for each individual at the GLE Commercial Facility, commensurate with assigned roles and responsibilities. Training and qualification requirements are met prior to personnel fully assuming the duties of safety-significant positions, and before assigned tasks are independently performed.

The objective of the Training Program is to ensure safe and efficient operation of the facility and ensure compliance with applicable regulatory requirements. Training requirements shall be applicable to, but not restricted to, those personnel who have a direct relationship to the operation, maintenance, testing, or other technical aspects of IROFS.

Contractor personnel shall meet the minimum training and qualification requirements. The line manager responsible for the contracted activity shall verify contractor training. The Radiological Contingency and Emergency Plan (RC&EP) provides additional information on personnel training for emergency response activities. Training courses are kept up-to-date to reflect facility modifications and changes to procedures when applicable.

- Required training may be grouped into one of five categories:
- General Employee Training (GET),
- Nuclear Safety Training,
- Industrial Safety Training,
- Technical Training, and
- Professional Development.

These categories of training are discussed in the following sections. Specific training requirements associated with the Emergency Response Organization (ERO) are addressed in the RC&EP.

11.3.2.1 General Employee Training

GET encompasses those Quality Assurance (QA), Radiation Protection (RP), Industrial Safety, Environmental Protection, Security and Emergency Response, and administrative procedures established by management and in accordance with applicable regulations. The Industrial Safety Training complies with 29 CFR 1910, *Occupational Safety and Health Standards (Ref. 11-3)*, and 10 CFR 19, *Notices, Instructions, and Reports to Workers: Inspection and Investigations (Ref. 11-4)*. Continuing training is conducted in these areas, as necessary, to maintain proficiency. All personnel (including contractors) must participate in GET. However, certain support personnel, depending on normal work assignment, may not participate in all topics of GET. Temporary maintenance and service personnel receive GET to the extent necessary to assure safe execution of assigned duties. Certain portions of GET may be included in New Employee Orientation. GET topics are listed below:

- General administrative controls and procedures and their use,
- QA policies and procedures,
- Nuclear safety (criticality and radiological),
- Industrial safety,
- RC&EP and implementing procedures associated with alarm response and evacuation,
- Fire protection and fire brigade,
- New employee orientation, and

- Environmental Protection.

11.3.2.2 Nuclear Safety Training

Training Programs are established for various job functions (for example, operations, RP technicians, contractor personnel) commensurate with criticality and RP responsibilities. Visitors to Radiological Controlled Areas (RCAs) are trained in the formal Training Program or are escorted by trained personnel.

Formal nuclear safety training includes information about radiation and radioactive materials, risks involved in receiving low-level radiation exposure in accordance with 10 CFR 19.12, *Instruction to Workers (Ref. 11-5)*, basic criteria and practices for RP, nuclear criticality safety (NCS) principles in conformance with applicable objectives contained in the American National Standards Institute (ANSI)/American Nuclear Society (ANS) 8.19-2005, *Administrative Practices for Nuclear Criticality Safety (Ref. 11-6)*, and ANSI/ANS 8.20-1991, *Nuclear Criticality Safety Training (Ref. 11-7)*.

The training policy requires employees to complete nuclear safety training prior to unescorted access in an RCA. Methods for evaluating the understanding and effectiveness of the training include passing an initial examination covering formal training contents and observations of operational activities during scheduled audits and inspections. Such training is typically computer based training, but may be performed by authorized instructors. The Training Program contents are reviewed on a scheduled basis by the NCS and RP functions to ensure the Training Program contents are current and adequate. Previously trained employees who are allowed unescorted access to an RCA are retrained annually at a minimum. The effectiveness of the Training Program is evaluated by either an initial training exam or a retraining exam. Visitors are trained commensurate with the scope of their visit and/or are escorted by trained employees.

11.3.2.3 Industrial Safety Training

Orientation of new or transferred employees to industrial safety is an important part of establishing the proper safety attitude among GLE employees, and insuring employees are aware of safety procedures, rules, and hazards involved in assigned duties. New employee orientation may include, as appropriate, the review of:

- Occupational Safety and Health Administration (OSHA) General Duty Clause,
- Employee/Employer Responsibilities,
- General Site Safety Rules,
- Hazard Communication Training,
- Laser Safety Training,
- Fire Extinguisher Training,
- Emergency Evacuation Procedure,
- Job Hazards Analysis (JHA) and Chemical Job Hazards Analysis (CJHA), and

- Lockout/Tagout Awareness.

11.3.2.4 Technical Training

Technical training is designed, developed, and implemented to assist Operations and Maintenance personnel gain an understanding of the applicable fundamentals, procedures, and technical practices common to a nuclear fuel enrichment facility. Technical training consists of initial training, OJT, continuing training, and special training, as applicable to specific assigned technical duties. This may include, but is not limited to: process specific training; mechanical maintenance; controls, instrumentation, electrical maintenance; and chemistry.

11.3.2.5 Professional Development

Professional development is a broad category implemented to assist GLE personnel in gaining additional understanding of fundamentals and technical practices common to their assigned job functions. Professional development typically utilizes internal or external professionals via formal workshop, tutorials, and select training programs.

11.3.3 Job-Specific Training Requirements

Operator training is performance-based and incorporates the structured elements of analysis, design, development, implementation, and evaluation commensurate with assigned duties. Minimum training requirements are developed for positions with activities that are relied on for safety. Initial identification of job-specific training requirements is based on individual employee experience. Entry-level criteria (such as, education, technical background, and experience) for these positions are contained in position descriptions. Job-specific training is performance-based and established with the relevant technical EHS safety discipline and Operations leadership to develop a list of qualifications for assigned duties. Changes to facilities, processes, equipment, or job duties are incorporated into revised lists of qualifications.

11.3.4 Basis of Training and Objectives

The Training Program is designed to prepare initial and replacement personnel for safe, reliable, and efficient operation of the GLE Commercial Facility. Emphasis is placed on safety requirements where human actions are important to safety.

Learning objectives are established to identify the training content and to define satisfactory trainee performance for the task, or a group of tasks, selected for training from the job analysis. Learning objectives state the requisite knowledge, skills, and abilities the trainee must demonstrate. The conditions under which the required actions take place and the standards of performance required of the trainee are also determined in development of the learning objectives. Learning objectives are sequenced within training materials based on the relationship to one another. Learning objectives are documented in lesson plans and training guides, and are revised as necessary, based on changes in procedures, facility SSCs, or job scope.

11.3.5 Organization of Instruction

Lesson plans are developed from learning objectives, which are based on job performance requirements. Lesson plans are reviewed by line management and by the responsible organization for the subject matter. Lesson plans are approved prior to issue or use.

11.3.6 Evaluation of Trainee Accomplishment

Trainee understanding and proficiency is evaluated through observation, demonstration, oral, or documented examinations, as appropriate. Such evaluations measure the trainee's skill and knowledge of job performance requirements. Evaluations are performed by individuals qualified in the training subject matter. Operator training and qualification requirements are met prior to process safety related tasks being independently performed or prior to startup, following significant changes to safety controls.

11.3.7 On-the-Job Training

OJT is a systematic method of providing the required job related skills and knowledge for a position. OJT is conducted in the work environment. Applicable tasks and related procedures make up the OJT Qualifications Program for each technical area which is designed to supplement and complement training received through formal classroom, laboratory, or simulator training. The objective of the program is to assure the trainee's ability to proficiently perform job duties as required for the assigned role. Completion of OJT is demonstrated through actual task actions using the conditions encountered during the performance of assigned duties including the use of references and tools, and equipment conditions reflecting the actual task to the extent practical.

11.3.8 Evaluation of Training Effectiveness

Periodic evaluations of Training Program content and requirements are performed to assess program effectiveness. The trainees provide feedback after completion of classroom or computer based training sessions to provide data for this evaluation. These evaluations identify program strengths and weaknesses, determine whether training content matches current job needs, and determines if corrective actions are needed to improve program effectiveness.

Independent audits of the EHS safety disciplines may also be used to provide independent evaluations of the overall Training Program effectiveness as it relates to the ISA, IROFS implementation, and protection of the public, worker, and environment. Evaluation objectives applicable to the overall organization and management of the Training Program may include, but are not limited to:

- Management and administration of training programs,
- Development and qualification of the matrix organization,
- Design and development of training programs, content, and conduct of training, and trainee examinations and evaluations,

- Training Program interface with the CM Program, and
- Training Program assessments and evaluations.

11.3.9 Personnel Qualification

The qualification requirements for key management positions are described in GLE LA Chapter 2. Qualification and training requirements for Operations personnel shall be established and implemented in accordance with approved written procedures.

11.3.10 Provisions for Continuing Assurance

Continuing or periodic retraining shall be established, when applicable, to ensure personnel remain proficient. Periodic training is generally conducted to ensure retention of knowledge and skills important to Operations. The training may consist of periodic retraining exercises, instructions, or review of subjects as appropriate to maintain the proficiency of personnel assigned to the facility. Retraining is required due to facility modifications, procedure changes, and QA Program changes resulting in new or changed information. The results of the retraining are documented.

11.4 PROCEDURES

GLE utilizes a hierarchy of policies, plans, and procedures to document management expectations and commitments, as well as to provide instructions and guidance to GLE personnel. Activities involving licensed special nuclear material (SNM) or IROFS are conducted in accordance with approved written procedures. Policies and plans are upper tier documents that define and describe senior management expectations and guidelines for safe operation of the GLE Commercial Facility and compliance with state and federal regulations, permits and licenses. Procedures are used to ensure implementation of the requirements set forth in policies and plans.

11.4.1 Types of Procedures

Procedures are categorized as management control procedures or operating procedures/instructions. Management control procedures describe administrative and general practices approved and issued by management at a level appropriate to the scope of the practice. These procedures direct and control activities across the various organizational functions, and assign functional responsibilities and requirements for these activities. Operating procedures provide specific direction for task-based work and are used to directly control process operations at the workstation.

Compliance with GLE procedures is mandatory. If any aspect of a procedure is unclear or incorrect as written, personnel shall safely stop the operation and/or activity and contact management. The operation and/or activity shall not restart until corrective action has been taken. If a situation is not defined in the procedure content or an unexpected response is obtained, management notification is also required. Deviations from operating procedures and unforeseen alternations in process conditions that affect nuclear criticality safety shall be reported to management, investigated promptly, corrected as appropriate, and documented.

11.4.1.1 Management Control Procedures

Management control procedures are used for activities that support the process operations. These procedures are used to manage activities such as design, CM, procurement, construction, RP, maintenance, QA, training and qualification, audits and assessments, incident investigations, RM, NCS, industrial safety, and reporting requirements.

11.4.1.2 Operating Procedures/Instructions

Operating procedures/instructions include direction for normal operations, off-normal operations, maintenance, alarm response, and emergency operations caused by failure of an IROFS or human error. These procedures provide reasonable assurance of RP, NCS, industrial safety, security and emergency preparedness, and environmental protection. Operating procedures/instructions contain the following elements, as applicable:

- Purpose,
- Regulations, policies, and guidelines governing the procedure,

- Type of procedure,
- Steps for each operating process phase,
- Initial startup,
- Normal operations,
- Temporary operations,
- Emergency operations and shutdown,
- Normal shutdown,
- Startup following an emergency or extended downtime,
- Hazards and safety considerations,
- Operating limits,
- Precautions necessary to prevent exposure to hazardous chemicals (resulting from operations with SNM) or to licensed SNM,
- Measures to be taken if contact or exposure occurs,
- IROFS associated with the process and associated functions, and
- The timeframe for which the procedure is valid.

Maintenance procedures involving IROFS for corrective and preventive maintenance, testing after maintenance, and surveillance maintenance activities describe the following, as needed:

- Qualifications of personnel authorized to perform the maintenance or surveillance,
- Controls on, and specification of, any replacement components or materials to be used,
- Post-maintenance testing to verify operability of the equipment,
- Tracking and RM of maintenance activities,
- Safe work practices (such as, lockout/tagout, confined space entry; moderation control or exclusion area requirements; radiation or hot work permits; and criticality, industrial, and environmental issues),
- Pre-maintenance activities require reviews of the work to be performed, including procedure reviews for accuracy and completeness, and

- Steps that require notification of affected parties (technicians and supervisors) before performing work and on completion of maintenance work. The discussion includes potential degradation of IROFS during the planned maintenance.

Alarm response procedures provide information that identifies the symptoms of the alarm, possible causes, automatic actions, the immediate operator action to be taken, and the required supplementary actions. Off-normal procedures describe actions to be taken during unusual or out of -the -ordinary situations. Emergency operating procedures direct actions necessary to mitigate potential events or events in progress that involve needed protection of onsite personnel; public health and safety; and the environment.

11.4.2 Procedure Development Process

11.4.2.1 Identification

Line managers, or designees, are responsible for the identification of procedures for assigned functional areas. Area managers are responsible for the identification of procedures incorporating control and limitation requirements established by the NCS, RP, Environmental Protection, and Industrial Safety functions. ISAs are used to identify procedures necessary for human actions important to safety. Approved written procedures have a unique identifier assigned by the Document Control function.

11.4.2.2 Development

Line managers, or designees, are responsible for procedure development. Procedure development is accomplished in accordance with approved written procedures. Procedures are initiated, developed, and controlled by the Document Control Program. Nuclear safety control requirements for workers are incorporated into the appropriate operating, maintenance, and test procedures for uranium enrichment operations.

Activities that require skills normally possessed by qualified personnel do not require detailed step-by-step delineation in a procedure. These activities are performed in accordance with documents of a type appropriate to the circumstances such as planning sheets, job descriptions, external manuals, or other applicable form.

11.4.2.3 Verification/Validation

Prior to initial use, procedures are verified and validated. Verification is a process that ensures the technical accuracy of the procedure. Validation verifies that the procedure can be performed as written. The document owner verifies the procedure during procedure development or during the change process. There are two basic attributes of the verification process. The first is the technical accuracy verification. This verification ensures technical information including formulas, set points, and acceptance criteria are correctly identified in the procedure. The second is administrative, in that it verifies the procedure format and style and verifies that the procedure meets the requirements in the approved written CM procedures.

The applicable guidance in NUREG-0700, *Human-System Interface Design Review Guidelines (Ref. 11-8)*, and NUREG-0711, *Human Factors Engineering Program Review Model (Ref. 11-9)*, is used to perform the procedural verification and validation.

The purpose of procedure validation is to ensure that no technical errors or human factor issues were inadvertently introduced during the procedure development or review process. Validation is required for new procedures and for procedure changes. Validation is performed in the field by qualified personnel, and may be accomplished by detailed scrutiny of the procedure as part of a walkthrough exercise or as part of a walkthrough drill (particularly for emergency or off-normal procedures). If the particular system or process is not available for a walkthrough validation, talk-through may be performed in the particular training environment. Performance of procedure validation is documented.

11.4.2.4 Review/Approval

Drafts of new procedures and procedure changes are distributed for technical reviews, safety discipline reviews (such as, NCS, Industrial Safety, and RP), and cross-discipline reviews, as needed. Comments/questions generated during the review process are resolved with the originating organizations. Following the resolution of review comments, procedures are approved. Approval authority rests with the applicable organization manager responsible for the activity. Managers have the responsibility to ensure that appropriate training is completed on new and revised procedures.

The QA function reviews QA implementing procedures for compliance and consistency with the QA Program and to ensure that the provisions of the QA Program are effectively incorporated into QA implementing procedures.

11.4.2.5 Issuance and Distribution

Controlled documents and approved revisions are distributed in a controlled manner in accordance with the Document Control Program. Line managers, or designees, shall be responsible for ensuring personnel doing work that requires the use of procedures have access to controlled copies of the required procedures.

11.4.3 Temporary Changes to Procedures

Temporary changes to procedures can be made, provided the change does not result in a change to the ISA as determined by the 10 CFR 70.72 review; and the change does not constitute an intent change (that is, a change in scope, method, or acceptance criteria that has safety significance). Temporary procedure changes must be documented per approved written procedures. Temporary procedure changes may be used for an identified period of time, which should not exceed 30 days or a period for which the temporary condition exists, whichever is greater. Temporary changes needing to exceed this period are assessed to ensure it is appropriate to extend the use of the temporary change or if a permanent change should be processed. Temporary changes may be made permanent once the change is reviewed and approved per the requirements of Section 11.4.2, *Procedure Development Process*.

11.4.4 Temporary Procedures

Temporary procedures are typically issued to address changes in normal conditions not addressed in operating procedures. These conditions can be related to safety, quality, production, or maintenance concerns. Three types of temporary procedures are used: (1) emergency; (2) standard (valid for up to 90 days from initial start); and (3) long-term (valid for periods not to exceed one year). Long-term temporary procedures are issued for major projects that require a long-term startup phase before facility acceptance and/or process qualification. New temporary procedures of this type require equivalent signatures to new operating procedures.

11.4.5 Periodic Reviews

Periodic reviews of procedures are performed to assure their continued accuracy and usefulness. At a minimum, operating procedures are reviewed every three (3) years, and emergency procedures are reviewed annually. In addition, procedures are reviewed following unusual incidents (such as, an accident, unexpected transient, significant operator error, or equipment malfunction) to determine if changes are appropriate based on the cause and corrective action determination for the particular incident. Periodic reviews of controlled documents shall be conducted at a frequency listed in Table 11-1, *Procedure Periodic Reviews*.

11.4.6 Use and Control of Procedures

Line managers and area managers ensure procedures are made readily available in the work area and that personnel are trained to the requirements of the procedures; compliance is mandatory. Personnel are trained to immediately report inadequate procedures or the inability to follow procedures.

11.4.7 Records

The Safety Program design requires the establishment and maintenance of approved written procedures for EHS limitations and requirements to govern the safety aspects of operations. Requirements for procedure control and approval authorities are documented.

11.4.8 Topics to be Covered in Procedures

Activities defined in Section 11.4.1, *Types of Procedures*, are the minimum activities to be covered by controlled documents. Maintenance activities listed below may be covered by approved written procedures, documented work instructions, or drawings; whichever is appropriate to the circumstance. The list below is not intended to be all-inclusive, as many other activities carried out during operations may be covered by procedures not included in the list. Similarly, this listing is not intended to imply that procedures need to be developed with the same titles as those in the list. This listing provides guidance on topics to be covered rather than specific procedures.

Management Control Procedures

- Training
- Audits and inspections
- Investigations and reporting
- Records management and document control
- Changes in facilities and equipment
- Modification design control
- QA
- Equipment control (lockout/tagout)
- Shift turnover
- Work and management control
- Nuclear criticality safety, fire safety, chemical process safety
- Radiation protection
- Radioactive waste management
- Maintenance
- Environmental protection
- Operations
- IROFS surveillances
- Calibration control
- Procurement

System Procedures that Address Start-Up, Operation, and Shutdown

- Electrical power
- Ventilation
- Shift routines, shift turnover, and operating practices
- Sampling
- UF_6 cylinder handling
- UF_6 material handling equipment

- Decontamination operations
- Facility air and nitrogen
- Cooling, sanitary, and facility water
- Temporary changes in operating procedures
- Purge and evacuation vacuum systems
- Installation and removal of centrifuge machines

Abnormal Operation/Alarm Response

- Loss of cooling, instrument air, and/or electrical power
- Fires
- Chemical process releases
- Loss of feed or withdrawal capacity
- Loss of purge vacuum

Maintenance Activities that Address System Repair, Calibration, Inspection, and Testing

- Repairs and preventive repairs of IROFS
- Calibration and functional testing of IROFS
- High-efficiency particulate air (HEPA) filter maintenance
- Safety system relief valve replacement
- Surveillance/monitoring
- Piping integrity testing
- Containment device testing
- Repair of UF_6 valves
- Testing of cranes
- UF_6 cylinder inspection and testing
- Centrifuge assembly/installation

Emergency Procedures

- Toxic chemical releases (including UF_6)

11.5 AUDITS AND ASSESSMENTS

GLE implements a system of audits and assessments to help ensure that the EHS functions, as described in this LA are adequate and effectively implemented. The system is designed to ensure comprehensive program oversight at least once every three (3) years.

11.5.1 Activities to be Audited or Assessed

11.5.1.1 Assessments

Management performs assessments to verify the effective implementation of the Safety Program elements (RP, NCS, Industrial Safety, Security and Emergency Preparedness, and Environmental Protection), management measures, and QA Program elements. Personnel from the area being assessed may perform the assessment, provided that they do not have direct responsibility for the specific activity being assessed. Results of assessments are documented. The responsible line manager resolves any observations from these programmatic assessments. In addition, GLE commits to perform independent assessments of its safety program elements. The assessment scope includes compliance to procedures, conformance to regulations, and the overall adequacy of the safety program. Assessment results are documented and reported as specified in the approved written procedures. Provisions are made for reporting and corrective action, where warranted, in accordance with the Corrective Action Program.

11.5.1.2 Audits

Representatives of the NCS, RP, and Industrial Safety functions conduct formal scheduled safety audits of uranium enrichment and process support areas in accordance with approved written procedures. These audits are performed to determine if operations conforms to NCS, RP, and Industrial Safety requirements. Audit results are reported in writing to the GLE Facility Manager, the GLE EHS Manager, the NCS Manager, area managers, the manager of the safety function being audited, and other line management as appropriate.

11.5.2 Scheduling of Audits and Assessments

An assessment of each management measure (such as CM) is performed annually. The assessment may focus on a single organizational element or the entire organization. NCS and RP audits are performed quarterly (at intervals not to exceed 110 days) under the direction of the manager of the NCS and RP functions. Facility personnel conduct weekly nuclear criticality safety walkthroughs of uranium enrichment and process support areas in accordance with approved written procedures. Walkthrough findings are documented and sent to the affected line manager or area manager for resolution. In addition, GLE commits to perform triennial independent assessments of its safety program elements. The Environmental Protection function develops an audit schedule for the Environmental Protection Program on an annual basis.

11.5.3 Procedures for Audits and Assessments

Industrial safety audits are performed under the direction of the Industrial Safety Manager. Audit results are communicated in writing to the responsible line manager, GLE Facility Manager, area managers, and to the GLE EHS Manager. Environmental Protection audits are conducted in accordance with approved written procedures to ensure operational activities conform to documented environmental requirements.

Required corrective actions are documented and approved by management and tracked to completion by the EHS function. Records of the audit or inspection, instructions and procedures, persons conducting the audits or inspections, audit or inspection results, and corrective actions for identified violations of license conditions are maintained in accordance with procedural requirements for a minimum period of three years.

11.5.4 Qualifications and Responsibilities for Audits and Assessments

Personnel performing audits do not report to the audited organization and have no direct responsibility for the function being audited. The audit team consists of appropriately trained and experienced individuals. The responsible line manager, or area manager, is responsible for nonconformance corrective action commitments in accordance with approved written procedures. The Environmental Protection Manager, or delegate, is responsible for resolution of identified nonconformances associated with the Environmental Protection Program. Audit results in the form of corrective action items are reported to the GLE Facility Manager and staff for monitoring of closure status.

11.6 INCIDENT INVESTIGATIONS

Incident investigations are performed to assure that the upset condition(s) is understood and appropriate corrective actions are identified and implemented to prevent recurrence. Management Measures include documenting upset conditions in unusual incident reports (UIRs). UIRs are documented and the associated corrective actions tracked to completion. The objectives of the incident investigation and reporting procedures are to establish the validity of the data related to the incident, to develop and implement corrective action plans (CAPs) when appropriate, to document an event which was or could become a danger to persons or property, and to ensure that proper levels of GLE Management and public agencies are notified.

11.6.1 Incident Identification, Categorization, and Notification

GLE commits to maintain a system to identify, track, investigate, and implement corrective actions for abnormal events (unusual incidents). Through this system, GLE will investigate abnormal events that may occur during operation of the facility, determine the specific or generic root cause(s) and generic implications, recommend corrective actions, and report to the U.S. Nuclear Regulatory Commission (NRC) as required by 10 CFR 70.50, *Reporting Requirements (Ref. 11-10)*, and 10 CFR 70.74, *Additional Reporting Requirements (Ref. 11-11)*. The Corrective Action System includes the following requirements and features:

- Operates in accordance with approved written procedures;
- Document, track, and report abnormal events to GLE management;
- Identify abnormal events associated with IROFS or their associated management measures;
- Consider each event in terms of regulatory reporting criteria and in terms of severity, where precursor events are considered unusual events and events concerning compliance with regulations or license conditions are considered potential noncompliances (PNC);
- UIRs require investigation, a determination of root or most probable (proximate) cause, and the identification of required corrective action(s);
- More significant UIRs and PNCs require a formal, systematic determination of root cause (typically using an independent qualified team), creation of a CAP, and a higher level management review and approval of the investigation and corrective actions;
- Issue monthly reports covering the status of UIRs and PNCs to GLE management;
- Grade events for the purpose of an ongoing management evaluation of facility performance and used as one element in driving safety culture focus;
- Maintain records of the events and the documented evidence of closure for a minimum of three years; and

- Use UIR and PNC information where appropriate when performing ISAs.

11.6.2 Conduct of Incident Investigations

Incident investigations are implemented according to approved written procedures. The investigation process includes a prompt risk-based evaluation. The investigator(s) is independent from the function(s) involved with the incident under investigation and are assured of no retaliation for participating in investigations. Investigations shall begin within 48 hours of the abnormal event, or sooner, depending on safety significance of the event. The record of IROFS failures, as required by 10 CFR 70.62(a)(3), *Safety Program and Integrated Safety Analysis (Ref. 11-12)*, shall be reviewed as part of the investigation. Record revisions necessitated by post-failure investigation conclusions shall be made within five working days of the completion of the investigation.

Qualified internal or external investigators are appointed to serve on investigating teams when required. The teams include at least one process expert and at least one team member trained in root cause analysis.

GLE maintains auditable records and documentation related to abnormal events, investigations, and root cause analyses so that "lessons learned" may be applied to future operations of the facility. For each abnormal event, the incident report includes a description, contributing factors, a root cause analysis, findings, and recommendations. Relevant findings are reviewed with affected personnel. Details of the event sequence are compared with accident sequences already considered in the ISA, and the ISA Summary will be modified, if necessary, to include evaluation of the risk associated with accidents of the type actually experienced. The Incident Investigation Process consists of the following steps:

- Investigate the problem;
- Derive an understanding of the issues and drivers, and determine the fundamental or root cause(s);
- Develop appropriate corrective and preventive actions;
- Assign responsible individual(s) to address each corrective or protective action, determine the required timing for each action, and provide scheduled target date for each action;
- Compile adequate records (hard copy or electronic files) to demonstrate completion or closure of the corrective actions;
- Conduct an investigation to determine if the corrective action(s) was appropriate;
- Assure identified corrective actions are completed in an appropriate and timely manner;
- Input the corrective action completion data, documentation, and any related notes of interest in a hard copy or electronic copy file;

- Provide appropriate GLE management with closure documentation for internal type items (such as, UIRs) or input the closure documentation electronically into the controlled electronic file in sufficient detail to demonstrate closure of the action; and

- Provide the Licensing Organization with closure documentation for external agency items (that is, NRC, State of North Carolina, American Nuclear Insurers, Factory Mutual, etc.) or input the documentation electronically into the controlled electronic file.

11.6.3 Written Follow-Up Report

Upon completion of the incident investigation, a report on the incident and the associated investigation is made to ensure sufficient corrective and preventive actions has been defined and completed. The report contains sufficient detail to demonstrate closure of the action. At least quarterly, a status report is issued by the EHS function and distributed to individuals responsible for corrective actions and management.

11.6.4 Corrective Actions

The line managers and area managers have the responsibility to ensure proper action is taken to control the incident in the assigned area of responsibility to include: consulting EHS for a determination as to whether or not the investigation of an incident is required, notifying appropriate management, participating in the investigation as required, and assuring adequate corrective actions are completed. The line managers and area managers are responsible for reviewing and approving the corrective actions associated with each UIR in their area of responsibility. This is accomplished by the creation of a corrective action within each UIR.

11.7 RECORDS MANAGEMENT

11.7.1 Records Management Program

RM shall be performed 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 written procedures. QA records are not considered valid until they are authenticated and dated by authorized personnel.

The GLE QA Program requires procedures for reviewing, approving, handling, identifying, retention, retrieval, and maintenance of QA records. These records include the results of tests and inspections required by applicable codes and standards, construction, procurement and receiving records, personnel certification records, design calculations, purchase orders, specifications and amendments, procedures, incident investigation results and approvals or corrective action taken, various certification forms, source surveillance and audit reports, component data packages, and any other QA documentation required by specifications or procedures. These records are maintained at locations where they can be reviewed and audited to establish that the required quality has been assured.

For computer codes and computerized data used for activities relied on for safety, as specified in the ISA Summary, procedures are established for maintaining readability and usability of older codes and data as computing technology changes. For example, procedures allow older forms of information and codes for older computing equipment to be transferred to contemporary computing media and equipment.

RM shall maintain a Master File to which access is controlled. Documents in the Master File shall be legible and identifiable as to the subject to which they pertain. Documents shall be considered valid only if stamped, initialed, signed or otherwise authenticated, and dated by authorized personnel. Documents in the Master File may be originals or reproduced copies. Computer storage of data may be used in the Master File. In order to preclude deterioration of records in the Master File, the following requirements are applicable:

- Records shall not be stored loosely. Records shall in binders or placed in folders or envelopes. Records shall be stored in steel file cabinets.
- Special processed records, such as, radiographs, photographs, negatives, microfilm, which are light-sensitive, pressure-sensitive, and/or temperature-sensitive, shall be packaged and stored as recommended by the manufacturer of these materials.
- Computer storage of records shall be done in a manner to preclude inadvertent loss and to ensure accurate and timely retrieval of data. Dual-facility records storage uses an electronic data management system and storage of backup tapes in a fireproof safe.

The Master File storage system shall provide for the accurate retrieval of information without undue delay. Approved written instructions shall be prepared regarding the storage of records in a Master File, and a supervisor shall be designated the responsibility for implementing the requirements of the instructions. These instructions shall include, but not necessarily be limited to, the following:

- A description of the location(s) of the Master File and an identification of the location(s) of the various record types within the Master File;
- The filing system to be used;
- A method for verifying that records received are in good condition and in agreement with any applicable transmittal documents. This is not required for documents generated within a section for use and storage in the same sections' satellite files;
- A method for maintaining a record of the records received;
- The criteria governing access to and control of the Master File;
- A method for maintaining control of and accountability for records removed from the Master File; and
- A method for filing supplemental information and for disposing of superseded records.

Record storage areas (including satellite files) shall be evaluated to assure records are adequately protected from damage by fire.

11.7.2 Record Retention

Records appropriate for ISAs, IROFS, the application of management measures to IROFS, NCS and RP activities, training/retraining, occupational exposure of personnel to radiation, releases of radioactive materials to the environment, and other pertinent safety activities are maintained in such a manner as to demonstrate compliance with license conditions and regulations.

Records of Criticality Safety Analyses (CSAs) are maintained in sufficient detail and form to enable independent review and audit of the calculational method and results. Records associated with personnel radiation exposures are generated and retained in such a manner as to comply with the relevant requirements of 10 CFR 20, *Standards for Protection Against Radiation (Ref. 11-13)*. In addition, the following RP records are maintained for at least three (3) years:

- Records of the Facility Safety Review Committee (FSRC) meetings,
- Surveys of equipment for release to unrestricted areas,
- Instrument calibrations,

- Safety audits,
- Personnel training and retraining,
- Radiation work permits,
- Surface contamination surveys,
- Concentrations of airborne radioactive material in the facility, and
- Radiological Safety Analyses (RSAs).

Records associated with Environmental Protection activities described in GLE LA Chapter 9, *Environmental Protection*, are generated and retained in such a manner as to comply with the relevant requirements of 10 CFR 20.

11.7.3 Organization and Administration

11.7.3.1 Responsibilities

The Quality Assurance and Infrastructure Program Manager is responsible for the RM Program during the design and construction phases of the project. The Infrastructure Program Manager is responsible for the RM Program during the Operations phase. The RM Program functions include directing the development, implementation, and maintenance of methods and procedures encompassing a RM Program, and assuring the laws, codes, standards, regulations, and company procedures pertaining to record keeping requirements are met.

11.7.3.2 Training and Qualifications

Appropriately trained and qualified personnel manage the RM Program. No specific experience related to the control of documents or management of records is required, although previous technical or RM experience is recommended.

11.7.3.3 Employee Training

General training in RM is provided to employees as part of the general topics covered in GET. Specific professional development training shall be provided on an as needed basis.

11.7.3.4 Examples of Records

The following are examples of the types of records maintained by the RM Program.

<u>General Information</u>

- Construction records
- Safety analyses, reports, and assessments

- Facility and equipment descriptions and drawings
- Design criteria, requirements, and bases for IROFS
- Records of facility changes and associated ISAs
- Records of site characterization measurements and data
- Records pertaining to onsite disposal of radioactive or mixed wastes in surface landfills
- Procurement records, including specifications for IROFS

Organization and Administration

- Administrative procedures with safety implications
- Change control records for Material Control and Accounting (MC&A) Program
- Organization charts, position descriptions, and qualification records
- Safety and health compliance records, medical records, personnel exposure records
- QA records
- Safety inspections, audits, assessments, and investigations
- Safety statistics and trends

Integrated Safety Analysis

- ISA and ISA-related analyses

Radiation Safety

- Bioassay data
- Exposure records
- Radiation protection (and contamination control) records
- Radiation training records
- Radiation work permits

Nuclear Criticality Safety

- Nuclear criticality control approved written procedures and statistics

- NCS evaluations
- Records pertaining to nuclear criticality inspections, audits, investigations
- Records pertaining to nuclear criticality incidents, unusual occurrences, or accidents
- Records pertaining to NCS evaluations

Chemical Safety

- Chemical process safety procedures, plans, diagrams, charts, and drawings
- Records pertaining to chemical process inspections, audits, investigations, and assessments
- Records pertaining to chemical process incidents, unusual occurrences, or accidents
- Chemical process safety reports and analyses
- Chemical process safety training

Fire Safety

- Fire Hazard Analysis
- Fire prevention measures, including hot-work permits and fire watch records
- Records pertaining to inspection, maintenance, and testing of fire protection equipment, and records pertaining to fire protection training and retraining of response teams
- Pre-fire emergency plans

Emergency Management

- Emergency plan(s) and procedures, and comments on emergency plan from outside emergency response organizations
- Emergency drill records
- Memoranda of understanding (MOU) with outside emergency response organizations
- Records of actual events, records pertaining to the training and retraining of personnel involved in Emergency Preparedness functions, and records pertaining to the inspection and maintenance of emergency response equipment and supplies

Environmental Protection

- Environmental release and monitoring records
- Environmental report and supplements to the environmental report, as applicable

Decommissioning

- Decommissioning records, cost estimates, and procedures
- Financial assurance documents
- Site characterization data
- Final survey data

Management Measures

- Configuration Management
  - Safety analyses, reports, and assessments that support the physical configuration of process designs and changes to those designs
  - Validation records for computer software used for safety analyses or MC&A
  - ISA documents, including process descriptions, facility drawings and specifications, purchase specifications for IROFS
  - Approved current operating procedures and emergency operating procedures
- Maintenance
  - Record of IROFS failures (required by 10 CFR 70.62)
  - PM records, including trending and root cause analysis
  - Calibration and testing data for IROFS
  - Corrective maintenance records
- Training and Qualification
  - Personnel training and qualification records
  - Training procedures and modules
- Operating procedures and functional test procedures
- Audits and Assessments of safety and environmental activities

- Incident Investigations
  - Investigation reports
  - Changes recommended by investigation reports, how and when implemented
  - Summary of reportable events for the term of the license
  - Incident investigation policy
- Records Management
  - Policy
  - Material storage records
  - Records of receipt, transfer, and disposal of radioactive material
- Other QA Elements
  - Inspection records
  - Test records
  - Corrective action records

11.8 OTHER QUALITY ASSURANCE ELEMENTS

GLE has developed a QA Program that applies to the design, construction, operation, and decommissioning of the GLE Commercial Facility. Application of the QA Program is mandatory for items (SSCs, equipment, and activities) identified as IROFS in accordance with 10 CFR 70.4, *Definitions (Ref. 11-14),* 10 CFR 70.61, *Performance Requirements (Ref. 11-15),* 10 CFR 70.64, and 10 CFR 21, *Reporting of Defects and Noncompliance (Ref. 11-16).* The QA Program, in conjunction with the other management measures, ensures IROFS will be available and reliable to perform the required safety functions when needed. The QA Program is described in the Quality Assurance Program Description for the Global Laser Enrichment LLC Commercial Facility (NEDE 33451).

11.9 REFERENCES

11-1. 10 CFR 70.72, *Facility Changes and Change Process*, U.S. Nuclear Regulatory Commission, 2008.

11-2. 10 CFR 70.64, *Requirements for New Facilities or New Processes at Existing Facilities*, U.S. Nuclear Regulatory Commission, 2008.

11-3. 29 CFR 1910, *Occupational Safety and Health Standard*, Occupational Safety and Health Administration, 2008.

11-4. 10 CFR 19, *Notices, Instructions, and Reports to Workers: Inspection and Investigation*, U.S. Nuclear Regulatory Commission, 2008.

11-5. 10 CFR 19.12, *Instruction to Workers*, U.S. Nuclear Regulatory Commission, 2008.

11-6. ANSI/ANS 8.19-2005, *Administrative Practices for Nuclear Criticality Safety*, American Nuclear Society, January 2005.

11-7. ANSI/ANS 8.20-1991 (R1999), *Nuclear Criticality Safety Training*, American Nuclear Society, January 1991.

11-8. NUREG-0700, *Human-System Interface Design Review Guidelines*, U.S. Nuclear Regulatory Commission, Revision 2, May 2002.

11-9. NUREG-0711, *Human Factors Engineering Program Review Model*, U.S. Nuclear Regulatory Commission, Revision 2, February 2004.

11-10. 10 CFR 70.50, *Reporting Requirements*, U.S. Nuclear Regulatory Commission, 2008.

11-11. 10 CFR 70.74, *Additional Reporting Requirements*, U.S. Nuclear Regulatory Commission, 2008.

11-12. 10 CFR 70.62, *Safety Program and Integrated Safety Analysis*, U.S. Nuclear Regulatory Commission, 2008.

11-13. 10 CFR 20, *Standards for Protection Against Radiation*, U.S. Nuclear Regulatory Commission, 2008.

11-14. 10 CFR 70.4, *Definitions*, U.S. Nuclear Regulatory Commission, 2008.

11-15. 10 CFR 70.61, *Performance Requirements*, U.S. Nuclear Regulatory Commission, 2008.

11-16. 10 CFR 21, *Reporting of Defects and Noncompliance,* U.S. Nuclear Regulatory Commission, 2008.

11-17. 10 CFR 70, *Domestic Licensing of Special Nuclear Material*, U.S. Nuclear Regulatory Commission, 2008.

Table 11-1. Procedure Periodic Reviews.

Document	Review Frequency	Reviewing and Approving Functional Manager
Business Policy	When changed	CEO of affected GEH business unit(s)
Management Control Procedure	When changed [a]	Area manager, line manager, and affected EHS functions (radiation, criticality, environmental, industrial [d], or material control and accounting)
Operating Procedure	Every 3 Years [c]	Area manager, line manager, and affected EHS functions (radiation, criticality, environmental, industrial [d], or material control and accounting)
Nuclear Safety Instruction	Every 2 Years [b]	Radiation and criticality safety
Environmental Protection Instruction	Every 2 Years [b]	Environmental protection
Emergency Procedure	Annually	Area manager, line manager, and affected EHS function

[a] The safety awareness portions of these procedures are reviewed and updated by the appropriate Environmental, Health, and Safety (EHS) function when warranted based on process related facility change requests.

[b] Every two (2) years means a maximum interval of 26 months.

[c] Every three (3) years means a maximum interval of 39 months.

[d] EHS function - industrial means normal worker safety, chemical safety, and fire and explosion protection.