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 topography of the Wilmington Site and the general area is level terrain; therefore, water accumulation (flooding) is expected to be a slow event providing ample warning to Operations personnel. GLE CF processes could be safely shut down prior to any potential threat of flooding. In addition, there are no dams on the Northeast Cape Fear River, but there are several dams upstream on the Cape Fear River. The closest dams and locks on the Cape Fear River are approximately 30, 50, and 70 miles upstream of where the Northeast Cape Fear River comes into the Cape Fear River. Seismic induced failures of these dams and locks could cause flooding at the GLE site due to the general area being an estuary and in general about the same elevation. The lower Cape Fear River estuary has a tidal reach extending all the way up to the first dam and lock. Due to the level terrain the flood potential due to upstream dam and lock failures (e.g., due to seismic activity) would not likely result in a higher level of flooding than the design basis event from probable maximum flood caused by rainfall on the watersheds.

The nearest river to the GLE site is the Northeast Cape Fear River. The Northeast Cape Fear River watershed covers 1750 square miles. Six miles south of the site, the Northeast Cape Fear River joins the Cape Fear River. The Cape Fear River has a watershed of 9149 square miles. The Design Basis Flood at the site is based on evaluating flooding on the Northeast Cape Fear River and also by evaluating flooding on the Cape Fear River. Flooding on the Cape Fear River has the potential for causing flooding on the Northeast Cape Fear River since the site is only six miles from where the two rivers join together. These conditions were shown to have the highest flood potential, and are used as the "Highly Unlikely" flood event, and could result in about 3 feet of flooding in the facility structures.

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 Virginia, 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.

Using 12,899 events published by the Virginia Tech Seismological Observatory, with augmented catalogs extracted from the Advanced National Seismic System (ANSS) for more recent events, 896 unique earthquakes were located within a 200-mile radius of the Wilmington Site between 1698 and 2007. The earliest instrument-based event locations were recorded in 1925; however, reliable, spatially diverse networks did not accumulate earthquake locations and magnitudes until the early 1970s.

Prior to 1924, there are no events whose horizontal location error is less than 12.4 miles and not until 1965 was the network large enough to provide constraints to locate events with a 2.5-mile error. The median horizontal error for events in the 200-mile radius is 51.6 miles, with an inter-quartile distance (spread) of 51 miles. Prior to 1973, there were no estimates of uncertainty for hypocenter depths. After installation of seismic arrays, event-depth uncertainty varies from 0 to 62 miles, with a median of 0.9 miles and an inter-quartile spread of 1.7 miles.

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, South Carolina earthquake of 1886 was felt in Wilmington, producing modified Mercalli intensities of V – VI, which is considered as light to moderate shaking effects. Since then there have been ten recorded seismological events in the Wilmington Area, all of which have been minor in nature, producing effects no greater than Mercalli IV in the Wilmington area. One example is the August 23, 2011 earthquake in the Piedmont Physiographic Province and the Central Virginia Seismic Zone. The 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 Virginia, 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.

As fully described in the ISA Summary, GLE will use a performance goal of 1×10^{-4}/year for "Highly Unlikely" seismic event.

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 identified as events of concern. 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.

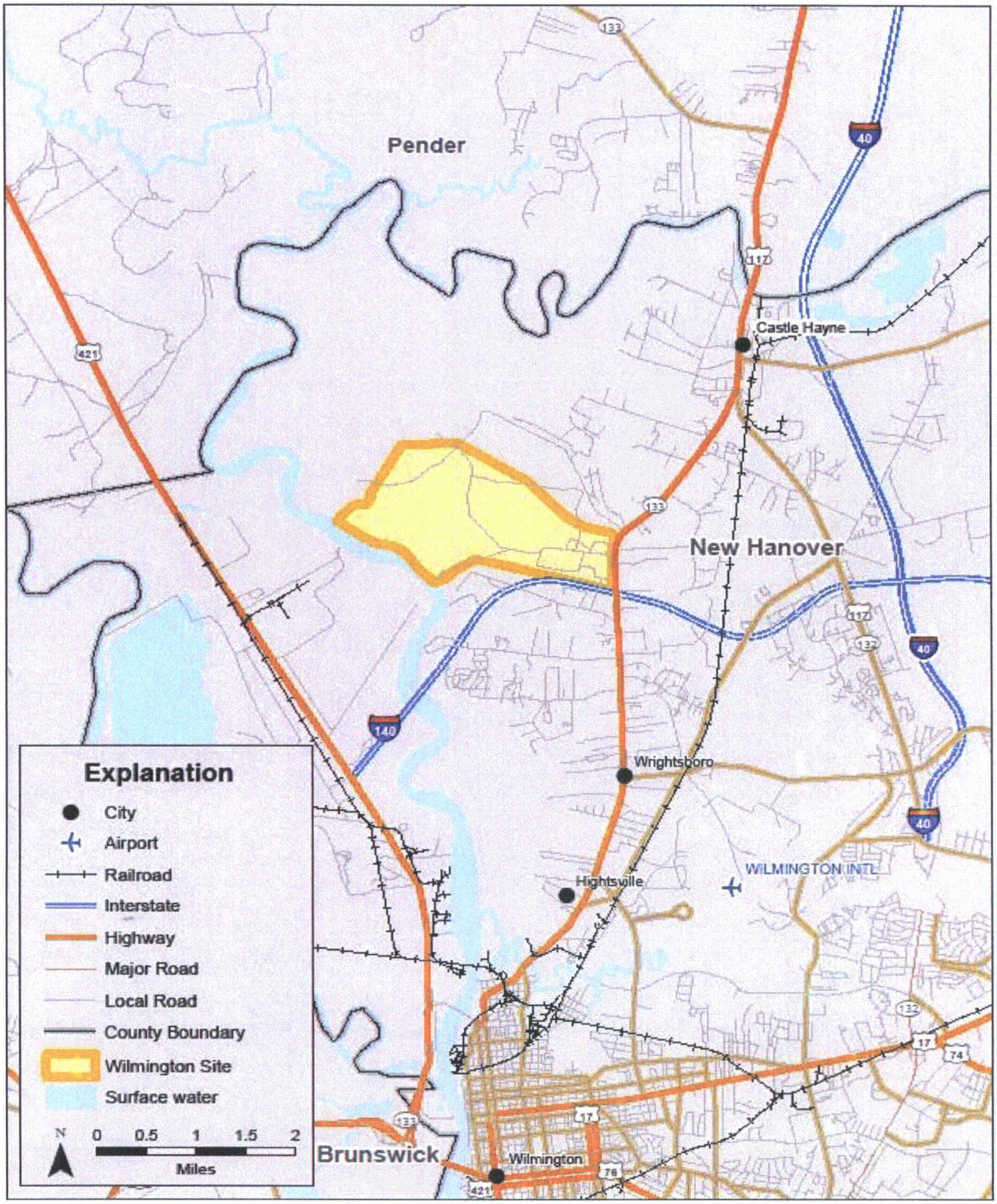
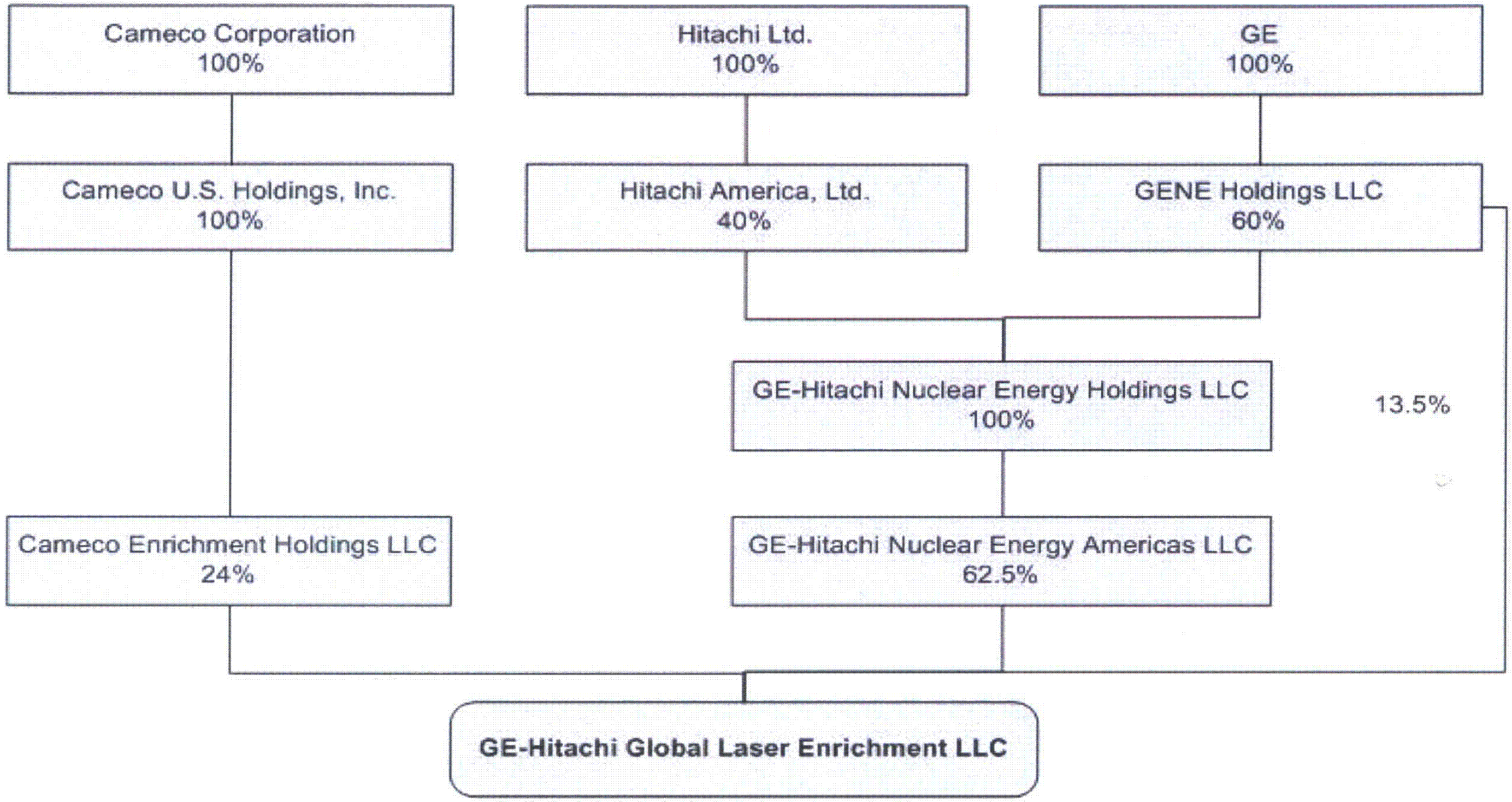


Figure 1-3. {{{Proprietary Information withheld from disclosure per 10 CFR 2.390}}}

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.

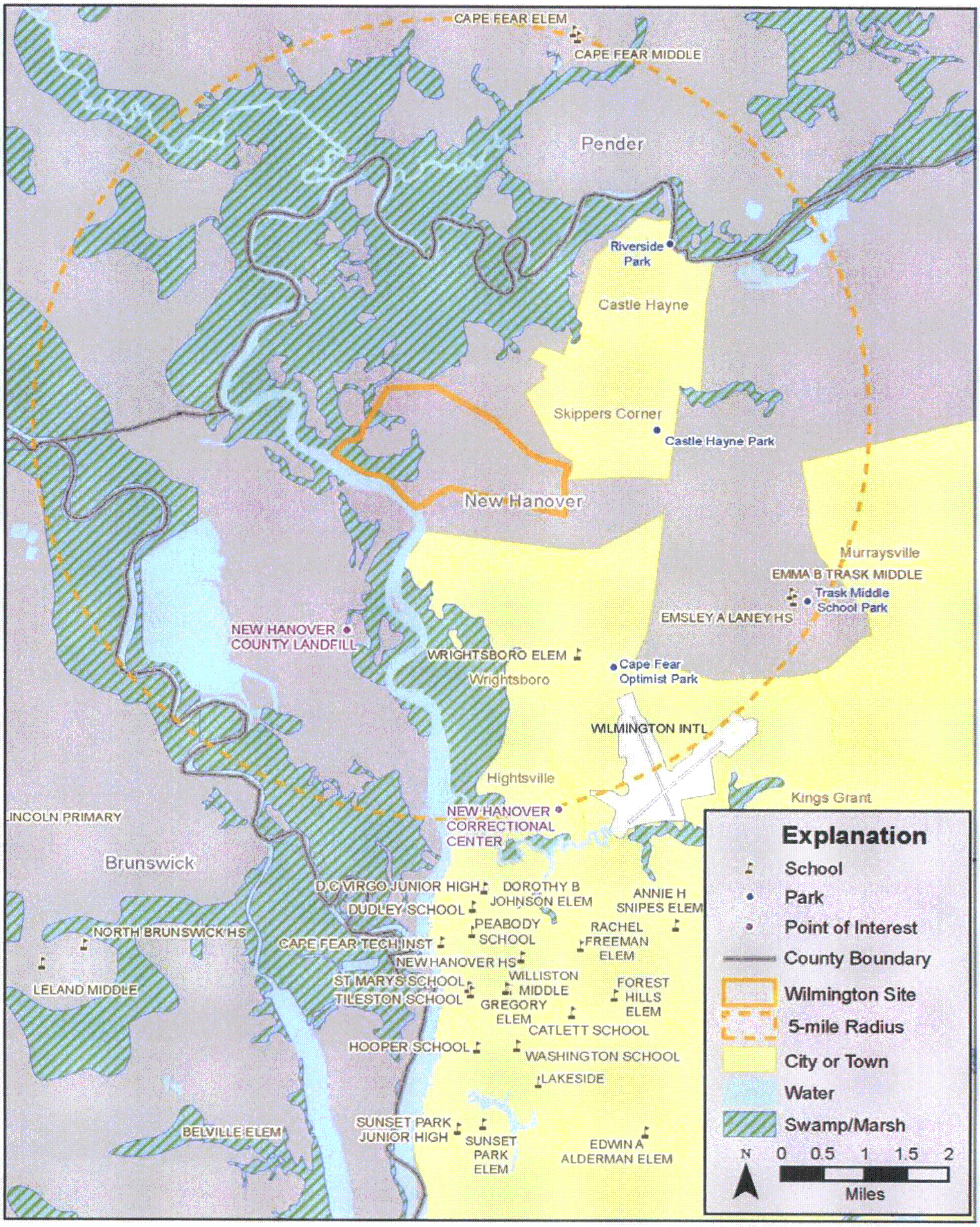
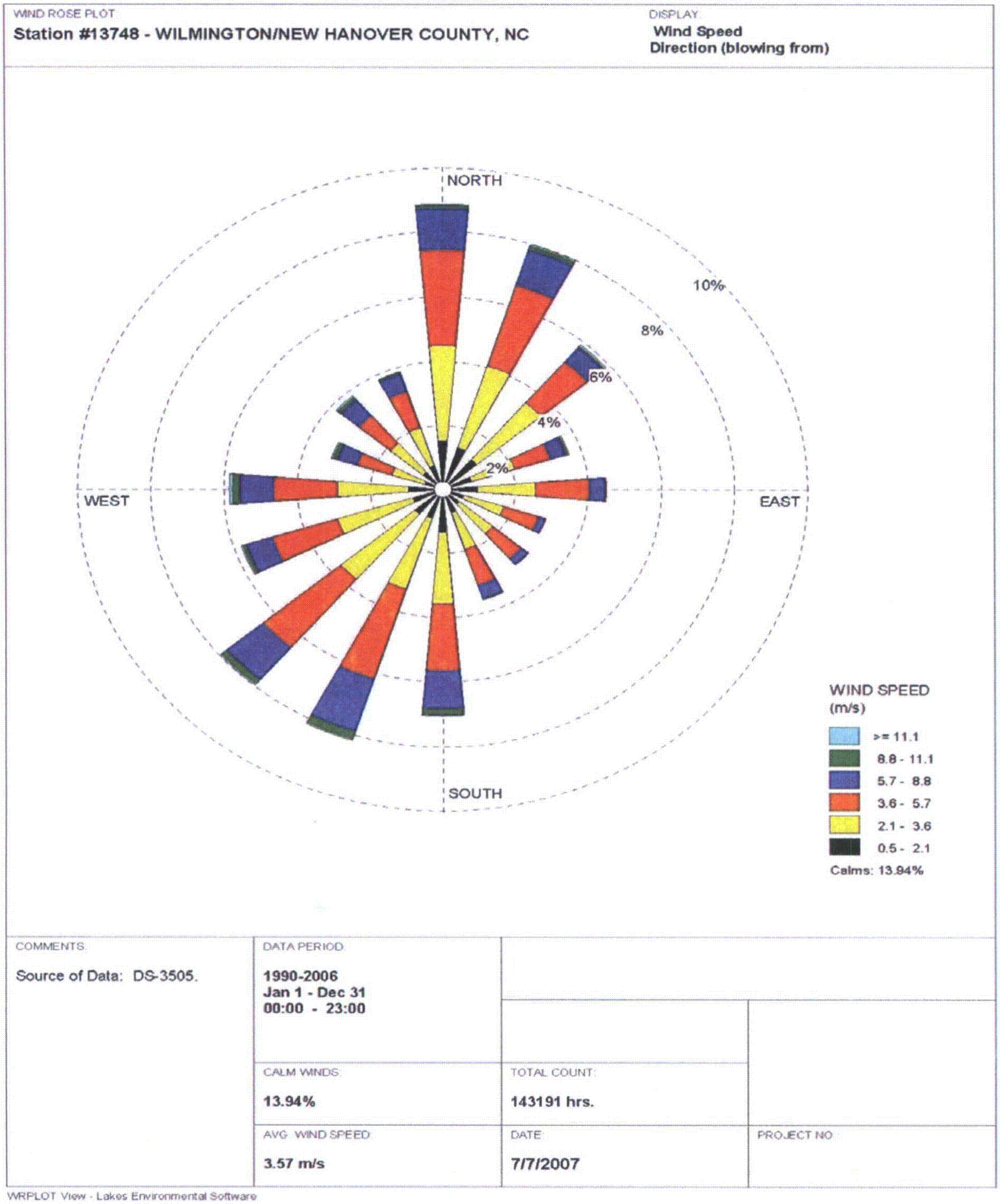


Figure 1-7. Wind Rose for Wilmington International Airport.



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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 $\beta\gamma$/ 100 cm^2	15,000 dpm $\beta\gamma$/ 100 cm^2	1,000 dpm $\beta\gamma$/ 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 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.
5	08/12/2011	ALL	Revised use terminology consistent with the ISA Summary
6	10/14/2011	12, 13, 21, 26, 31, 41, 44	Revised Section 3.2.4.4.10 to incorporate specific electrical and instrumentation and control commitments. Revised Section 3.2.5.5.1 to reference Table 3-11. Updated Section 3.2.8 to include addition of a sole IROFS. Revised Table 3-1 to include additional standard revisions. Revised Table 3-7 to remove "Credible" designation from table entries and to update footnote. Added Table 3-11.

Rev.	Effective Date	Affected Pages	Revision Description

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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.

GLE commits to the applicable portions of the following standards taking into consideration the note attached to Table 3-1. Applicability is based on the level of credit applied to the IROFS, as identified in the ISA Summary. Instrumentation and control IROFS components and systems will be qualified to meet the guidance in IEEE-323, "IEEE Standard for Qualifying Class 1 E Equipment for Nuclear Power Generating Stations" (IEEE, 1983). GLE will use design criteria appropriate for maintaining electrical independence between safety related and non-safety related systems. Independence and isolation will be achieved using the appropriate guidance of Regulatory Guide 1.75 (2005) and IEEE 384, "Standard Criteria for Independence of Class 1E Equipment and Circuits" (IEEE, 1992) for establishing separation criteria between IROFS and non-IROFS equipment. For seismic qualification of IROFS equipment that are required to remain operable during and after seismic events, GLE will use IEEE 323 (IEEE, 1983) and IEEE 344, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generation Stations" (IEEE, 2004).

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, *Example Procedure for Accident Sequence Evaluation*, to identify hazards and evaluate accident sequences. This approach employs a semi-quantitative risk index method for categorizing unmitigated event sequences in terms of their consequences of concern and their likelihood of occurrence. The risk index method framework identifies which unmitigated event sequences have consequences that could exceed the performance requirements of 10 CFR 70.61; and therefore, are identified as accident sequences that require the designation of IROFS and supporting management measures. Descriptions of these general types of higher-consequence accident sequences are reported in the ISAS. 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, such as, What-If Analysis, What-If/Checklist, or Hazard and Operability Analysis (HAZOP) depending type of activity analyzed, to identify the hazards relevant to each node or the facility in general. The ISA Team reviewed the hazards qualitatively identified for the "credible worst-case" consequences. The credible high or intermediate severity consequence unmitigated event sequences were assigned accident description identifiers, accident descriptions, qualitative frequency or probability, and then, a risk index determination was performed. The risk index was used to evaluate unmitigated risk as acceptable or unacceptable.

For each unmitigated event sequence having an unacceptable unmitigated risk index, potential safeguards/IROFS were identified by the team. A Quantitative Risk Analysis (QRA) was performed to quantify both the likelihood and the consequence severity in detail. Where the unmitigated risk continued to be unacceptable, IROFS were identified to mitigate the likelihood or the consequence for an accident sequence. X`

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 sequences that result in undesired outcomes for each initiating event. Each QRA report provides details concerning an accident sequences'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 event sequences 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. Likelihood – 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. Safeguards/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 Event Sequences

The goal is to identify credible event sequences by analyzing single initiating events. Using approved methods, the ISA Team identified potential unmitigated event sequences 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 unmitigated event sequences identified and recorded during the analysis process. An unmitigated event sequence 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 unmitigated event sequence is a sequence of specific real events.

When analyzing unmitigated event sequences, 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 event sequences with adverse consequences. The team-listed sequences considered not credible. In addition to normal conditions, the team considered abnormal conditions including startup, shutdown, maintenance, and process upsets.

For each unmitigated event sequence, enabling conditions, and conditional events that affect the outcome of the unmitigated event sequence (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 event sequence 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 unmitigated event sequences at the GLE Commercial Facility, it is necessary to determine which event sequences are considered not credible and which are credible. When conducting the PHA, the ISA Team considered each unmitigated event sequence as credible, unless the sequence 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 the unmitigated event sequence 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 unmitigated event sequence 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 unmitigated event sequence 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 unmitigated event sequence, 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: "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 unmitigated event sequence occurring was determined for the unmitigated case (unmitigated likelihood). Unmitigated likelihood is the likelihood or frequency that the initiating event or cause of the 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 unmitigated event sequence 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 unmitigated event sequence where multiple initiating events have been identified, the team estimated the likelihood for the most credible initiating event. This helped ensure that the unmitigated event sequence 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 below into a guideline limiting the frequency of individual accidents to 10^{-5} per-event per-year for probabilistic consideration. For selected NPHs, deterministically defined events, as opposed to probabilistically identified events were applied where frequency data is difficult to define quantitatively and varies from site to site. The deterministically defined events, developed consistent with guidance for nuclear power plants, are being applied qualitatively, to the proposed facility as "Highly Unlikely" events per the allowances of Interim Staff Guidance-8, *Natural Phenomenon Hazards.* Where NPH frequency data is available and the deterministic derivation of magnitude and return period does not compare favorably, "Highly Unlikely" will be adjusted to a larger magnitude, less frequent event that does compare favorably. As the goal is to have no such accidents, accident frequencies should be reduced substantially below this guideline when feasible.

The "Highly Unlikely" NPH event criteria, as derived in Chapter 2 of the ISA Summary using the above definition, are presented in Table 3-11.

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 x 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 unmitigated event sequences 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 unmitigated event sequence, 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 unmitigated event sequence indicates the relative importance of the associated controls. Unmitigated event sequences in 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*. These event sequences are carried forward as accident sequences.

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 unmitigated event sequence having an unacceptable unmitigated risk index, IROFS must be defined and the mitigated likelihood determined to develop the accident sequence. 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. As an alternative to the quantification of the accident sequence and the use of an event tree, the accident sequence may be qualitatively presented. Determination of the mitigated likelihood for an accident sequence 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 sequences that result in undesired outcomes for each initiating event. Each QRA report provides details concerning an accident sequence's quantification (or qualification), 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 (or qualification) results from each QRA are summarized in this ISA Summary.

The mitigated likelihood of the accident sequence 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 sequence. The values of the index numbers for an accident sequence, 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 sequences 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. Additional controls in the form of design features, administrative controls, and/or administrative programs may be selected to provide defense-in-depth, but are not IROFS and are not credited with preventing and/or mitigating accident sequences. 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 and defense-in-depth 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 sequence 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 unmitigated event sequences that are of low consequence, or that have a risk index of 4 or less without IROFS applied, 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 unless absolutely necessary. The instances where using a sole IROFS have been determined necessary are with regard to selected use of the Building and Equipment Support Structures IROFS (to address seismic events) and an IROFS for placing or confirming operations in Standby Mode and evacuating personnel from the proposed facility (to address high wind events). For other instances a minimum of two independent IROFS are typically selected.

Information pertaining to sole IROFS Nos. IC-01, *Building and Equipment Support Structures*, and IC-02, *Standby Operations and Personnel Evacuation for High Wind Events*, can be found in the ISA Summary, Chapter 4.16, Appendix B, and Appendix C.

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	43-05	2005	Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities
ASCE	7-5/7-10	2005/2010	Minimum Design Loads for Buildings and Other Structures
ASCE	4-98	1998	Seismic Analysis of Safety-Related Nuclear 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

Code Group / Reference	Code Number	Year or Edition	Title
IAEA	TS-R-1	2009	Regulations for the Safe Transport of Radioactive Material
IBC	2006	2006	2006 International Building Code,
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	1983/2008	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	1992/2008	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

Code Group / Reference	Code Number	Year or Edition	Title
IEEE	1202	2006	IEEE Standard for Flame Testing of Cables For Use in Cable Tray in Industrial and Commercial Occupancies
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®

Code Group / Reference	Code Number	Year or Edition	Title
NFPA	75	2009	Protection of Information Technology Equipment
NFPA	80	2007	Standard for Fire Doors and Other Opening Protectives
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

Code Group / Reference	Code Number	Year or Edition	Title
NFPA	780	2008	Standard for the Installation of Lightning Protection Systems
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

Code Group / Reference	Code Number	Year or Edition	Title
NUREG	1391	1991	Chemical Toxicity of Uranium Hexafluoride Compared to Acute Effects of Radiation
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	**3**	More than or equal to 10^{-4} per-event per-year
Unlikely-	**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.
*The value of 10^{-5} is for items probabilistically considered. For selected NPHs, deterministically defined events, as opposed to probabilistically identified events were applied. The deterministically defined events, developed consistent with guidance for nuclear power plants, are discussed in Section 3.2.5.5.1 and shown in Table 3-11, *Defined "Highly Unlikely" NPH Event Criteria*, and are likely associated with event probabilities in the 10^{-4} range.

Table 3-8. Determination of Likelihood Category.

Likelihood Category	Likelihood Index T* (= sum of index numbers)
1	T ≤ -5
2	-5 < T ≤ -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. Likelihood categories for the "Highly Unlikely" NPH events are assigned 1.

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)

Table 3-11. Defined "Highly Unlikely" NPH Event Criteria

NPH Event	"Highly Unlikely" Event Definition*
Earthquake	Probability of 1×10^{-4}/yr for seismic events performance goal as demonstrated using ASCE 43-05 (See ISA Summary Chapter 2, Section 2.5.1.4, *Seismic Hazard Characterization*)
Hurricane	157.5 mph, 3-second gust wind speed (See ISA Summary Chapter 2, Section 2.5.5, *Hurricanes*)
Tornado	The initiating event of a tornado impacting the facility is "Highly Unlikely" (probability of $<1 \times 10^{-4}$/yr) (See ISA Summary Chapter 2, Section 2.5.6, *Tornadoes*)
Flood	3 feet water level above 25 feet Mean Sea Level (See ISA Summary Chapter 2, Section 2.5.3, *Floods*)
Extreme Rain	Flood potential bounded by above (See ISA Summary Chapter 2, Section 2.5.3, *Floods*)
Extreme Snow	25 psf loading, drifts capable of higher loading (up to 85 psf) (See ISA Summary Chapter 2, Section 2.5.7.1, *Ice and Snow Accumulations*)
Tsunami	"Highly Unlikely" for a tsunami to impact facility (NUREG/CR-6966) (See ISA Summary Chapter 2, Section 2.5.4, *Tsunami*)
Volcano	"Highly Unlikely" for a volcano to impact facility (per USGS, no known or perceived volcanic activity in the southeastern region of the United States) (See ISA Summary Chapter 2, Section 2.5.2, *Volcanoes*)

* Deterministically defined NPH events where reliable frequency data are available are considered "Highly Unlikely" when the deterministically defined NPH events meet the performance goal of 1×10^{-4}/year or less or occur at frequencies of 1×10^{-4}/year or less

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