Global Nuclear Fuel

A Joint Venture of GE, Toshiba, & Hitachi

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April 2, 2007

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Subject: License Renewal Application for Global Nuclear Fuel – Americas LLC
Reference: NRC License SNM-1097, Docket 70-1113

Global Nuclear Fuel – Americas LLC (GNF-A) hereby makes application to renew Special Nuclear Materials License SNM-1097 for a term of 40 years. The current license expires June 30, 2007. On the basis of this application GNF-A further requests that the authorized activities in the existing license continue in effect until the application has been finally determined as provided in 10 CFR 2.109 (a).

In June 2006 GNF-A met with members of the NRC Staff and discussed our plan for the license renewal application. Noted at that time was the fact that the current license had been in effect for nearly 10 years and had performed well with no questions of interpretation during inspections and therefore most portions of the license needed little change other that bringing them up-to-date. It was also noted that 10 CFR 70, Subpart H had imposed requirements that required new chapters for "Integrated Safety Analysis" and "Management Measures". A supplement to the site Environmental Report covering the period from the last supplement dated April 1996 (data through 1994) was to be provided, and the Radiological Contingency & Emergency Plan and Decommission and Closure Plan were to be updated to be current at the time of the renewal request. No action was to be required for the Material Control and Accounting Plan or the Site Security Plan.

Consistent with the June 2006 renewal plan, GNF-A has taken the following actions:

1. Reviewed and updated License Renewal Application Chapters 1, 2, 4, 6, 7, 8, 9 and 10 to be current and accurate. Chapter 1 includes provisions for "Laboratory Scale Laser Enrichment Research and Development" (1.2.3.3) and "Possession of Classified Matter" (1.2.3.8).

2. Updated License Renewal Application Chapter 5 to reflect improved validations for the criticality safety codes and methods already reviewed and approved by the NRC and

remove Table 6.0 "Plant Systems and Parameter Controls", which has been replaced with the identification of IROFS and provided in the ISA Summaries.

3. Prepared a new License Renewal Application Chapter 3 "Integrated Safety Analysis" patterned after the NRC approved plan previously submitted to meet the requirements of 10 CFR 70.62 (c) (3) (i).

4. Prepared a new License Renewal Application Chapter 11 "Management Measures" generally following the review criteria in NUREG – 1520.

5. Prepared an Environmental Report Supplement covering 1995 – 2005.

6. Submitted an updated Radiological Contingency & Emergency Plan dated 1/8/07 on January 18, 2007.

7. Submitted and updated the Decommission and Closure Plan dated 2/2/07 on February 2, 2007.

Attachment 1 to this letter is the complete updated SNM License Renewal Application. Attachment 2 to this letter is the updated Site Environmental Report Supplement for the period 1995 – 2005.

We would be pleased to discuss this application with you and your staff and to have NRC review personnel visit our facility as appropriate to facilitate the review process.

If you have questions please contact me at (910) 675 5950.

Sincerely,

S.P. Murray, Manager
Licensing and Liabilities COE

Attachments

cc: SPM 07-017
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O. Lopez, NRC Region II, Atlanta, GA
B. Hall, NCDENR Raleigh, NC

Attachment 1

License Renewal Application
For
Global Nuclear Fuel – Americas LLC

NRC License SNM-1097, Docket 70-1113

GLOBAL NUCLEAR FUEL – AMERICAS, L.L.C. (GNF-A)

WILMINGTON, NORTH CAROLINA

LICENSE RENEWAL APPLICATION

March 30, 2007

USNRC MATERIALS LICENSE SNM-1097

DOCKET 70-1113

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CHAPTER 1.0
GENERAL INFORMATION

1.1 FACILITY AND PROCESS DESCRIPTION

The primary purpose of the Global Nuclear Fuel - Americas, L.L.C. facility in Wilmington, North Carolina (identified in this document as GNF-A) is the manufacture of fuel assemblies for commercial nuclear reactors. Nuclear materials enriched to less than or equal to 5 weight percent U-235 are utilized in the product manufacturing operations authorized by this license. The safety, environmental, quality assurance and emergency preparedness aspects of the manufacturing operations are managed and controlled as described in this license.

1.1.1 SITE DESCRIPTION AND LOCATION

GNF-A is situated on a 1,664-acre tract of land, located on N.C. Highway 133 and approximately six miles north of the City of Wilmington, North Carolina in New Hanover County (refer to Figures 1.1 and 1.2). New Hanover County is situated in the southern coastal plains section of southeastern North Carolina, with the Atlantic Ocean on the east and the Cape Fear River on the west. The Atlantic Ocean lies approximately 10 miles east and 26.4 miles south of GNF-A. The surrounding terrain is low-lying, with an average elevation of less than 40 feet above mean sea level.

Castle Hayne, the nearest community, is approximately three miles north of GNF-A. The region around the site is lightly settled with large areas of heavily timbered tracts of land. Farms, single-family dwellings and light commercial activities are located along N.C. Highway133. The Wilmington airport is located approximately 3.5 miles southeast of the site.

1.1.2 FACILITY DESCRIPTION

The location and arrangement of buildings at GNF-A, and their relative distance from the site boundary are shown in Figure 1.3. Located on the property are the following major facilities: (1) the Aviation

FIGURE 1.1
PLANT SITE - STATE AND COUNTY LOCATIONS

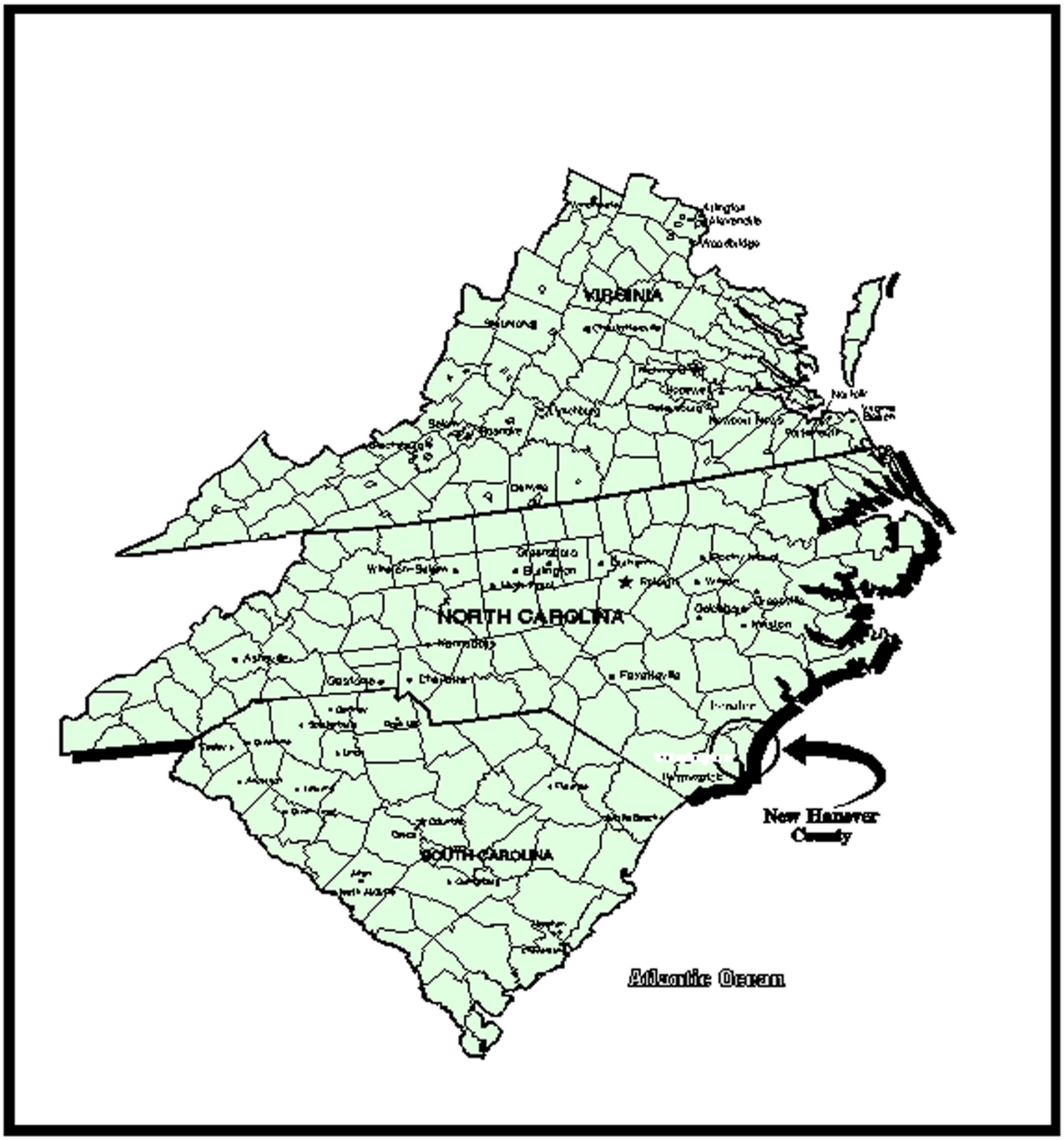


FIGURE 1.2
NEW HANOVER COUNTY AND ADJACENT COUNTIES

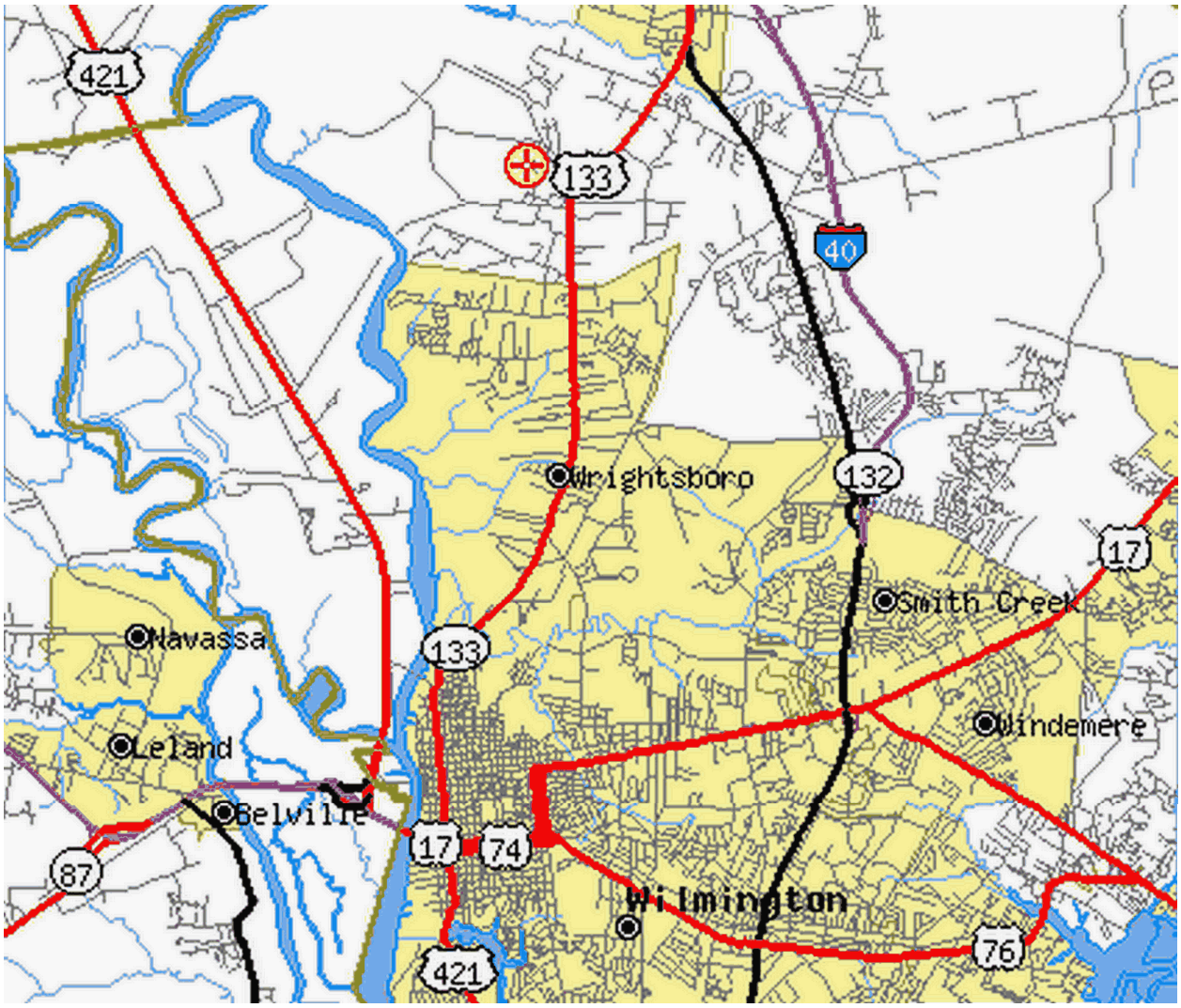


FIGURE 1.3
GLOBAL NUCLEAR FUEL – AMERICAS, L.L.C. (GNF-A)
SITE PLAN

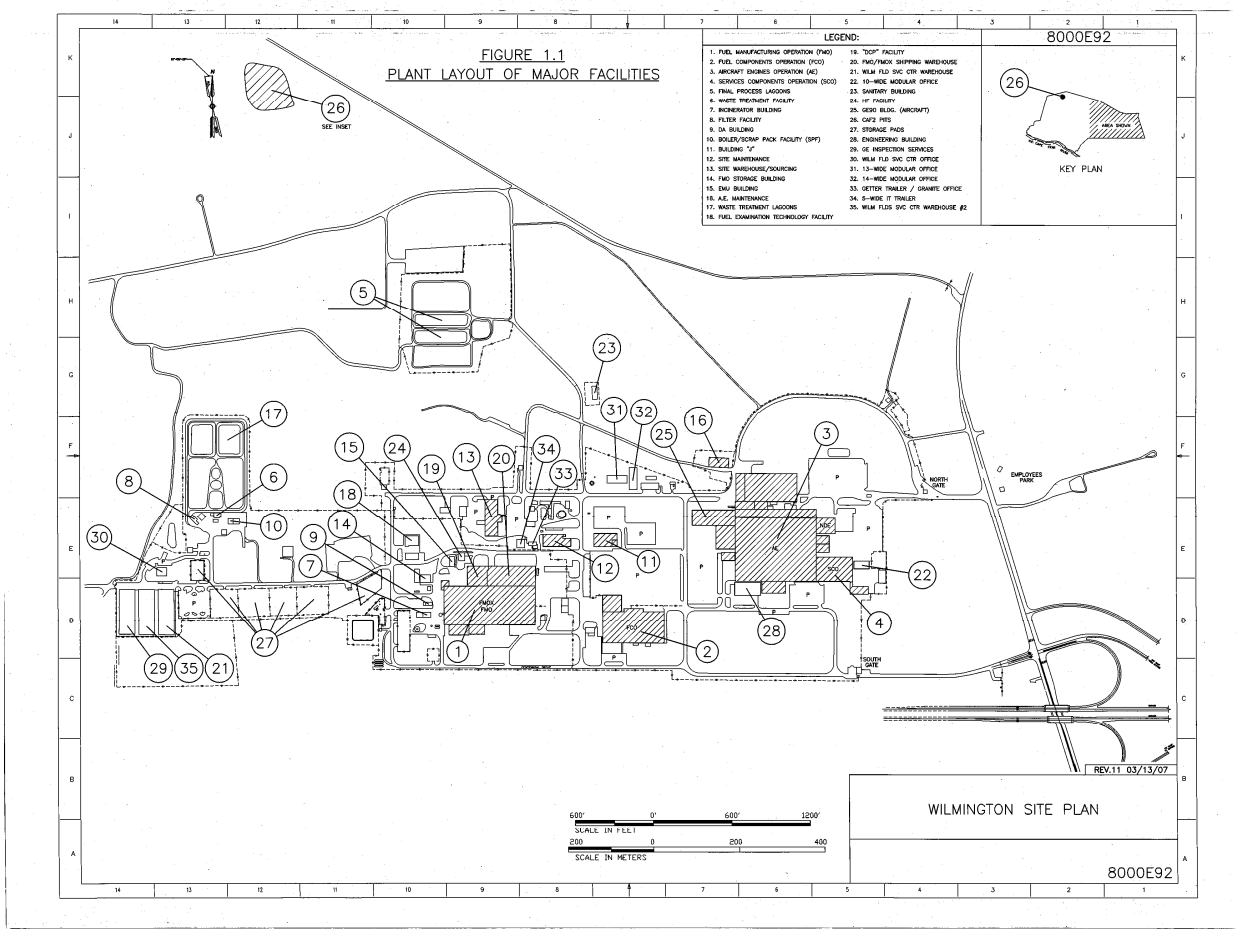


FIGURE 1.3 (Continued)

GLOBAL NUCLEAR FUEL – AMERICAS, L.L.C. (GNF-A)
SITE PLAN LEGEND

LEGEND:

1. FUEL MANUFACTURING OPERATION (FMO)
2. FUEL COMPONENTS OPERATION (FCO)
3. AVIATION OPERATION
4. SERVICES COMPONENTS OPERATION (SCO)
5. FINAL PROCESS LAGOONS
6. WASTE TREATMENT FACILITY
7. INCINERATOR BUILDING
8. FILTER FACILITY
9. DA BUILDING
10. BOILER/SCRAP PACK FACILITY (SPF)
11. BUILDING "J"
12. SITE MAINTENANCE
13. SITE WAREHOUSE/SOURCING
14. FMO STORAGE BUILDING
15. FUEL GROWTH PROJECTS BLDG
16. A.E. MAINTENANCE
17. WASTE TREATMENT LAGOONS
18. FUEL EXAMINATION TECHNOLOGY FACILITY
19. "DCP" FACILITY
20. FMO/FMOX SHIPPING WAREHOUSE
21. WILM FLD SVC CTR WAREHOUSE
22. 10–WIDE MODULAR OFFICE
23. SANITARY BUILDING
24. HF FACILITY
25. GE90 BLDG. (AVIATION)
26. CAF2 PITS
27. STORAGE PADS
28. ENGINEERING BUILDING
29. GE INSPECTION SERVICES
30. WILM FLD SVC CTR OFFICE
31. 13–WIDE MODULAR OFFICE
32. 14–WIDE MODULAR OFFICE
33. GETTER TRAILER / GRANITE OFFICE
34. 5–WIDE IT TRAILER
35. WILM FLDS SVC CTR WAREHOUSE #2

facility which is not involved in the nuclear fuel manufacturing operation, (2) The GE Services Components Operation (SCO) facility where non-radioactive reactor components are manufactured, (3) the Fuel Components Operation (FCO) facility where non-radioactive components for reactor fuel assemblies are manufactured, and (4) the fuels complex containing the fuel manufacturing facility. The fuels complex, which includes the Fuel Manufacturing Operation and Dry Conversion Process (FMO/FMOX & DCP) buildings and supporting facilities, is enclosed by a fence with restricted access. This complex is called the Controlled Access Area (CAA). Additionally, inside the CAA is the Wilmington Field Services Center (WFSC) operation which cleans and refurbishes contaminated and non-contaminated equipment used at reactor sites. This area is regulated by a North Carolina Division of Radiation Protection license.

1.1.3 FACILITY RESISTANCE TO ENVIRONMENTAL EVENTS

In the coastal area of North Carolina where GNF-A is located, the greatest severe weather threat in this area is due to high winds from hurricanes and possible tornadoes. Facility construction meets or exceeds local codes (in effect when designed and constructed) for strength and, in the case of hurricanes, advance notice provides an opportunity for further mitigating actions. Since high winds could impact electrical power, key safety systems are protected with adequate back-up power supplies or fail safe features. Earthquakes are not considered a major threat because this section of the southern Atlantic Seaboard is an area of relatively low seismic activity.

The Fuel Manufacturing Operation building in which radioactive materials are processed and stored, is designed to provide for containment of material under adverse environmental conditions such as fire, wind, flooding or earthquake to the limits of the building code. The roof construction meets Factory Mutual requirements for fire hazard and wind resistance.

Detailed information regarding the facility resistance to the effects of potential credible accident sequences is contained in the facilities ISA Summaries submitted in accordance with 10 CFR 70.65(b).

1.1.4 PROCESS DESCRIPTION

The product manufacturing operations authorized by this license consist of receiving low-enriched, (less than or equal to 5.0 weight percent U-235), uranium hexafluoride; converting the uranium hexafluoride to uranium dioxide powder; and processing the uranium dioxide through pelletizing steps, fuel rod loading and sealing, fuel assembly, storage and product shipment.

The manufacturing operations are served by support systems such as waste disposal, laboratory, and manufacturing technology development, which are also described in this license.

1.2 INSTITUTIONAL INFORMATION

GNF-A's NRC license number is SNM-1097 (Docket #70-1113).

1.2.1 IDENTITY AND ADDRESS

This application for license renewal is filed by the Global Nuclear Fuel - Americas, L.L.C., a wholly-owned subsidiary of Global Nuclear Fuel Holding Co., L.L.C., in which General Electric holds a majority ownership interest. Global Nuclear Fuel – Americas, L.L.C. is headquartered in Wilmington, North Carolina.

The full address is as follows: Global Nuclear Fuel - Americas, L.L.C., mailing address P.O. Box 780, Wilmington, NC 28402 or location address 3901 Castle Hayne Road, Wilmington, NC 28401.

1.2.2 TYPE, QUANTITY, AND FORM OF LICENSED MATERIAL

Uranium normally will be used at GNF-A in the Controlled Access Area (CAA) only. Conversion and fabrication is conducted within the fuel manufacturing building (FMO/FMOX & DCP). Small quantities (i.e., less than one safe batch of uranium in a non-dispersible form) may be temporarily moved to other buildings or site locations outside of the CAA for special tests under special authorizations and controls.

The following types, maximum quantities, and forms of special nuclear materials are authorized:

1) 50,000 kilograms of U-235 contained in uranium enriched to a maximum enrichment of less than or equal to 5%, in any chemical or physical form,
2) 350 grams of U-235 at nominal enrichments >5% to <10% in any form for use in laboratory and process development operations,
3) 350 grams of U-235 in any form and at any enrichment for use in measurement and detection instruments, check sources and instrument response standards,
4) Plutonium - 1 milligram in samples for analytical purposes, 1 milligram as standards for checking the alpha radiation response of radiation detection instrumentation, and in nuclear fuel rods at a level of less than 1×10^{-6} gram of plutonium per gram of U^{235}, and
5) 50 milligrams U-233 for analytical purposes.

1.2.3 ACTIVITY

GNF-A complies with applicable parts of Title 10, Code of Federal Regulations, unless specifically amended or exempted by NRC staff.

Authorized activities at GNF-A include:

1.2.3.1 Product Processing Operations

- UF_6 Conversion - Conversion of uranium hexafluoride to uranium oxides by the ADU process and the Dry Conversion Process.
- Fuel Manufacture - Fabrication of nuclear reactor fuel (powder, pellets, or assemblies) containing uranium.
- Scrap Recovery - Reprocessing of uranium bearing material from GNF-A fuel fabrication activities and from other sources with nuclear safety characteristics not significantly different from GNF-A's in-process materials.
- Waste Recovery - Recovery of uranium from wet and dry material stored in on-site pits and basins. The recovered uranium will be returned to the fuel processing facility.

1.2.3.2 Process Technology Operations

- Development and fabrication of reactor fuel, fuel elements and fuel assemblies of advanced design in small amounts.
- Development of scrap recovery processes.
- Determination of interaction between fuel additives and fuel materials.
- Chemical analysis and material testing, including physical and chemical testing and analysis, metallurgical examination and radiography of uranium compounds, alloys and mixtures.
- Instrument research and calibration, including development, calibration and functional testing of nuclear instrumentation and measuring devices.
- Conversion of UF_6 to UO_2 and other intermediate compounds using chemical and dry processes.
- Other process technology development activities related to, but not limited by, the above.

1.2.3.3 Laboratory Scale Laser Enrichment Research and Development

- Operation of a laboratory scale research and development activity related to the enrichment of uranium using laser energy involving the extension of investigative findings and theories of a technical nature based on original work by Silex Limited.
- Experimental production of slightly enriched uranium from natural uranium feed material to test models, devices, equipment, materials and processes for the sole purpose of developing a practical application of the technology.
- Withdrawal of samples from the test-loop for the purpose of evaluating process performance.
- Other enrichment technology development activities related to, but not limited by the above.

1.2.3.4 Laboratory Operations

Chemical, physical or metallurgical analysis and testing of uranium compounds and mixtures, including but not limited to, preparation of laboratory standards.

1.2.3.5 General Services Operations

- Storage of unirradiated fuel assemblies, uranium compounds and mixtures in areas arranged specifically for maintenance of criticality and radiological safety.
- Design, fabrication and testing of uranium prototype processing equipment.
- Maintenance and repair of uranium processing equipment and auxiliary systems.
- Storage and nondestructive testing of fuel rods containing trace amounts of plutonium as authorized in the license.

1.2.3.6 Waste Treatment and Disposal

- Treatment, storage and disposal and/or shipment of liquid and solid wastes whose discharges are regulated.
- Decontamination of non-combustible contaminated wastes to reduce uranium contamination levels, and subsequent shipment of such low-level radioactive wastes to licensed burial sites for disposal or as authorized by the NRC.
- Treatment or disposal of combustible waste and scrap material by incineration pursuant to 10 CFR 20.2002 and 10 CFR 20.2004.

1.2.3.7 Off-site Activities

Testing, demonstration, non-destructive modification and storage of materials and devices containing uranium, provided that such materials and devices shall be under GNF-A's control at all times.

1.2.3.8 Possession of Classified Matter

Authorization to use, possess, store, reproduce, transmit, handle and transport classified matter under conditions specified in the standard Practice and Procedures Plan (SPPP) submitted and approved in accordance with 10 CFR Part 95 requirements. The licensee is to maintain the SPPP in accordance with 10 CFR 95.19.

1.3 SPECIAL AUTHORIZATIONS AND EXEMPTIONS

1.3.1 AUTHORIZATIONS TO MAKE CHANGES TO LICENSE COMMITMENTS

1.3.1.1 Changes Requiring Prior NRC Approval

GNF-A will not make changes to the site, structures, processes, systems, equipment, components, computer programs or personnel activities unless those changes are authorized in accordance with10 CFR 70.72 and/or this license. Requests for amendment of the license will be made in accordance with 10 CFR 70.34 and 10 CFR 70.65.

1.3.1.2 Changes Not Requiring Prior NRC Approval

GNF-A is authorized to make changes to the site, structures, processes, systems, equipment, components, computer programs and activities of personnel without prior NRC approval provided the changes do not:

- Create new types of accident sequences that, unless prevented or mitigated, would exceed the performance requirements of 10 CFR 70.61, and that have not previously been described in the ISA Summary
- Use new processes, technologies or control systems for which GNF-A has no prior experience
- Remove, without at least an equivalent replacement of the safety function, an item relied on for safety (IROFS) that is listed in the ISA Summary
- Alter any IROFS, listed in the ISA Summary, that is the sole IROFS for an accident sequence that would be expected to exceed the performance requirements of 10 CFR 70.61
- Violate any other NRC regulation, license condition or order.

1.3.1.3 On-Site Documentation

GNF-A shall maintain records of changes to its facilities carried out under 10 CFR 70.72. These records will contain an evaluation that provides the basis for the determination that the changes do not require prior NRC approval.

These records will be maintained until the termination of the license.

For changes covered by 10 CFR 70.72, the affected on-site documentation is updated promptly not to exceed 3-months from the approval of the change.

1.3.1.4 Annual Reporting of Changes

For changes not requiring pre-approval by the NRC, GNF-A shall submit annually, within 30-days after the end of the calendar year during which the change occurred, a brief summary of all the changes to the records required by 10 CFR 70.62 (a) (2).

For all changes that affect the ISA Summary, GNF-A shall submit annually, within 30-days after the end of the calendar year during which the change occurred, revised ISA Summary pages.

1.3.2 AUTHORIZED GUIDELINES FOR CONTAMINATION-FREE ARTICLES

Authorization to use the guidelines, contamination and exposure rate limits specified at the end of this Section, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material," US NRC, April 1993 for decontamination and survey of surfaces or premises and equipment prior to abandonment or release for unrestricted use.

1.3.3 AUTHORIZED TRANSFER OF CONTAMINATION-FREE LIQUIDS

1.3.3.1 Transfer of Hydrofluoric Acid (HF) for Testing

Authorization to transfer test quantities of HF to potential buyers/customers or laboratories for the purpose of analyzing, examination or evaluation, without continuing NRC controls as described in a letter to the NRC dated February 27, 1996.

Test quantities may not contain more than 3 PPM uranium with an enrichment not to exceed 5% U-235.

The recipients will be advised that this material is not a nuclear hazard, but will be advised that the material should be handled carefully and in such a manner so as not to be consumed by humans nor used in products used on or in the body or in the food chain.

1.3.3.2 Transfer of Hydrofluoric Acid as Product

Authorization, pursuant to 10 CFR 70.42(b)(3), to transfer liquid hydrofluoric acid to any commercial chemical company/supplier without either company possessing an NRC or Agreement State license for special nuclear material, provided that the concentration of uranium does not exceed three parts per million by weight of the liquid and the enrichment is less than or equal to 5 weight percent U-235.

The hydrofluoric acid is transferred and used in such a manner that the minute quantity of uranium does not enter into any food, beverage, cosmetic, drug or other commodity designated for ingestion or inhalation by, or application to, a human being such that the uranium concentration in these items would exceed that which naturally exists. Additionally, the acid is used in a process which will not release the low levels of radioactivity to the atmosphere as airborne material and whose residues will remain in a wastewater or other treatment system.

Prior to shipment, each transfer is sampled and measured to assure that the concentration does not exceed three parts per million of uranium (3 ppmU).

GNF-A shall maintain records under this condition of license including, as a minimum, the date, uranium concentration and quantity of hydrofluoric acid transferred.

1.3.3.3 Transfer of Nitrate-Bearing Liquids

Authorization to transfer nitrate-bearing liquids, provided that the uranium concentration does not exceed a 30-day average of 5 parts per million by weight of the liquids and the enrichment is less than or equal to 5 weight percent U-235 by transport to an off-site liquid treatment system located at International Paper, Riegelwood, North Carolina (or similar commercial paper operation), in which decomposition of the nitrates will occur and from which the denitrified liquids will be discharged in the effluent from the system.

Environmental samples will be taken periodically to monitor effluent releases.

1.3.4 AUTHORIZATION TO TRANSFER TEST QUANTITIES OF CALCIUM FLUORIDE

Authorization to transfer test quantities of calcium fluoride (CaF_2) to potential buyers for the purpose of their examination and evaluation as described in a letter to the NRC dated September 24, 1992.

Test quantities may not contain more than 30 pCi per gram on a dry weight basis and are limited to 1 gram U-235 at each off-site location.

Test activities and end uses of the material will be limited to those that do not allow chemical separation of the uranium or entry of the product into the food chain.

1.3.5 AUTHORIZATION TO TRANSFER CALCIUM FLUORIDE (CAF_2) TO VENDORS FOR BENEFICIAL REUSE

Authorization to transfer quantities of industrial waste treatment products (primarily CaF_2) to commercial firms, for the purpose of briquette manufacturing and use as a steel flux forming material in the production of steel as described in a letter to the NRC dated December 20, 1989.

Measurements are made using a sample plan to provide at a 95% confidence level that the population mean for each shipment is less than 30pCi of uranium per gram of material on a dry weight basis.

Activities and end use of the material will be limited to those that do not allow chemical separation of the uranium or entry of the product into the food chain.

1.3.6 AUTHORIZATION TO DISPOSE OF INDUSTRIAL WASTE TREATMENT PRODUCTS

Notwithstanding any requirements for state or local government agency disposal permits, GNF-A is authorized to dispose of industrial waste treatment products without continuing NRC controls provided that either of the two following conditions are met:

1.3.6.1 Free-standing liquid shall be removed prior to shipment.

The uranium concentration in the material shipped for disposal shall not exceed 30 pCi per gram after free-standing liquid has been removed.

The licensee shall possess authorization from appropriate state officials prior to disposing of the waste material. The authorization shall be available for inspection at GNF-A.

1.3.6.2 The uranium concentration in the material shipped for disposal only at approved facilities such as Pinewood, South Carolina (licensed by the State of South Carolina), shall not exceed 250 pCi per gram of uranium activity, of which no more than 100 pCi per gram shall be soluble.

1.3.7 AUTHORIZATION TO STORE SANITARY SLUDGE PENDING FINAL DISPOSAL

Dried sanitary sludge is collected and disposed of at approved offsite facilities in

accordance with Section 1.3.6. Authorization to store treated sanitary sludge containing trace amounts of uranium in the sanitary sludge land application area pending final disposal.

1.3.8 AUTHORIZATION FOR THE USE OF MATERIALS AT OFF-SITE LOCATIONS

1.3.8.1 Authorization to store at nuclear reactor sites, uranium fully packaged for transport in any NRC approved package, in accordance with the conditions of a license authorizing delivery of such containers to a carrier for NRC approved transport, at locations in the United States providing such locations minimize the severity of potential accident conditions to be no greater than those in the design bases for the containers during transportation.

Provisions for compliance with applicable 10 CFR 73 requirements are described in the NRC-approved GNF-A's Physical Security Plan as currently revised in accordance with regulatory provisions.

Storage at nuclear reactor sites is subject to the financial protection and indemnity provision of 10 CFR 140. Storage of the fuel is under the direct supervision of a member of the GNF-A or GE Nuclear staff. This person shall comply with applicable reactor license and procedural requirements as directed by the reactor site representative.

1.3.8.2 Authorization to transfer, possess, use and store unirradiated reactor fuel of GNF-A's manufacture or procured to GNF-A's specification at nuclear reactor sites, for purposes of inspection, fuel bundle disassembly and assembly, including fuel rod replacement, provided that the following conditions are met:

- A valid NRC license has been issued to the reactor licensee, which authorizes receipt, possession and storage of the fuel at the reactor site. GNF-A possesses the fuel only within the indemnified location.
- For dry fuel reconstitution, not more than 99 (9x9 or 10x10 lattices) or 88 (8x8 lattices) unassembled fuel rods may be possessed by GNF-A at any one reactor site at any one time, except when the fuel has been packaged for transport.

- For underwater fuel reconstitution, not more than one fuel assembly plus unassembled fuel rods so that the total number of rods, including the assembly, possessed by GNF-A at any one reactor site at any one time does not exceed 99 (9x9 or 10x10 lattices) or 88 (8x8 lattices), except when the fuel has been packaged for transport or as described in Section 1.3.8.2.
- Operations involving the fuel are conducted by or under the direct supervision of a GNF-A authorized person who shall be responsible for work on the fuel element assembly. The person shall comply with applicable reactor license and procedure requirements as directed by reactor site representatives, including appropriate actions that are to be taken in the event of emergencies at the site.
- Loose rods are stored in RA-series inner metal containers.
- Fuel is handled in accordance with pertinent provisions of the reactor license, and also in accordance with applicable GNF-A or GE Nuclear procedures which are jointly verified by GE and the reactor licensee.
- Records of the operation, including the procedures used, are maintained at GNF-A.

1.3.9 AUTHORIZATION TO USE A DILUTION FACTOR FOR AIRBORNE EFFLUENTS

Pursuant to 10 CFR 20.1302, GNF-A is authorized to utilize a dilution factor of 100 to the measured stack discharges for the purpose of evaluating the airborne radioactivity at the closest site boundary.

This conservative dilution factor is derived using standard diffusion models and conservative assumptions regarding physical and atmospheric characteristics of the site. Records of the derivation of this factor are maintained on site for inspection.

1.3.10 AUTHORIZATION FOR WORKPLACE AIR SAMPLING ADJUSTMENTS

Authorization to adjust Derived Air Concentration (DAC) limits and Annual Limit of Intake (ALI) values in process areas to reflect chemical and physical characteristics of the airborne uranium.

1.3.11 EXEMPTION TO CRITICALITY MONITORING SYSTEM REQUIREMENTS

Authorization that it is not necessary to maintain the criticality accident monitoring system requirements of 10 CFR 70.24 when it is demonstrated that a credible criticality risk does not exist for each area in which there is not more than:

1.3.11.1 A quantity of finished reactor fuel rods equal to or less than 45% of a minimum critical number under conditions in which double batching is credible, or equal to or less than 75% of a minimum critical number under conditions in which double batching is not credible, or

1.3.11.2 The quantity of uranium authorized for delivery to a carrier when fully packaged as for transport according to a valid NRC authorization for such packages without limit on the number of such packages, provided storage locations preclude mechanical damage and flooding, or

1.3.12 EXEMPTION TO POSTING REQUIREMENTS

Authorization to post areas within the Controlled Access Area in which radioactive materials are processed, used, or stored, with a sign stating "Every container in this area may contain radioactive material" in lieu of the labeling requirements of 10 CFR 20.1904.

1.3.13 EXEMPTION TO EXTREMITY DOSE DETERMINATION REQUIREMENTS

Authorization to use a skin thickness of 38 milligrams/cm^2 in the assessment of worker fingertip doses from uranium and for determining compliance to NRC extremity dose limits.

1.3.14 UNRESTRICTED RELEASE OF THE NORTHWEST CAF_2 STORAGE AREA

Pursuant to the CaF_2 Survey and Release Plan dated 3/27/96 as amended, and the Final Status Survey Report for the Northwest CaF_2 Storage Area, dated October 1999, the area described as the Northwest CaF_2 Storage Area has been adequately remediated, and therefore meets the unrestricted release criteria.

1.3.15 AUTHORIZATION TO USE ICRP 68

DAC and ALI values based on dose coefficients published in ICRP Publication No. 68 will be used in lieu of the values in Appendix B of 10 CFR Part 20 in accordance with internal procedures.

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

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.

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 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

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

CHAPTER 2.0
ORGANIZATION AND ADMINISTRATION

2.1 POLICY

The GNF-A policy is to maintain a safe work place for its employees, to protect the environment, and to assure operational compliance within the terms and conditions of special nuclear material licenses and applicable NRC regulations.

2.2 ORGANIZATIONAL RESPONSIBILITIES AND AUTHORITY

2.2.1 KEY POSITIONS WITH RESPONSIBILITIES IMPORTANT TO SAFETY (FIGURE 2.1)

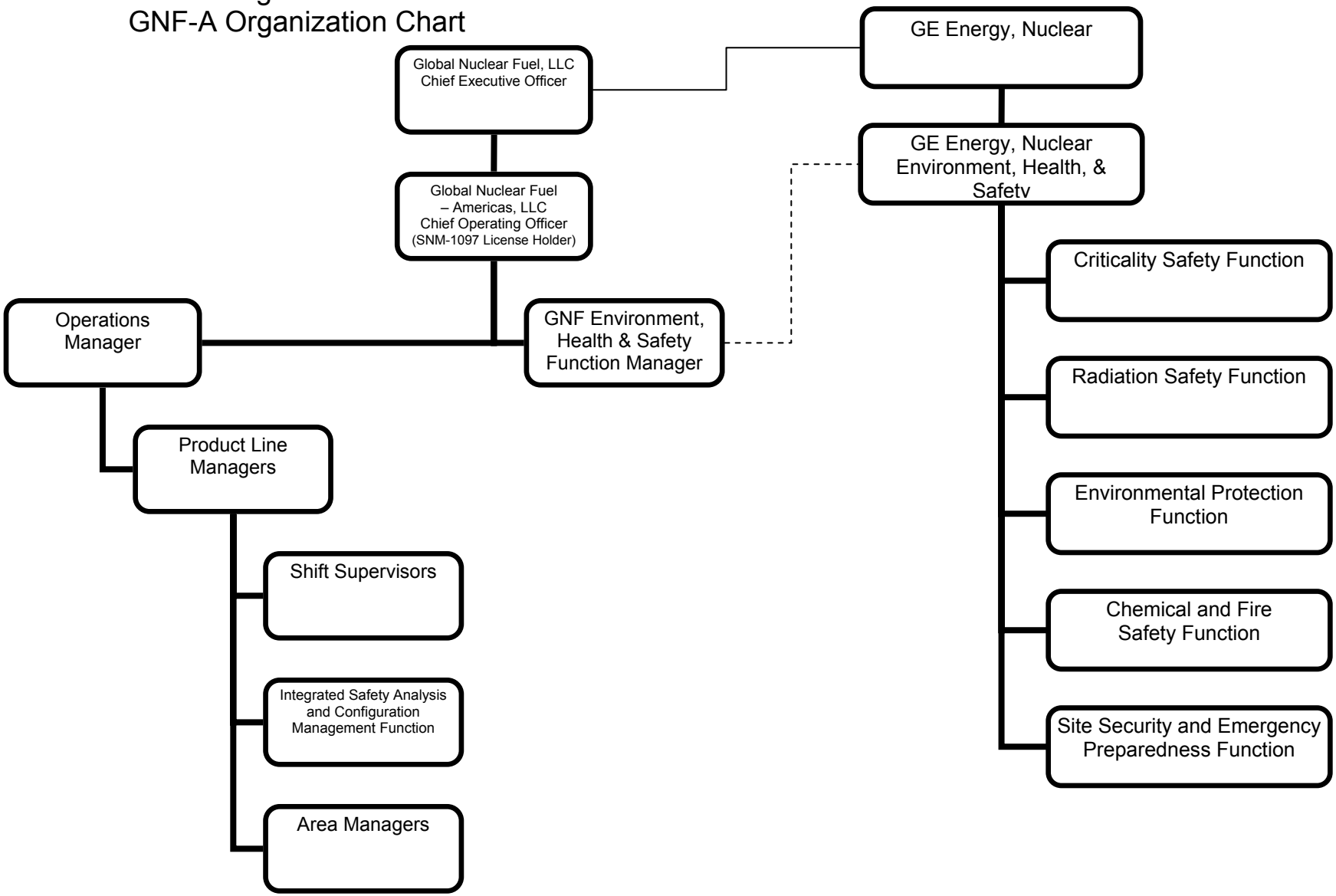
Responsibilities, authorities, and interrelationships among the GNF-A organizational functions with responsibilities important to safety are specified in approved position descriptions and in documented and approved practices.

2.2.1.1 GNF-A's Facility Manager

The GNF-A Facility Manager is the individual who has overall responsibility for safety and activities conducted at GNF-A. The Facility Manager directs operations by procedure, or through other management personnel. The activities of the Facility Manager are performed in accordance with GNF-A's policies, procedures, and management directives. The Facility Manager provides for safety and control of operations and protection of the environment by delegating and assigning responsibility to qualified Area Managers.

The minimum qualifications of a Facility Manager is a BS or BA degree and two years experience in manufacturing operations. The Facility Manager is knowledgeable of the safety program concepts as they apply to the overall safety of a nuclear facility, and has the authority to enforce the shutdown of any process or facility. The Facility Manager must approve restart of an operation they request be shutdown.

Figure 2.1
GNF-A Organization Chart



2.2.1.2 Area Manager

The Area Manager is the designated individual who is responsible for ensuring that activities necessary for safe operations and protection of the environment are conducted properly within their designated area of the facility in which uranium materials are processed, handled or stored. Designated Area Manager responsibilities include:

- Assure safe operation, maintenance and control of activities
- Assure safety of the environs as influenced by operations
- Assure performance of integrated safety analyses for the assigned facility area, as required
- Assure application of assurance elements to safety controls, as appropriate
- Assure configuration control for safety controls for the assigned facility area, as required
- Use approved written operating procedures which incorporate safety controls and limits
- Provide adequate operator training

The minimum qualifications of an Area Manager is a BS or BA degree in a technical field, and two years of experience in manufacturing operations, one of which is in nuclear fuel manufacturing; or a high school diploma with five years of manufacturing experience, two of which are in nuclear fuel manufacturing.

Area Managers shall be knowledgeable of the safety program procedures (including chemical, radiological, criticality, fire, environmental and industrial safety) and shall have experience in the application of the program controls and requirements, as they relate to their areas of responsibility. The assignment of individuals to the position of Area Manager is approved by the Facility Manager, and the listing of Area Managers by area of responsibility is maintained current at the facility.

2.2.1.3 Integrated Safety Analysis and Configuration Management Function

The integrated safety analysis and configuration management function is administratively part of the fuel production operations at GNF-A. Designated responsibilities include:

- Establish and maintain the integrated safety analysis program and identify items relied on for safety (IROFS)
- Establish and maintain the assurance program for safety controls
- Provide advice and counsel to Area Managers on matters of the integrated safety analysis program
- Establish and maintain the configuration control system for fuel manufacturing equipment and safety controls, and related record retention
- Establish and maintain the operating procedure systems

Minimum qualification requirements for the manager of the integrated safety analysis and configuration management function are a BS or BA degree in science or engineering and two years experience in related manufacturing assignments; or a high school diploma with eight years of manufacturing experience. The manager of the integrated safety analysis and configuration management function shall have experience in the understanding and management of the assigned programs.

2.2.1.4 Shift Supervisor

Shift supervisors are provided as the interface between management and facility operators. Shift supervisor responsibilities include:

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

The minimum qualifications for shift supervisor is one year supervisory experience in a technical field. Shift supervisors shall be knowledgeable of the applicable safety program procedures (including chemical, radiological, criticality, fire, environmental and industrial safety).

2.2.1.5 Criticality Safety Function

The criticality safety function is administratively independent of production responsibilities and has the authority to shutdown potentially unsafe operations. This function must approve restart of an operation they request be shutdown.

Designated responsibilities include:

- Establish the criticality safety program including design criteria, procedures and training
- Provide criticality safety support for nuclear operations including integrated safety analyses and configuration control
- Assess normal and credible abnormal conditions
- Determine criticality safety limits for controlled parameters
- Perform methods development and validation to support criticality safety analyses
- Perform neutronics calculations, write criticality safety analyses and approve proposed changes in process conditions or equipment involving fissionable material
- Specify criticality safety control requirements and functionality
- Provide advice and counsel to Area Managers on criticality safety control measures, including review and approval of operating procedures
- Support emergency response planning and events
- Assess the effectiveness of the criticality safety program through audit programs

The criticality safety function manager shall hold a BS or BA degree in science or engineering, have at least four years experience in assignments involving regulatory activities, and have experience in the understanding, application and direction of nuclear criticality safety programs.

Minimum qualifications for a senior engineer within the criticality safety function are a BS or BA degree in science or engineering with at least three years of nuclear industry experience in criticality safety. A senior engineer shall have experience in the assigned safety function, and has authority and responsibility to conduct activities assigned to the criticality safety function.

Minimum qualifications for an engineer within the criticality safety function are a BS/BA degree in science or engineering. An engineer shall have experience in the assigned safety function, and has authority and responsibility to conduct activities assigned to the criticality safety function, with the exception of independent verification of criticality safety analyses.

2.2.1.6 Radiation Safety Function

The radiation safety function is administratively independent of production responsibilities and has the authority to shutdown potentially unsafe operations. This function must approve restart of an operation they request be shutdown.

Designated responsibilities include:

- Establish the radiation protection and radiation monitoring programs
- Establish the radiation protection design criteria, procedures and training programs to control contamination and exposure to individuals
- Evaluate radiation exposures of employees and visitors, and ensure the maintenance of related records
- Conduct radiation and contamination monitoring and control programs
- Evaluate the integrity and reliability of radiation detection instruments
- Provide radiation safety support for integrated safety analyses and configuration control
- Provide analysis and approval of proposed changes in process conditions and process equipment involving radiological safety
- Provide advice and counsel to Area Managers on matters of radiation safety
- Support emergency response planning and events
- Assess the effectiveness of the radiation safety program through audit programs

The radiation safety function manager shall hold a BS or BA degree in science or engineering, have at least two years experience in assignments that include responsibility for radiation safety, and have experience in the understanding, application and direction of radiation safety programs.

Minimum qualifications for a senior member of the radiation safety function are a BS or BA degree in science or engineering with at least two years of nuclear industry experience in the assigned function. Alternate minimum experience qualification for a senior member of the radiation safety function is professional certification in health physics. A senior member shall have experience in the assigned safety function, and has authority and responsibility to conduct activities assigned to the radiation safety function.

2.2.1.7 Environmental Protection Function

The environmental protection function is administratively independent of production responsibilities and has the authority to shutdown operations with potentially uncontrolled environmental conditions. This function must approve restart of an operation they request be shutdown.

Designated responsibilities include:

- Identify environmental protection requirements from federal, state and local regulations which govern the GNF-A operation
- Establish systems and methods to measure and document adherence to regulatory environmental protection requirements and license conditions
- Provide advice and counsel to Area Managers
- Evaluate and approve new, existing or revised equipment, processes and procedures involving environmental protection activities
- Provide environmental protection support for integrated safety analyses and configuration control
- Assure proper federal and state permits, licenses and registrations for non-radiological discharges from the facilities

Minimum qualifications for the manager of the environmental protection function are a BS or BA degree in science or engineering and two years of experience in assignments involving regulatory activities or equivalent.

2.2.1.8 Chemical and Fire Safety Function

The chemical and fire safety function is administratively independent of the production responsibilities and has the authority to shutdown operations with potentially hazardous health and safety conditions. This function must approve restart of an operation they request be shutdown.

Designated responsibilities include:

- Identify fire protection requirements from federal, state, and local regulations which govern the GNF-A operations
- Develop practices regarding non-radiological chemical safety affecting nuclear activities
- Provide advice and counsel to Area Managers on matters of chemical and fire safety
- Provide consultation and review of new, existing or revised equipment, processes and procedures regarding chemical safety and fire protection
- Provide chemical and fire safety support for integrated safety analyses and configuration control

Minimum qualifications of the manager of the chemical and fire safety function are a BS or BA degree in science or engineering and two years of experience in related assignments.

2.2.1.9 Site Security and Emergency Preparedness Function

The site security and emergency preparedness function is administratively independent of the production responsibilities. Designated responsibilities include:

- Provide physical security for the site
- Establish and maintain the emergency preparedness program, including training and program evaluations
- Provide advice and counsel to Area Managers on matters of physical security and emergency preparedness
- Maintain agreements and preparedness with off-site emergency support groups

Minimum qualifications are a BS or BA degree in science or engineering, one year of experience in related assignments, or a high school diploma with eight years of experience in related assignments.

2.2.1.10 Environment, Health & Safety (EHS) Function

The EHS function is administratively independent of production responsibilities but has the authority to enforce the shutdown of any process or facility in the event that controls for any aspect of safety are not assured. This function has designated overall responsibility to establish the radiation safety, criticality safety, environmental protection, chemical safety, fire protection and emergency preparedness programs to ensure compliance with federal, state and local regulations and laws governing operation of a nuclear manufacturing facility. These programs are designed to ensure the health and safety of employees and the public as well as protection of the environment. The managers of the criticality safety, radiation safety, environmental protection, chemical and fire safety, and site security and emergency preparedness functions report to the EHS function manager.

The manager of the EHS function must hold a BS or BA degree in science or engineering and have five years of management experience in assignments involving regulatory activities. The manager of the EHS function must have appropriate understanding of health physics, nuclear criticality safety, environmental protection, and chemical and fire safety programs.

2.2.2 MANAGEMENT CONTROLS

Management controls for the conduct and maintenance of GNF-A's health, safety and environment protection programs are contained in documented plant practices described in Section 11, and approved by cognizant management. Such practices are part of a controlled document system, and appropriately span the organizational structure and major plant activities to control interrelationships, and to specify program objectives, responsibilities and requirements. Personnel are appropriately trained to the requirements of these management controls, and compliance is monitored through internal and independent audits and evaluations.

Management controls documented in practices address requirements including:

- Configuration Management

- Integrated Safety Analysis
- Radiation Safety
- Criticality Safety
- Environmental Protection
- Chemical Safety
- Fire & Explosion Safety
- Emergency Preparedness
- Quality Assurance
- Training
- Procedures
- Maintenance
- Audits
- Incident Investigation & Reporting
- Fissile Material Accountability and Control
- Worker Concerns Program
- Management Measures Necessary to Maintain Items Relied on for Safety

2.3 TRAINING AND CONTINUING ASSURANCE

Personnel training and continuing assurance is conducted as necessary to provide reasonable assurance individuals are qualified, continue to understand, and recognize the importance of safety while performing assigned activities.

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

Formal training relative to safety includes radiation and radioactive materials, risks involved in receiving low level radiation exposure in accordance with 10 CFR 19.12, basic criteria and practices for radiation protection, nuclear criticality safety principles not verbatim, but in general conformance with ANSI/ANS 8.19 and

ANSI/ANS 8.20 guidance, chemical and fire safety, maintaining radiation exposures and radioactivity in effluents As Low As Reasonably Achievable (ALARA), and emergency response.

The system established for management assurance and record retention of training and retraining is described in Chapter 11.

2.3.1 NUCLEAR SAFETY TRAINING

Training policy requires that employees complete formal nuclear safety training prior to unescorted access in the airborne radioactivity controlled area (see Chapter 11, Section 11.4.2.2).

2.3.2 OPERATOR TRAINING

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

2.4 SAFETY COMMITTEES

2.4.1 WILMINGTON SAFETY REVIEW COMMITTEE

The functions of the Wilmington Safety Review Committee include responsibility for the following:

- An annual ALARA review which considers:
  - Programs and projects undertaken by the radiation safety function and the Radiation Safety Committee
  - Performance including, but not limited to, trends in airborne concentrations of radioactivity, personnel exposures, and environmental monitoring results

    - Programs for improving the effectiveness of equipment used for effluent and exposure control
- Review of major changes in authorized plant activities which may affect nuclear or non-nuclear safety practices
- Professional advice and counsel on environmental protection, and criticality, radiation, chemical and fire safety issues affecting the nuclear activities.

The committee is responsible to the Facility Manager. Its proceedings, findings and recommendations are reported in writing to the Facility Manager and to appropriate staff level managers responsible for operations which have been reviewed by the committee. Such reports shall be retained for at least three years.

The committee holds at least three meetings each calendar year with a maximum interval of 180 days between any two consecutive meetings.

2.4.2 RADIATION SAFETY COMMITTEE

The objective of the Radiation Safety Committee is to maintain occupational radiation exposures as low as reasonably achievable (ALARA) through improvements in fuel manufacturing operations.

The committee meets monthly to maintain a continual awareness of the status of projects, performance measurement and trends, and the current radiation safety conditions of shop activities. The maximum interval between meetings does not exceed 60 days.

A written report of each Radiation Safety Committee meeting is forwarded to cognizant Area Managers and the manager of the EHS function. Records of the committee proceedings are maintained for three years.

The committee consists of managers or representatives from key manufacturing functions with activities affecting radiation safety.

CHAPTER 3.0

INTEGRATED SAFETY ANALYSIS

3.1 GNF-A SAFETY PROGRAM

GNF-A has established and maintains a safety program that demonstrates compliance with the performance requirements of 10 CFR 70.61. The safety program utilizes written and approved procedures for performing Integrated Safety Analysis (ISA) that is of appropriate detail for the complexity of each process. It applies graded management measures and assurances commensurate with the reduction of the risk attributable to that item.

3.2 PROCESS SAFETY INFORMATION

Process safety information is used in performing the ISA and identifying and understanding the hazards associated with processes.

Hazardous material information for all material used or produced in plant processes is available to employees and ISA teams in the form of Material Safety Data Sheets (MSDS's). These document physical and chemical properties including toxicity, acute exposure limits, reactivity data, corrosivity data, and thermal and chemical stability data.

Technical reports used by the ISA team typically include system overviews, process chemistry, intended inventories, safe upper and lower limits for process variables (such as temperature, pressure, flow, and composition), and define safety measures required. Criticality safety analyses assess both normal and credible abnormal conditions for plant processes, and evaluate safety consequences of process deviations are also referenced by ISA teams.

Process equipment information is maintained in accurate condition through configuration management. Examples include P&IDs, materials of construction, electrical classification, ventilation system design, items relied on for safety (IROFS) and safety systems including interlocks, detections, and suppression systems.

3.3 INTEGRATED SAFETY ANALYSIS

To analyze the identified hazards and evaluate accident sequences and process disturbances caused by process deviations or events incident to the facility and credible external events including natural phenomena, GNF-A uses hazard analysis methodologies identified in the following references: *Guidelines for Hazard Evaluation Procedures,* AIChE, and *NUREG – 1513 Integrated Safety Analysis Guidelines Document.*

The ISA is initiated by systematically breaking each process down into well-defined pieces (nodes) in which the ins, outs, and internal activity/flows can be defined, in order to allow interactions to be studied. All licensed operations are treated in this manner so that the entire facility is evaluated in a logical flow approach.

The initial phase of the hazards analysis identifies criticality, radiation, chemical, process, fire, and explosion hazards for the designated process. Hazards are defined as materials, equipment, or energy sources with the potential to cause injury or illness to humans. For this facility they typically include chemicals, radiation exposure, fissile material, and sources of fire or explosion potential. These hazards are developed from the process, procedures, operating experience, and the inventories of hazardous materials.

The hazards identified for the designated process are analyzed using process hazard analysis techniques. To perform this analysis, the accident sequences are identified, consequence and unmitigated likelihood of occurrence are established, (IROFS) that prevent and/or mitigate the accident sequences are identified, and the reliability assurances are specified.

3.4 IDENTIFICATION OF POTENTIAL ACCIDENT SEQUENCES

When analyzing accident sequences, the following are to be considered: process deviations, human errors, internal facility events, and hazardous credible external events. Where postulated sequences are considered not to be credible (see discussion and definition in Section 3.4.5 regarding credibility), justification must be documented. Where preventative actions and/or control measures are required to prevent and/or mitigate accident sequences, common mode failure and systems interaction are also evaluated.

The process used in determining the process hazard analysis technique depends on the complexity of the process being evaluated, the perceived risks associated with the

process, and the personnel doing the analysis (which includes their process knowledge, experience, and knowledge of process hazards analysis). The ISA leader selects an appropriate process hazard analysis technique giving due consideration to these factors. The following summaries of techniques commonly used at this facility does not preclude the use of other accepted process hazard analysis techniques.

3.4.1 HAZARD AND OPERABILITY ANALYSIS

The Hazard and Operability Analysis (HAZOP) technique is generally applied to complex operations. It focuses on both safety hazards and operability. It may be used both during and after the process design phase. It is applicable for both continuous and batch flow processes.

HAZOP uses the synergy of an interdisciplinary team and a systematic approach to identify hazards and operability problems resulting from deviations from the process's design intent that could lead to undesirable consequences. Typically a fixed set of guide words (i.e., no, low, high, etc.) are combined with process parameters (i.e., flow, temperature, pressure, etc.) and applied at specified points (nodes) to evaluate potential outcomes.

3.4.2 WHAT-IF/CHECKLIST ANALYSIS

This is a hybrid approach that combines the best features of What-If creative brainstorming with the discipline of Checklist Analysis. It depends on an experienced team. It is very effective for the simpler, straightforward processes where a high degree of resolution is not required (e.g. powder blending, pellet pressing, grinding, etc.). It can be used at every stage in the life of the process.

3.4.3 WHAT-IF ANALYSIS

The What-If Analysis technique is a brainstorming approach that builds on the synergy of an experienced group. While inherently not as structured as some techniques (e.g. HAZOP) it is flexible and effective for the more simple processes (e.g. mechanical steps of assembling a fuel bundle, scanning, etc.). It can be used at every stage in the life of the process; however, analysis reliability is increased by experience.

3.4.4 CHECKLIST ANALYSIS

Checklist Analysis is a simple and effective technique for verifying the status of a system. It is highly disciplined and effective for verifying compliance (e.g., lockout-tagout, fall prevention, rod storage, etc.). It can be used at any stage of a process's lifetime but is dependent upon the experience and knowledge of those preparing the checklist.

3.4.5 ACCIDENT SEQUENCE CREDIBILITY

In considering accident sequences at this facility it is necessary to determine which ones will be considered not credible and which ones are credible. Accident sequences that do not meet the definition of "not credible" are therefore considered "credible" and treated in accord with 10CFR70.61. In this regard, for an accident sequence to be considered not credible, it must have the qualities associated with one or more of the following criteria:

a) Represent an external event for which the frequency of occurrence can conservatively be estimated less than once in a million years.

b) Represent process deviations for which there is a sound argument, based on physical laws or sound engineering/technical data that the deviations are not possible, or are extremely unlikely. The validity of the argument must be independent of any feature, design, or materials controlled by a system of IROFS, or of management measures.

Example(s) Process Deviations Considered "Not Credible"

(i) *Most Reactive Chemical Form* - Occurrence in any process or operation of any form of uranium more reactive than UO_2 (e.g. U Metal) - i.e. with a material density greater than the theoretical UO_2 compound density of 10.96 g/cc.

At this facility no chemical processes exist that are capable of producing uranium compounds with a higher density than that of UO_2. The basis for this conclusion is sound technical data of the published densities of uranium compounds.

(ii) *Maximum Uranium Enrichment* - Occurrence in any GNF-A process or operation of uranium with a fissile isotopic abundance greater than 5.00 wt. % U-235. No GNF-A fuel manufacturing or support processes enrich uranium beyond the maximum enrichment occurring in the process.

At this facility, this maximum enrichment is 5.00 wt.% U-235. The basis of this conclusion is Physical Law.

Example(s) Process Deviations Considered Credible

(i) In the case of a dissolution process in which Uranium is dissolved in nitric acid, it would not be considered Not Credible that in an upset condition, other uranium compounds may be present at greater densities than uranyl nitrate. The safety basis would therefore be based on the density of UO_2 as the worst-case since UO_2 is more dense than uranyl nitrate.

(ii) Represent a process deviation that consists of an accident sequence of many unlikely human actions or errors for which there is no reason or motive. In determining that there is no reason for such actions, consideration must be given to a wide range of possible motives, without a show of intent to cause harm. Necessarily, no such accident sequence could ever have actually happened in any fuel cycle facility.

Example(s) Human Actions Considered Not Credible

(i) *UF_6 Enrichments Higher than 5.00% in the Factory* - Introduction of > 5.00% enriched uranium into a process in the factory in sufficient quantities; for example, the unauthorized procurement and processing of by a Company employee.

At this facility, introduction to the process of UF_6 greater than 5.00 % would not be considered credible for the following reasons:

The facility is licensed for maximum 5.00 wt % U235.

Established written procedural controls dictate the procurement process for fissile materials. The controls establish the approval requirements, and notification steps required for purchase authorization.

Procedural facility controls, dictate the receipt operation for fissile materials. These controls establish the approval requirements, and notification steps when fissile material is received on site.

Appreciable quantities are transported in approved containers for a maximum 5.00 wt % U-235, and the containers for the transportation of higher enrichment are dramatically different – and certain to be detected by operations personnel.

Example(s) Human Actions Considered Credible

(i) In the case of a facility or operation where equipment was restricted to process UF_6 enriched to less than 5.00 wt %, it would not be considered Not Credible that the introduction of a weight percent greater than the restricted value could be introduced into the system. Controls for prevention/mitigation would therefore be implemented based on the consequence severity.

3.5 CONSEQUENCE AND LIKELIHOOD DETERMINATIONS

The credible hazards identified for the facility, which have the capability of producing conditions that exceed the performance requirements of 10 CRF 70.61 (b), (c) or (d) are included in the scope of the ISA. These accident sequences are rated by the ISA teams in terms of severity of consequence and unmitigated likelihood of an initiating event according to criteria defined in written plant procedures. In accordance with these plant procedures, this information is used to rank the relative importance of the barriers that limit the risk of the accident sequence, judge the adequacy of the prevention/mitigation (highly unlikely, unlikely), and apply assurance and management measures necessary to ensure the barriers are in-place when required to perform their function.

The credible accident sequences identified are categorized in relation to severity of consequence as described in the following section. The ISA teams use the ranking from this categorization along with unmitigated likelihood (discussed subsequently) to establish a risk ranking of the relative importance of the Items Relied On For Safety (IROFS) that limit the risk of the accident sequence. This in turn allows the ISA team to make judgment in accord with written procedural guidelines as to the

type and number of assurances and management measures necessary to assure the IROFS are in place when required to perform the function. This ranking and evaluation is performed in accordance with written facility procedures.

Severity of Consequence

Each accident sequence is categorized in terms of the performance requirements outlined in 10 CFR 70.61 (b), (c) and (d). The ISA team uses plant experience and guidance as found in NUREG/CR-6410, *Nuclear Fuel Cycle Accident Analysis Handbook* when evaluating and categorizing the severity of accident consequences. The consequence severity ranking in use at the facility is summarized in Table 3.1.

Table 3.1 Facility Consequence Severity Categories based on 10CFR 70.61

Consequence Severity	Workers	Offsite Public	Environment
3	D>1 Sv (100 rem) >ERPG-3 or a criticality	D>.25 Sv (25 rem) 30 mg sol U intake >ERPG-2 or a criticality	a criticality
2	.25 Sv (25 rem)<D≤ 1 Sv (100 rem) >ERPG-2 but ≤ERPG-3	.05 Sv (5 rem)<D≤ .25 Sv (25 rem) >ERPG-1 but ≤ERPG-2	Radioactive release >5000 x Table 2 App B 10 CFR Part 20
1	Accidents of lesser radiological and chemical exposures to workers than those above	Accidents of lesser radiological and chemical exposures to the public than those above	Radioactive releases producing effects less than those specified above

When performing the ISA, the ISA team determines two distinctly different likelihoods associated with the accident sequences that have the potential to exceed the performance requirements of 10 CFR 70.61.

The likelihood of the accident sequence occurring is first determined for the unmitigated case (unmitigated likelihood). This likelihood value is combined with the severity of the consequence to derive a relative risk associated with the consequence that exceeds the performance criteria (severity x unmitigated likelihood = unmitigated risk). The team uses this unmitigated risk to determine the degree of assurance to apply to the IROFS that prevent the unacceptable consequence.

The ISA team in making their determination of unmitigated likelihood, uses their understanding of the process, the accident sequence information, operating history and manufacturers/product information to determine which category of likelihood is appropriate. The guidelines are demonstrated in table 3.2

Table 3.2 Unmitigated Likelihood

LEVEL	FREQUENCY	LIKELIHOOD
3	more frequent than once every two years	likely to occur in the immediate future
2	every two to fifty years	likely to occur during the life of the facility
1	less frequent than once every fifty years	not likely to occur during the life of the facility

Unmitigated risk is determined by multiplying the severity ranking and the unmitigated likelihood. The unmitigated risks are then sub-divided into three groups as shown in table 3.3

Table 3.3 Unmitigated Risk Assignment Matrix

Consequence Severity			
3	**INTERMEDIATE**	**HIGH**	**HIGH**
2	**INTERMEDIATE**	**INTERMEDIATE**	**HIGH**
1	**LOW**	**LOW**	**LOW**
	1	**2**	**3**

Unmitigated Likelihood

The “high”, “intermediate” and “low” risk assignment is used by the ISA team as defined in written internal procedures to assign the appropriate type and number of assurance measures to each of the IROFS. In addition, Management Measures are applied to all elements of the safety system. Chapter 11 includes a discussion of control assurances and Management Measures.

The second consideration of likelihood relates to the (mitigated) likelihood of the accident sequence occurring with the preventive/mitigating IROFS in-place and is directly related to the requirements in 10 CFR 70.61 which requires unacceptable consequences to be limited.

3.6 IROFS IDENTIFICATION AND EVALUATION

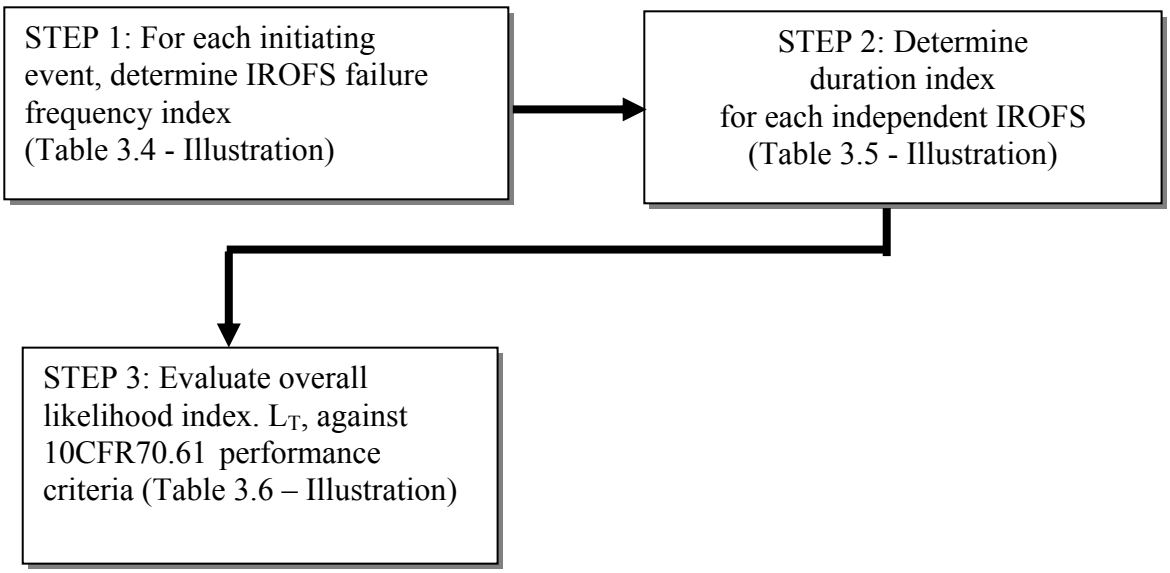
Items Relied On For Safety (IROFS) are those engineered or administrative controls, or control systems, which comprise the structures, systems and components, that form the preventative and/or mitigating barriers identified by the ISA. They are the barriers that prevent and/or mitigate the unacceptable consequences identified by the performance requirements of 10 CFR 70.61 (b), (c) and (d).

GNF-A commits to identify IROFS as a part of the ISA and include the identification of the IROFS in the ISA Summary Report prepared and maintained for the facility. The IROFS are defined in such a way to delineate their boundaries and to include the characteristics of the preventative/mitigating function, and the assumptions and conditions under which the item is relied on.

When evaluating accident sequences, the overall likelihood of the accident scenario must be determined and the adequacy of IROFS and management measures preventing or mitigating the accident sequence clearly identified.

Overall Likelihood – The *overall likelihood* is an index related to the failure of the IROFS identified with the accident scenario. Considerations include frequency of the initiating event (e.g, IROFS failure frequency), time period (duration of vulnerability) of IROFS failed condition prior to detection/response, and multiple independent IROFS (if present), which mitigate the progression of the accident sequence.

The methodology used to determine the overall likelihood, L_T, of each potential accident sequence that is identified by the ISA team is presented as in a simplified process map as follows:



Qualitative indices are assigned to the IROFS failure frequencies and duration indices and then "combined" together with factors representing the immunity to common mode failure to assign a score to the overall (total) likelihood, **(Table 3.6)**. The overall likelihood index, L_T, is then evaluated against the applicable limit for the corresponding consequence category.

Illustrated in **Tables 3.4 and 3.5,** respectively, are some indices for frequency and duration of failed IROFS. The frequencies shown are representative of a continuum of failure or duration frequencies normalized on a "per year" basis. In general, the factors in both of these tables are based on values for the potential failure of IROFS that can be determined and evaluated by suitably experienced ISA team members. Determination is based upon process knowledge, accident scenarios, operating history, and aspects of the control being credited. The same is true for the factors representing immunity from common mode failures. The scores are assigned using the guidance provided, but do not represent an absolute conclusion regarding the likelihood or consequence severity for accident scenarios.

Table 3.4 Index of IROFS Failure Frequencies

λ_f Index *	Failure Frequency	Description
1	Up to 10 occurrences per year	Very frequent failure, usually indicative of inadequate IROFS
0	No more than 1 failure per year	A less frequent failure typical of a single administrative IROFS (e.g., failure of trained operator performing a routine or non-routine task) with no failure detection system).
-1	No more than 1 failure in 10 years	Typical failure rate for a single active engineered control (AEC), or augmented administrative control, or administrative control with failure detection system, IROFS with demonstrated large safety margin or a redundant administrative IROFS (e.g., one over one verification.)
-1.5	No failure of this type in this facility in 30 years	Typical failure rate for single passively engineered controls (ENG) over the life of the facility (e.g., favorable geometry thick walled steel vessels).
-2	No more than 1 failure in 100 years	Failure rate of IROFS as based on performance of multiple units over the life of the facility. Exceptionally robust passive engineered IROFS (ENG), or an inherently safe process; or two independent active engineered controls (AECs), passive engineered controls (ENG), or enhanced administrative IROFS.
-6	External event with frequency no more than 1 occurrence per million years (10^{-6}/year)	Not physically possible (not credible) – events determined to be not credible as described in Section 2 of the GNF-A ISA Plan

*Index = Log_{10}(Failure Frequency per Year)

Table 3.5 Duration Index for IROFS Failures

λ_d Index*	Duration of Failure [a]	Description
. . .		
-5.7	Up to 1 Minute	Very short duration, very common
-5.0	Up to 5 Minutes	Short duration that is more common
-4.0	Up to 53 Minutes	Approximately 1 hour, very typical
-3.0	Up to 8 Hours	About 1 shift, less common
-2.0	Up to 3.5 Days	One half of a week, not very common at all
-1.0 . .	Up to 30 Days	About once a month, not very common

[a] i.e. Prior to identification and correction or remediation.
*Index = Log_{10}(Failure Duration/Total No. of Duration Units per Year)

The qualitative scoring method used by the ISA team is a conservative approach that provides a consistent, repeatable risk assessment. When appropriate, an independent risk analysis may be performed to ensure compliance with the performance requirements of 10CFR70.61. For example, a detailed criticality safety analysis (CSA) for a system, performed in accordance with the license requirements in Chapter 5 and internal implementing procedures, may determine that the performance requirements of the regulations are met.

Definitions of "highly unlikely" and "unlikely" likelihood terms are contained within **Table 3.6** and are not provided in the text. Section 3.7 illustrates the GNF ISA methodology for determining overall likelihood indices and for evaluating them against the applicable limits.

Table 3.6 Overall Likelihood for Accident Scenario

Index Limit (L_I)*	Likelihood per Year	Description of Limit
. . .		
-6.0	1.0E-06	
-5.0	1.0E-05	
-4.0	1.0E-04	Acceptable for High Consequence Accidents (highly unlikely)
-3.0	1.0E-03	Acceptable for Intermediate Consequence Accidents(unlikely)
-2.0	1.0E-02	
-1.0	1.0E-01	
. . .		

*See section 3.7

3.7 DETERMINATION OF OVERALL LIKELIHOOD

The *overall likelihood index* for accident sequence being reviewed, L_T, is calculated using equation 1 below from the terms described and the methods outlined in Table 3.4 and 3.5. For *sole IROFS*, L_T is just the Table 3.4 value (i.e. the base 10 log of the frequency index of failure per year). For multiple IROFS, it is the sum of the indices for the failures of the individual IROFS including the frequency index of initiating event (λ_f), duration index (λ_d), and independence from common mode failure (λ_{cmf}) as follows:

$$(1) \qquad L_T = \left[\sum_{i=1}^{N-1} (\lambda_{fi} + \lambda_{di}) * \lambda_{cmf,i} \right] + \lambda_{fN}$$

Where individual terms are defined as follows:

$\mathbf{L_T}$ = overall likelihood index for accident sequence being reviewed.

λ_{fi} = frequency index of failure per year of an individual (e.g., ith) IROFS considered in mitigating the accident sequence. Example values of the failure per year of a single IROFS is provided in Table 3.4.

λ_{di} = duration index of an individual (e.g., ith) IROFS considered in mitigating the accident sequence. Some values of the duration index for a single IROFS is provided in Table 3.5.

$\lambda_{cmf,i}$ = common mode failure factor. If the ith IROFS is independent[1] from all other IROFS being considered, its common mode failure factor is equal to 1.0. If it is completely dependent, e.g., subject to common mode failure with same initiating event that led to failure of the primary IROFS, the factor is 0.0.

λ_{fN} = frequency index of failure per year of a final individual (e.g., Nth) IROFS considered in mitigating the accident sequence to prevent the accident from occurring. Example values of the failure per year of a single IROFS is provided in Table 3.4. NOTE: It is conservative to use the $IROFS_{i=N}$ that exhibits the shortest duration index as the final IROFS.

[1] Independent as used here is interpreted in the same context as the double contingency principle (ANS/ANSI 8.1) that stipulates at least two unlikely, independent and concurrent changes take place before an accident initiating event can take place.

Worked Example:

For the purposes of this example, assume the accident sequence being reviewed by the ISA team consists of two *independent* IROFS, with an unmitigated severity, S_U = HIGH, and the following data are known about the two IROFS:

Known data for IROFS (i=1,2):

f1 = 1 per 10 years; ➔ $\boldsymbol{\lambda_{f1}} = \text{Log}_{10}(1/10) -1$
d1 = 0.01 years; ➔ $\boldsymbol{\lambda_{d1}} = \text{Log}_{10}(1/100) = -2$

f2 = 1 per 100 years; ➔ $\boldsymbol{\lambda_{f2}} = \text{Log}_{10}(1/100) = -2$
d2 = 0.001 years; ➔ $\text{Log}_{10}(1/1000) = \boldsymbol{\lambda_{d2}} = -3$

$\lambda_{cmf} = 1$

Substituting known data into equation (1), it follows that the overall likelihood index L_T is given by,

$\mathbf{L_T} = [-1 + -2] * (1) + -2 = -5$

NOTE: this example conservatively applies $IROFS_2$ as the final IROFS mitigating further progression of the accident sequence since it exhibits the shortest duration (e.g, most negative duration $\boldsymbol{\lambda_d}$ index)

In this example, the overall likelihood index $L_T = -5$, and it therefore meets the "applicable" acceptance limit from Table 3.6. For this accident sequence, $IROFS_1$ and $IROFS_2$ are necessary and sufficient in mitigating the accident sequence to an acceptable risk level.

3.8 ISA CHANGE MANAGEMENT

As described in Chapter 11, a formal configuration management process, governed by written, approved practices, ensures that plant design changes do not adversely impact the ISA at GNF-A. Every physical, documentation, and temporary change is evaluated by a multidiscipline (criticality, radiation, chemical, industrial, fire and/or explosion) safety review to determine the potential effect to the site license, the ISA, and to assure safe implementation and operation of the change.

Change that requires NRC prior approval per 10 CFR 70.72(c), will be submitted with ISA Summary revisions, but may not be implemented until NRC approval is obtained. An annual update to the ISA Summary is also submitted for changes implemented that do not require pre-approval by the NRC or otherwise affect the ISA Summary.

Change that does not require NRC prior approval, but may effect the ISA, require formal evaluation by the ISA team to determine the effects to any ISA documentation, including the ISA Summary. ISA methods are utilized to evaluate the adequacy of existing IROFS and associated management measures, and to designate new or additional IROFS and appropriate management measures as required. Modification to existing IROFS are evaluated to assure that capability, availability, and reliability of the IROFS are at least as capable as the original IROFS approved by the NRC.

A trained ISA facilitator is responsible for the development of modifications to the ISA documentation per written, approved procedures. ISA updates are approved prior to operation of any change.

Any IROFS' performance deficiencies will be corrected and result in changes to the ISA.

3.9 TRAINING AND QUALIFICATION OF ISA TEAMS

To ensure the adequacy of the results of the ISA, the analysis are performed by teams composed of individuals with expertise in engineering and process operations and in accord with internal procedures.

Each team consists of persons experienced and knowledgeable in the hazards that are known to exist in the study area (e.g. criticality, radiation, chemical, industrial, fire and explosion).

In addition, the team will include a cognizant engineer with experience and knowledge specific to the process being evaluated and a person directly experienced with the operations.

The team will include a Team Leader determined by management to be knowledgeable in the ISA process and procedures in use at the facility. Management may elect to augment Team Leader skills with a qualified facilitator familiar with the methods being used. The Team Leader assignment will be formally documented in writing.

3.10 MANAGEMENT MEASURES

The reliability and availability of IROFS to perform is a function of both assurance measures and management measures. The assurance measures assigned by the ISA teams and discussed in the ISA Summary Report are summarized in Table 3.7.

Table 3.7 Assurance Measures

	Importance		
	High (required)	Mid-level* (required)	Low
Passive Engineered Safeguards			ISA TEAM ASSIGNS
Technical report and/or drawings	X	X	
Manufacturing tolerance, corrosion tolerance for geometry controls	X		
Periodic verification of effectiveness	X		
Pre-operational audit	X	X	
ISA report	X	X	
Active Engineered Safeguards			
Periodic functional test and/or purchased material quality safeguard	X		
Calibration	X	X	
Periodic Maintenance	X	X	
Verification following maintenance	X		
Drawings	X	X	
Pre-operational audit	X	X	
ISA report	X	X	
Augmented Administrative/Administrative Safeguards			
Periodic verification	X	X	
Pre-operational audit	X	X	
Training Record	X	X	
ISA Report	X	X	
Controlled Procedure	X	X	X

* Deviations are described and approved in the Technical Report

The GNF safety management system (management measures) provides the overall management oversight and assurance that the safety program is maintained and functions properly, and is described in detail in Chapter 11.

CHAPTER 4.0
RADIATION SAFETY

4.1 ALARA (AS LOW AS IS REASONABLY ACHIEVABLE) POLICY

GNF-A's standard of care for occupationally exposed individuals is to maintain exposures below the limits established by the U.S. Nuclear Regulatory Commission. Beyond the standard of care, GNF-A's radiation protection staff has a commitment to establish, maintain, and implement an effective radiation protection program. This includes program commitment to maintain employee exposures As Low As Reasonably Acheivable (ALARA) which is delineated by documented radiation protection program practices and procedures. Area Managers are responsible for implementing the ALARA program via engineered controls and supervision of operations personnel.

The radiation safety function ensures that occupational radiation exposures are maintained ALARA via timely exposure monitoring and interaction via Radiation Safety Committee participation with manufacturing personnel, and annual ALARA program assessments with senior management.

The Wilmington Safety Review Committee (Chapter 2) also plays a role in the overall ALARA program at GNF-A.

4.2 RADIATION SAFETY PROCEDURES AND RADIATION WORK PERMITS (RWPS)

Routine work performed in radiation controlled areas is administered by the use of standard practices and procedures described in Chapter 11.0. Non-routine activities, particularly those performed by non-GNF-A employees, which generally are not covered by documented procedures, are administered by the Radiation Work Permit (RWP) system. The RWP system is described in documented plant practices and procedures.

RWPs are issued by a radiation safety technician or supervisor for non-routine operations not addressed by an operating procedure when special radiation control requirements are necessary. The RWP specifies the necessary radiation safety controls, as appropriate, including personnel monitoring devices, protective clothing,

respiratory protective equipment, special air sampling, and additional precautionary measures to be taken. RWPs are reviewed by radiation safety supervision.

The RWP requirements are reviewed by each affected individual and a copy is made available to the radiation safety function throughout the duration of the activity. Work is monitored by the radiation safety function as required. RWPs have expiration dates and the status of issued RWPs is reviewed on a weekly basis by a radiation safety technician or supervisor.

4.3 VENTILATION REQUIREMENTS

4.3.1 INTER-AREA AIR FLOW DESIGN

Ventilation equipment is designed to provide air flow from areas of lesser potential contamination to areas of higher potential contamination. Direction of air flow between areas is checked monthly or after significant changes to the ventilation system. If insufficient air flow results in airborne concentrations greater than 10 above the established DAC, then the affected processes are shut down. Specific facilities and capabilities of ventilation systems are detailed in Table 4.1.

4.3.2 ENCLOSURES AND LOCALIZED VENTILATION

Hoods and other localized ventilation designs are utilized to minimize personnel exposure to airborne uranium. Activities and process equipment that generate airborne uranium are designed with filtered enclosures, hoods, dust capturing exhaust ports and other devices which maintain air concentrations of radioactivity in work areas such that personnel exposures are below 10 CFR 20 limits under normal operating conditions.

Air flows through hood openings and localized vents are maintained in accordance with Table 4.1. Additionally, differential pressure indicators are installed across exhaust system filters to monitor system performance. The flows and differential pressures are checked monthly or after significant changes to the ventilation system. If insufficient air flow results in airborne concentrations greater than 10 DAC, then the affected processes are shut down in accordance with plant procedures.

4.3.3 EXHAUST SYSTEM

Potentially contaminated air is exhausted through high efficiency filter media which are at least 99.97% efficient for removal of 0.3 micron particles. HEPA filters in the exhaust system are equipped with a device for measuring differential pressure. Differential pressures greater than four inches of water are investigated. In no case will filters be operated at a differential pressure which exceeds the manufacturer's ratings for the filter.

Water scrubbers or other appropriate devices are provided where necessary to treat effluents before filtration. Such scrubbers are installed so that effectiveness of filters is maintained.

4.3.4 AIR RECIRCULATION

Room air may be recirculated within the uranium processing areas after being filtered. Room air recirculated within areas where airborne concentrations are likely to exceed 0.1 DAC is filtered by HEPA filters and/or water scrubbers.

4.4 AIR SAMPLING PROGRAM

Air samples are continuously taken from each main process area where airborne concentrations are likely to exceed 0.1 DAC when averaged over 40 hours to assess the concentrations of uranium in air. The air samples are collected in such a way that the concentrations of uranium measured are representative of the air which workers breathe. Air sampling results and individual personnel exposure assignments are monitored by the radiation safety function to evaluate the effectiveness of personnel exposure controls.

Evaluations of air sampling representativeness are performed in accordance with the methods and acceptance criteria in Table 2 of Regulatory Guide 8.25, "Air Sampling in the Workplace".

Filters from air samplers are changed each shift during normal operating periods or at more frequent intervals following the detection of an event that may have released airborne uranium, based upon knowledge of the particular circumstances. Filters are not changed as frequently during periods when no work is in progress. The filters are processed to determine the uranium concentration in air for each area.

Each air sampler is equipped with a rotameter to indicate flow rate of air sampled. These rotameters are calibrated or replaced at least every 18 months.

Air sampling results in excess of 2.5 DAC (8 hr. sample) and not resulting from a specific known cause are investigated to determine the probable cause. Operations or equipment will be shut down, and immediate corrective action will be taken, at locations where an air sample exceeds 10 DAC without a specific known cause. Corrective actions are implemented and documented based on the frequency and magnitude of events causing releases of airborne uranium.

Routine air sampling is supplemented by portable air sample surveys as required to evaluate non-routine activities or breaches in containment. Based on these surveys, additional radiation protection requirements for the particular operation may be established.

4.5 CONTAMINATION CONTROL

4.5.1 SURVEYS

Routine contamination survey monitoring is performed for uranium process and manufacturing areas including non-controlled areas such as hallways and lunch rooms immediately adjacent to controlled areas. Removable contamination measurements are made based on the potential for contamination in these areas and operational experience. Survey frequencies are determined by the radiation safety function. Survey results are compared to action guide values as specified in plant procedures and appropriate responses are taken.

The minimum survey frequencies and maximum removable contamination action levels are as follows:

Area	Frequency	Action Limit (dpm α/100 cm^2)
Controlled Areas (Floors & Other Readily Accessible Surfaces)	Weekly	≥5,000
Eating Areas used primarily by Controlled Area Personnel	Weekly	≥220
Non-controlled Areas	Monthly	≥220

When contamination levels in excess of action limits are found, mitigating actions are taken within 24 hours.

Personnel contamination surveys for external contamination on clothing and the body are required by personnel when exiting the change rooms. If contamination is found in excess of background levels, the individual attempts self-decontamination at the facilities provided in the change rooms. If decontamination attempts are not successful, decontamination assistance will be provided by the radiation safety function. If skin or personal clothing is still found contaminated above background levels, the individual may not leave the area without prior approval of the radiation protection function.

4.5.2 ACCESS CONTROL

Routine access points to controlled areas are established through change rooms. Each change room includes a step-off area provided between the contamination controlled and non-controlled areas. Instructions controlling entry and exit from controlled area are posted at the entry points. Personnel survey meters are provided in the step-off area of each change room for use by personnel leaving the controlled areas. Posted instructions address the use of the survey meters and appropriate decontamination methods.

Alternate access points to controlled areas are established for specific activities that are not accommodated by the change rooms. Such access is governed by approved

procedures, or Radiation Work Permits, which establish controls to prevent the spread of contamination to non-controlled areas.

4.5.3 PROTECTIVE CLOTHING

Protective clothing is provided to persons who are required to enter the controlled areas where personnel contamination potential exists as determined by the radiation safety function. The amount and type of protective clothing required for a specific area or operation is determined by operational experience and the contamination potential. Available clothing includes caps, hoods, laboratory coats, coveralls, safety glasses, boots overshoes, shoe covers, rubber and cloth gloves and safety shoes.

The minimum clothing requirement for airborne controlled area entry is as follows:

Area Workers	Inspectors and Visitors Only Observing Operations
Shoe covers or work area shoes	Shoe covers
Coveralls	Laboratory coats
Rubber gloves	Rubber gloves (as needed)
Safety glasses	Safety glasses

The protective clothing is removed upon exit in the controlled area change rooms.

In laboratory areas where uranium is handled the minimum protective clothing requirement for entry is a laboratory coat and safety glasses.

4.5.4 LEAK TESTING OF PLUTONIUM ALPHA SOURCES

The sources when not in use shall be stored in a closed container adequately designed and constructed to contain plutonium which might otherwise be released during storage.

The sources shall be tested for loss of plutonium at intervals not to exceed 110 days, using radiation detection instrumentation capable of detecting 0.005 μCi of alpha contamination.

If any survey or measurement performed as required by the preceding paragraph discloses the loss of more than 0.005 μCi of plutonium from the source, or if a source has been damaged or broken, the source shall be deemed to be losing plutonium. The licensee shall immediately withdraw it from use, and cause the source to be decontaminated and repaired, or disposed of in accordance with the Commission regulations.

Records of test results shall be kept in units of microcuries and maintained for inspection by the Commission.

Notwithstanding the periodic test required above, any plutonium alpha source containing not more than 0.1 μCi of plutonium is exempted from the above requirements.

4.6 EXTERNAL EXPOSURE

Deep-dose equivalent and shallow-dose equivalent from external sources of radiation are determined by individually assigned dosimeters. Personnel dosimeters are processed by a National Voluntary Laboratory Accreditation Program (NVLAP) accredited vendor. The capability exists to process dosimeters expeditiously if there is an indication of an exposure in excess of established action guides. Action guides for external exposures are established in plant procedures. Maximum radiation exposure action levels are specified in Section 4.9.

External exposures may be calculated by the radiation safety function on the basis of data obtained by investigation when the results of individual monitoring are unavailable or are invalidated by unusual exposure conditions.

4.7 INTERNAL EXPOSURE

Intakes are assigned to individuals based upon one or more types of measurements as follows: air sampling (described in Section 4.4), urinalysis and in vivo lung counting. Intakes are converted to committed dose equivalent (CDE) and committed effective dose equivalent (CEDE) for the purposes of limiting and recording occupational doses. Action levels are established in plant procedures to prevent an individual from exceeding the occupational exposure limits specified in 10 CFR 20. Maximum radiation exposure action levels are specified in Section 4.9. Control actions include temporarily restricting the individual from working in an area

containing airborne radioactivity, and actions are taken as necessary to assure against recurrence.

4.7.1 URINALYSIS PROGRAM

The urinalysis program is conducted primarily to evaluate the intake of soluble uranium to assure that the 10 CFR 20 intake limit of 10 mg is not exceeded. Individuals assigned to work in areas where soluble airborne uranium compounds are present in concentrations that are likely to result in intakes in excess of 10 percent of the applicable limits in 10 CFR 20 are monitored by urinalysis. The minimum sampling frequency for these individuals is biweekly. Urinalysis may also be used to monitor individuals involved in non-routine operations, perturbations or incidents.

Urine sampling frequencies and action levels are established in plant procedures based on the appropriate biokinetic models for the uranium compounds present. Results above the applicable action level are investigated. Urinalysis action levels are based on maximum radiation exposure action levels specified in Section 4.9. Results that exceed action levels result in a temporary work restriction for the individual to prevent additional exposure and allow a more accurate assessment of the intake.

4.7.2 IN VIVO LUNG COUNTING

Routine in vivo lung counting frequencies are established for individuals who normally work in areas where non-transportable uranium compounds are processed. Baseline and termination counts are performed when feasible. Lung counting frequencies are based upon individual airborne exposure assignments and previous counting results. The minimum count frequency is annual for individuals with an assigned intake greater than 10 percent of the Annual Limit on Intake (ALI).

Appropriate actions are taken based upon in vivo lung counting results to ensure the ALI will not be exceeded. If an individual's lung burden indicates an intake greater than the applicable action level, the individual is temporarily restricted from working in areas containing airborne uranium. In vivo lung counting action levels are based on the maximum radiation exposure action levels specified in Section 4.9.

4.8 SUMMING INTERNAL AND EXTERNAL EXPOSURE

Internal and external exposures determined as described in the preceding sections of this application are summed in accordance with the requirements of 10 CFR 20 for the purposes of limiting occupational doses and recording individual monitoring results.

4.9 ACTION LEVELS FOR RADIATION EXPOSURES

Work activity restrictions will be imposed when an individual's exposure exceeds 80% of the applicable 10 CFR 20 limit.

4.10 RESPIRATORY PROTECTION PROGRAM

The respiratory protection program shall be conducted in accordance with the applicable portions of 10 CFR 20. Respiratory protection equipment specifically approved by the National Institute for Occupational Safety and Health (NIOSH) is utilized.

4.10.1 QUALIFICATIONS OF RESPIRATOR USERS

Individuals designated to use respiratory protection equipment are evaluated by the medical function and periodically thereafter at a frequency specified by the medical function to determine if the individual is medically fit to use respiratory protection devices. If there are no medical restrictions precluding respirator use, the individual is provided respiratory training and fitting by a qualified instructor. Additional training on the use and limitations of self-contained breathing devices is provided to designated individuals.

An adequate fit is determined for all face-sealing respirators using either a quantitative fit test method or a qualitative method. Qualitative fit testing is acceptable if (1) it is capable of verifying a fit factor of 10 times the assigned protection factor (APF) for facepieces operated in a negative pressure mode or (2) it is capable of verifying a fit factor of $\geq$100 for facepieces operated in a positive pressure mode. Mask fits are re-evaluated annually.

4.10.2 RESPIRATORY PROTECTION EQUIPMENT

Only NIOSH approved respiratory protection equipment is utilized. Protection factors specified in 10 CFR 20 Appendix A are used for selecting the proper equipment and estimating personnel exposures.

4.10.3 EQUIPMENT MAINTENANCE

Respiratory protection equipment is cleaned, serviced, tested and inspected in accordance with the instructions specified by the manufacturer per the NIOSH certification and 10 CFR 20 for each respiratory protection device. Equipment maintenance is always conducted in accordance with the applicable portions of 10 CFR 20.

4.11 INSTRUMENTATION

Appropriate radiation detection instruments are available in sufficient number to ensure adequate radiation surveillance can be accomplished. Selection criteria of portable and laboratory counting equipment is based on the types of radiation detected, maintenance requirements, ruggedness, interchangeability and upper and lower limits of detection capabilities. The radiation safety function annually reviews the appropriateness of the types of instruments being used for each monitoring function. Table 4.2 lists examples of the types and uses of available instrumentation.

4.11.1 CALIBRATION

Portable instrumentation is calibrated before initial use, after major maintenance, and on a routine basis at least six months following the last calibration. Calibration consists of a performance check on each range scale of the instrument with a radioactive source of known activity traceable to a recognized standard such as the National Institute of Standards and Technology (NIST).

Prior to each use, operability checks are performed on monitoring and laboratory counting instruments. The background and efficiency of laboratory counting instruments are determined on a daily basis when in use.

TABLE 4.1
SPECIFIC FACILITIES & CAPABILITIES OF VENTILATION SYSTEMS

Facility	Alarms, Interlocks & Safety Features	Purpose
Hoods	Air flow during operation $\geq$ 80 linear feet per minute	Prevents spread of radioactive materials
	Effluent air filtered with HEPA filters	Prevents release of radioactive materials to environs
High Velocity Local Ventilation	Air flow designated to maintain an average of 200 linear feet per minute	Prevents spread of radioactive materials from work area to immediate room area
Recirculating Air Systems & Exhaust Air Systems	Air filtered in potentially contaminated zones with HEPA filters or water scrubbers	Removes essentially all contaminants from room and exhaust to environs
	Pressure drop indicator set to alarm at $\geq$4" $H_2O\Delta P$ across final filter	Maintains adequate circulation for removal of dust and contaminants from the room air
	Effluent air filtered with HEPA filters	Prevents release of radioactive materials in environs

TABLE 4.2
TYPES & USES OF AVAILABLE INSTRUMENTATION (TYPICAL)

Type	Typical Range	Routine Use
DOSE RATE METERS		
GM Low Range	0.01 mR - 2000 mR	Area Dose Rate Survey, Shipment Survey
GM High Range	0.1 mR - 1000 R	Emergency Monitoring
Ion Chamber - Low Range	0.1 mR - 10 R	Area Dose Rate Survey, Shipment Survey
Ion Chamber - High Range	1 mR - 1000 R	Emergency Monitoring
ALPHA SURVEY METERS	50 cpm - 2 x 10^6 cpm	Direct Personnel & Equipment Surveys
NEUTRON METERS	0.5 mR - 5 R	Special Dose Rate Surveys
LABORATORY INSTRUMENTATION		
Automatic air sample counter	N/A	Lab Analysis
Fixed geometry Geiger-Mueller counter	N/A	Lab Analysis
Scintillation Counter	N/A	Lab Analysis
In Vivo Lung Counter	N/A	Lung Deposition Measurements

CHAPTER 5.0
NUCLEAR CRITICALITY SAFETY

5.1 NUCLEAR CRITICALITY SAFETY PROGRAM MANAGEMENT

5.1.1 CRITICALITY SAFETY DESIGN PHILOSOPHY

The Double Contingency Principle as identified in nationally recognized American National Standard ANSI/ANS-8.1 (1998) is the fundamental technical basis for design and operation of processes within the GNF-A fuel manufacturing operations using fissile materials. As such, "process designs shall incorporate sufficient margins of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible." For each significant portion of the process, a defense of one or more system parameters is documented in the criticality safety analysis, which is reviewed and enforced.

The established design criteria and nuclear criticality safety reviews are applicable to:

- all new processes, facilities or equipment that process, store, transfer or otherwise handle fissile materials, and
- any change in processes, facilities or equipment which may have an impact on the established basis for nuclear criticality safety.

GNF-A nuclear criticality safety (NCS) program management commits to the following objectives:

a) providing sufficient safeguards and demonstrate adequate margin of safety to prevent an inadvertent criticality during conversion, production, storage, or shipment of enriched uranium product

b) protecting against the occurrence of an identified accident sequence in the ISA Summary that could lead to an inadvertent nuclear criticality

c) complying with the NCS performance requirements of 10 CFR 70.61

d) establishing and maintaining NCS controlled parameters and procedures

e) establishing and maintaining NCS subcritical limits for identified IROFS

f) conducting NCS evaluations (herein referred to as criticality safety analyses (CSAs) to assure that under normal and credible abnormal conditions, all fissile uranium processes remain subcritical, and maintain an adequate margin of safety

g) establishing and maintaining NCS IROFS, based on current NCS determinations

h) complying with established internal nuclear criticality safety design criteria

i) complying with the NCS ISA Summary requirements in 10 CFR 70.65(b)

j) complying with the NCS ISA Summary change process requirements in 10 CFR 70.72

5.1.2 EVALUATION OF CRITICALITY SAFETY

5.1.2.1 Changes to Facility

As part of the design of new facilities or significant additions or changes in existing facilities, Area Managers provide for the evaluation of nuclear hazards, chemical hazards, hydrogenous content of materials (including firefighting materials), and mitigation of inadvertent unsafe acts by individuals. Specifically, when criticality safety considerations are impacted by these changes, the approval to operate new facilities or make significant changes, modification, or additions to existing facilities is documented in accord with established facility practices and conform to the ISA change management process described in Chapters 3 and 11.

Change requests are processed in accordance with configuration management requirements described in Chapter 11. Change requests which establish or involve a change in existing criticality safety parameters require a senior engineer who has been approved by the criticality safety function to disposition the proposed change with respect to the need for a criticality safety analysis.

If an analysis is required, the change is not placed into operation until the criticality safety analysis is complete and other preoperational requirements are fulfilled in accordance with established configuration management practices.

5.1.2.2 Role of the Criticality Safety Function

Qualified personnel as described in Chapter 2.0 assigned to the criticality safety function determine the basis for safety for processing fissile material. Assessing both normal and credible abnormal conditions, criticality safety personnel specify functional requirements for criticality safety controls commensurate with design criteria and assess control reliability. Responsibilities of the criticality safety function are described in Chapter 2.0.

5.1.3 OPERATING PROCEDURES

Procedures that govern the handling of enriched uranium are reviewed and approved by the criticality safety function.

Each Area Manager is responsible for developing and maintaining operating procedures that incorporate limits and controls established by the criticality safety function. Area Managers assure that appropriate area engineers, operators, and other concerned personnel review and understand these procedures through postings, training programs, and/or other written, electronic or verbal notifications.

Documentation of the review, approval and operator orientation process is maintained within the configuration management system. Specific details of this system are described in Chapter 11.

5.1.4 POSTING AND LABELING

5.1.4.1 Posting of Limits and Controls

Nuclear criticality safety requirements for each process system that are defined by the criticality safety function are made available to work stations in the form of written or electronic operating procedures, and/or clear visible postings.

Posting may refer to the placement of signs or marking of floor areas to summarize key criticality safety requirements and limits, to designate approved work and storage areas, or to provide instructions or specific precautions to personnel such as:

- Limits on material types and forms.

- Allowable quantities by weight or number.
- Allowable enrichments.
- Required spacing between units.
- Control limits (when applicable) on quantities such as moderation, density, or presence of additives.
- Critical control steps in the operation.

Storage postings are located in conspicuous places and include as appropriate:

- Material type.
- Container identification.
- Number of items allowed.
- Mass, volume, moderation, and/or spacing limits.

Additionally, when administrative controls or specific actions/decisions by operators are involved, postings include pertinent requirements identified within the criticality safety analysis.

5.1.4.2 Labeling

Where practical, process containers of fissile material are labeled such that the material type, U-235 enrichment, and gross weights can be clearly identified or determined. Deviations from this process include: large process vessels, fuel rods, shipping containers, waste boxes/drums, contaminated items, UF_6 cylinders containing heels, cold trap cylinders, samples, containers of 1 liter volume or less, or other containers where labeling is not practical, or where the enrichment of the material contained is unknown (e.g. cleanout material).

5.2 ORGANIZATION AND ADMINISTRATION

5.2.1 GENERAL ORGANIZATION AND ADMINSTRATION METHODS

Procedures that govern the handling of enriched uranium are reviewed and approved by the criticality safety function.

Each Area Manager is responsible for developing and maintaining operating procedures that incorporate limits and controls established by the criticality safety function. Area Managers assure that appropriate area engineers, operators, and other concerned personnel review and understand these procedures through postings, training programs, and/or other written, electronic or verbal notifications.

Documentation of the review, approval and operator orientation process is maintained within the configuration management system. Specific details of this system are described in Chapter 11.

5.2.2 NCS ORGANIZATION

Specific details of the criticality safety function responsibilities and qualification requirements for manager, senior engineer, and engineer are described in Chapter 2.0.

Criticality safety function personnel are specifically authorized to perform assigned responsibilities in Chapter 2.0. All nuclear criticality safety function personnel have authority to shutdown potentially unsafe operations.

5.3 MANAGEMENT MEASURES

5.3.1 GENERAL CONFIGURATION MANAGEMENT

In accordance with ANSI/ANS-8.19 (2005), the criticality safety analysis is a collection of information that "provides sufficient detail clarity, and lack of ambiguity to allow independent judgment of the results." The CSA documents the physical/safety basis for the establishment of the controls. The CSA is a controlled element of the Integrated Safety Analysis (ISA) defined in Chapter 3.

Documented CSAs establish the nuclear criticality safety bases for a particular system under normal and credible abnormal conditions. A CSA is prepared or updated for new or significantly modified fissile units, processes, or facilities within GNF-A in accordance with established configuration management control practices defined in Chapter 3.

5.3.2 NCS CONFIGURATION MANAGEMENT

5.3.2.1 Training and Qualification of NCS Staff

A formalized Criticality Safety Engineer Training and Qualification Program shall be developed and maintained by more senior GNF-A NCS staff. This traininig and qualification program shall be premised on on-the-job training, demonstration of proficiency, periodic required technical classes or seminars, and participation in off-site professional development activities.

The established internal CSE Training and Qualification Program content emphasizes on-the-floor experience to fully understand the processes, procedures, and personnel required to assure that NCS controls on identified criticality safety parameters are properly implemented and maintained. The most effective training is on-the-job facility-specific training, which shall be documented by senior NCS management.

5.3.2.2 Auditing, Assessing and Upgrading the NCS Program

Details of the facility criticality safety audit program are described in Chapter 11. Criticality safety audits are conducted and documented in accordance with a written procedure and personnel approved by the criticality safety function. Findings, recommendations, and observations are reviewed with the Environment, Health & Safety (EHS) function manager to determine if other safety impacts exist. NCS audit findings are transmitted to Area Managers for appropriate action and tracked until closed.

Routine surveillance inspections of the processes and associated conduct of operations within the facility, including compliance with operating procedures, postings, and administrative guidelines, are also conducted as described in Chapter 11.

A nuclear criticality safety program review is conducted on a planned scheduled basis by nuclear criticality safety professionals independent of the GNF-A fuel

manufacturing organization in accordance with Section 11.6. This provides a means for independently assessing the effectiveness of the components of the nuclear criticality safety program.

The audit team is composed of individuals recommended by the manager of the criticality safety function and whose audit qualifications are approved by the GNF-A Facility Manager or Manager, EHS. Audit results are reported in writing to the manager of the nuclear criticality safety function, who disseminates the report to line management. Results in the form of corrective action requests are tracked to closure.

5.3.2.3 ISA Summary Revisions

(See Chapter 3)

5.3.2.4 Modifications to Operating and Maintenance Procedures

Procedures that govern the handling of enriched uranium are reviewed and approved by the criticality safety function.

Each Area Manager is responsible for developing and maintaining operating procedures that incorporate limits and controls established by the criticality safety function. Area Managers assure that appropriate area engineers, operators, and other concerned personnel review and understand these procedures through processes such as: postings, training programs, and/or other written, electronic or verbal notifications.

Documentation of the review, approval and operator orientation process is maintained within the configuration management system. Specific details of this system are described in Chapter 11.

5.3.2.5 Criticality Warning Systems (CWS) Design and Performance Requirements

The criticality accident alarm system (CWS) radiation monitoring unit detectors are located to assure compliance with appropriate requirements of ANSI/ANS-8.3

(2003). The location and spacing of the detectors are selected, taking into account shielding by massive equipment or materials. Spacing between detectors is reduced where high density building materials such as brick, concrete, or grout-filled cinder block shield a potential accident area from the detector. Low density materials of construction such as wooden stud construction walls, asbestos, plaster, or metal-corrugated panels, doors, non-load walls, and steel office partitions are accounted for with conservative modeling approximations in determining the detector placement.

The criticality accident alarm system initiates immediate evacuation of the facility. Employees are trained in recognizing the evacuation signal. This system, and proper response protocol, is described in the Radiological Contingency and Emergency Plan for GNF-A.

The nuclear criticality alarm system is a safety-significant system and is maintained through routine response checks and scheduled functional tests conducted in accordance with internal procedures. In the event of loss of normal power, emergency power is automatically supplied to the criticality accident alarm system.

5.3.2.6 Corrective Action Program

A GNF-A internal regulatory compliance tracking system is in place to track planned corrective or preventative actions in regard to procedural, operational, regulatory, or safety related deficiencies. The regulatory & compliance tracking (REGTRACK) is maintained by the Licensing organization and is standardized, site-wide system used by Operations, EHS and Quality organizations.

5.3.2.7 NCS Records Retention

Records of criticality safety analyses are maintained in sufficient detail and form to permit independent review and audit of the method of calculation and results. Such records are retained during the conduct of the activities and for six months following cessation of such activities to which they apply or for a minimum of three years.

A CSA is prepared or updated for each new or significantly modified unit or process system within GNF-A in accordance with established configuration management control practices defined in Chapter 11. Refer to Section 5.4.5.5 of this Chapter to see an example scope and content for a CSA.

5.4 METHODOLOGIES AND TECHNICAL PRACTICES

5.4.1 CONTROL PRACTICES

Criticality safety analyses identify specific controls necessary for the safe and effective operation of a process. Prior to use in any enriched uranium process, nuclear criticality safety controls are verified against criticality safety analysis criteria. The ISA program described in Chapter 3 implement performance based management of process requirements and specifications that are important to nuclear criticality safety.

5.4.1.1 Verification Program

The purpose of the verification program is to assure that the controls selected and installed fulfill the requirements identified in the criticality safety analyses. All processes are examined in the "as-built" condition to validate the safety design and to verify the installation. Criticality safety function personnel observe or monitor the performance of initial functional tests and conduct pre-operational audits to verify that the controls function as intended and the installed configuration agrees with the criticality safety analysis.

Operations personnel are responsible for subsequent verification of controls through the use of functional testing or verification. When necessary, control calibration and routine maintenance are normally provided by the instrument and calibration and/or maintenance functions. Verification and maintenance activities are performed per established facility practices documented through the use of forms and/or computer tracking systems. Criticality safety function personnel randomly review control verifications and maintenance activities to assure that controls remain effective.

5.4.1.2 Maintenance Program

The purpose of the maintenance program is to assure that the effectiveness of IROFS designated for a specific process are maintained at the original level of intent and functionality. This requires a combination of routine maintenance, functional testing, and verification of design specifications on a periodic basis. Details of the maintenance program are described in Chapter 11.

5.4.2 MEANS OF CONTROL

The relative effectiveness and reliability of controls are considered during the criticality safety analysis process. Passive Engineered Controls (Section 5.4.2.1) are preferred over all other system controls and are utilized when practical and appropriate. Active Engineered Controls (Section 5.4.2.2) are the next preferred method of control. Administrative Controls (Section 5.4.2.3) are least preferred, however augmented administrative controls are preferred over simple administrative controls. A criticality safety control must be capable of preventing a criticality accident independent of the operation or failure of any other criticality control for a given credible initiating event.

5.4.2.1 Passive Engineered Controls

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

5.4.2.2 Active Engineered Controls

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

5.4.2.3 Administrative Controls

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

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

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

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

5.4.3 SPECIFIC PARAMETER LIMITS

The **safe geometry** values of Table 5.1 below are specifically licensed for use at GNF-A. Application of these geometries is limited to situations where the neutron reflection present does not exceed that due to full water reflection. Acceptable geometry margins of safety for units identified in this table are 93% of the minimum critical cylinder diameter, 88% of the minimum critical slab thickness, and 76% of the minimum critical sphere volume.

When cylinders and slabs are not infinite in extent, the dimensional limitations of Table 5.1 may be increased by means of standard buckling conversion methods; reactivity formula calculations which incorporate validated K-infinities, migration areas (M^2) and extrapolation distances; or explicit stochastic or deterministic modeling methods.

The **safe batch** values of Table 5.2 are specifically licensed for use at GNF-A. Criticality safety may be based on U235 mass limits in either of the following ways:

- If double batch is considered credible, the mass of any single accumulation shall not exceed a safe batch, which is defined to be 45% of the minimum critical mass. Table 5.2 lists safe batch limits for homogeneous mixtures of UO_2 and water as a function of U235 enrichment over the range of 1.1% to 5% for

uncontrolled geometric configurations. The safe batch sized for UO_2 of specific compounds may be adjusted when applied to other compounds by the formula:

kgs X = (kgs UO_2 • 0.88) / f

where,	**kgs X**	**= safe batch value of compound 'X'**
	kgs UO_2	**= safe batch value for UO_2**
	0.88	**= wt. % U in UO_2**
	f	**= wt. % U in compound X**

- Where engineered controls prevent over batching, a mass of 75% of the minimum critical mass shall not be exceeded.

Subject to provision for adequate protection against precipitation or other circumstances which may increase concentration, the following **safe concentrations** are specifically licensed for use at GNF-A:

- A concentration of less than or equal to one-half of the minimum critical concentration.
- A system in which the hydrogen to U235 atom ratio (H/U235) is greater than 5200.

Table 5.1 Safe Geometry Values

Homogeneous UO_2-H_2O Mixtures	Weight Percent U235	Infinite Cylinder* Diameters (Inches)	Infinite Slab* Thickness (Inches)	Sphere Volume* (Liters)
	2.00	16.70	8.90	105.0
	2.25	14.90	7.90	75.5
	2.50	13.75	7.20	61.0
	2.75	12.90	6.65	51.0
	3.00	12.35	6.25	44.0
	3.25	11.70	5.90	38.5
	3.50	11.20	5.60	34.0
	3.75	10.80	5.30	31.0
	4.00	10.50	5.10	29.0
	5.00	9.50	4.45	24.0
Homogeneous Aqueous Solutions	**Weight Percent U235**	**Infinite Cylinder Diameters (Inches)**	**Infinite Slab Thickness (Inches)**	**Sphere Volume (Liters)**
	2.00	16.7	9.30	106.4
	2.25	15.0	8.40	80.5
	2.50	14.0	7.80	66.8
	2.75	13.3	7.30	56.2
	3.00	12.9	7.00	49.7
	3.25	12.5	6.70	44.8
	3.50	12.1	6.50	41.0
	3.75	11.9	6.30	38.0
	4.00	11.7	6.00	34.9
	5.00	9.5	4.80	26.0
Heterogeneous Mixtures or Compounds	**Weight Percent U235**	**Infinite Cylinder Diameters (Inches)**	**Infinite Slab Thickness (Inches)**	**Sphere Volume (Liters)**
	2.00	11.10	5.60	35.7
	2.25	10.50	5.10	30.7
	2.50	10.10	4.80	27.3
	2.75	9.70	4.60	24.7
	3.00	9.40	4.40	22.6
	3.25	9.20	4.30	20.9
	3.50	9.00	4.20	19.2
	3.75	8.90	4.10	18.2
	4.00	8.80	4.00	16.9
	5.00	8.30	3.60	13.0

*** These values represent 93%, 88% and 76% of the minimum critical cylinder diameter, slab thickness, and sphere volume, respectively. For enrichments not specified, smooth curve interpolation may be used.**

Table 5.2 Safe Batch Values for UO_2 and Water*

Nominal Weight Percent U235	Homogeneous UO_2 Powder & Water Mixtures (Kgs UO_2)	Heterogeneous UO_2 Pellets & Water Mixtures (Kgs UO_2)	Nominal Weight Percent U235	Homogeneous UO_2 Powder & Water Mixtures (Kgs UO_2)	Heterogeneous UO_2 Pellets & Water Mixtures (Kgs UO_2)
1.10	2629.0	510.0	4.00	25.7	24.7
1.20	1391.0	341.0	4.20	23.7	22.9
1.30	833.0	246.0	4.40	21.9	21.4
1.40	583.0	193.0	4.60	20.2	20.0
1.50	404.0	158.0	4.80	19.1	18.8
1.60	293.3	135.0	5.00	18.1	18.1
1.70	225.0	116.0			
1.80	183.0	102.0			
1.90	150.6	90.5			
2.00	127.5	81.6			
2.10	109.2	73.1			
2.20	96.8	66.4			
2.30	84.3	61.0			
2.40	74.7	56.1			
2.50	68.9	52.1			
2.60	60.5	48.8			
2.70	56.6	45.4			
2.80	52.2	42.9			
2.90	47.6	40.1			
3.00	44.5	38.1			
3.20	38.9	34.1			
3.40	34.6	31.0			
3.60	31.1	28.5			
3.80	28.3	26.4			

***NOTE: These values represent 45% of the minimum critical mass. For enrichments not specified, smooth curve interpolation of safe batch values may be used.**

5.4.4 CONTROL PARAMETERS

Nuclear criticality safety is achieved by controlling one or more parameters of a system within established subcritical limits. The internal ISA change management process may require nuclear criticality safety staff review of proposed new or modified processes, equipment, or facilities to ascertain impact on controlled parameters associated with the particular system. All assumptions relating to processes, equipment, or facility operations including material composition, function, and operation, including upset conditions, are justified, documented, and independently reviewed.

Identified below are specific control parameters that may be considered during the NCS review process:

5.4.4.1 **Geometry** - Geometry may be used for nuclear criticality safety control on its own or in combination with other control methods. Favorable geometry is based on limiting dimensions of defined geometrical shapes to established subcritical limits. Structure and/or neutron absorbers that are not removable constitute a form of geometry control. At GNF-A, favorable geometry is developed conservatively assuming unlimited water or concrete equivalent reflection, optimal hydrogenous moderation, worst credible heterogeneity, and maximum credible enrichment to be processed. Examples include cylinder diameters, annular inner/outer dimensions, slab thickness, and sphere diameters.

Geometry control systems are analyzed and evaluated allowing for fabrication tolerances and dimensional changes that may likely occur through corrosion, wear, or mechanical distortion. In addition, these systems include provisions for periodic inspection if credible conditions exist for changes in the dimensions of the equipment that may result in the inability to meet established nuclear criticality safety limits.

5.4.4.2 **Mass** - Mass control may be used for a nuclear criticality safety control on its own or in combination with other control methods. Mass control may be utilized to limit the quantity of uranium within specific process operations or vessels and within storage, transportation, or disposal containers. Analytical or non-destructive methods may be employed to verify the mass measurements for a specific quantity of material.

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

5.4.4.3 **Moderation** - Moderation control may be used for nuclear criticality safety control on its own or in combination with other control methods. When moderation is used in conjunction with other control methods, the area is posted as a 'moderation control area'. When moderation control is the primary design focus and is designated as a the primary criticality safety control parameter, the area is posted 'moderation restricted area'.

When moderation is the primary criticality safety control parameter the following graded approach to the design control philosophy is typically applied in accordance with established facility practices (in decreasing order of restriction):

- At each enriched uranium interface involving intentional and continuous introduction of moderation (e.g., insertion of superheated steam into reactor), at least three controls are required to assure that the moderation safety factor is not exceeded. At least two of these controls must be active engineered controls.
- At enriched uranium interfaces involving intentional but non-continuous introduction of moderation at least three controls are required to assure that the moderation safety factor is not exceeded. At least one of these controls must be an active engineered control, unless a moderation safety factor greater than 3 is demonstrated.
- For situations where moderation is not intentionally introduced as part of the process, the required number of controls for each credible failure mode must be established in accordance with the double contingency principle.

When the maximum credible accident is considered, the safety moderation limit (i.e., % H_2O or equivalent) must provide sufficient factor of safety above the process moderation limit. This 'moderation safety factor', which is the ratio of the safety moderation limit to the process moderation limit, will normally be three or higher,

but never less than two. The value of the moderation safety factor depends on the likelihood and time required for this system being considered to transition from the process moderation limit to the safety moderation limit.

In some cases, as described above, increased depth of protection may be required, but the minimum protection is never less than the following: two independent controls prevent moderator from entering the system through a defined interface and must fail before a criticality accident is possible. The quality and basis for selection of the controls is documented in accordance with Integrated Safety Analysis process described in Chapter 3. Controls for the introduction and limited usage of moderating materials (e.g. for cleaning or lubrication purposes) within areas in which the primary criticality safety parameter is moderation are approved by the criticality safety function.

5.4.4.4 **Concentration (or Density)** - Concentration control may be used for nuclear criticality safety control on its own or in combination with other control methods. Concentration controls are established to ensure that the concentration level is maintained within defined limits for the system. When concentration is the only parameter controlled to prevent criticality, concentration may be controlled by two independent combinations of measurement and physical control, each physical control capable of preventing the concentration limit being exceeded in a location where it would be unsafe. The preferred method of attaining independence being that at least one of the two combinations is an active engineered control. Each process relying on concentration control has in place controls necessary to detect and/or mitigate the effects of internal concentration within the system (e.g., Dynatrol density meter, Rhonan density meter, etc.), otherwise, the most reactive credible concentration (density) is assumed.

5.4.4.5 **Neutron Absorbe**r - Neutron absorbing materials may be utilized to provide a method for nuclear criticality safety control for a process, vessel or container. Stable compounds such as boron carbide fixed in a matrix such as aluminum or polyester resin; elemental cadmium clad in appropriate material; elemental boron alloyed stainless steel, or other solid neutron absorbing materials with an established dimensional relationship to the fissionable material are recommended. The use of

neutron absorbers in this manner is defined as part of a passive engineered control.

Credit may be taken for neutron absorbers such as gadolinia in completed nuclear fuel bundles (e.g., packaged and stored onsite for shipment) provided the following requirements are met:

- The presence of the gadolinia absorber in completed fuel rods is documented and verified using non-destructive testing; and the placement of rods in completed fuel bundles is documented in accordance with established quality control practices.

Credit may be taken for neutron absorbers that are normal constituents of filter media (e.g., natural boron) provided the following requirements are met:

- The failure or loss of the media itself also prevents accumulation of significant quantities of fissile material.
- The neutron absorber content is certified.

For fixed neutron absorbers used as part of a geometry control, the following requirements apply:

- The composition of the absorber are measured and documented prior to first use.
- Periodic verification of the integrity of the neutron absorber system subsequent to installation is performed on a scheduled basis approved by the criticality safety function. The method of verification may take the form of traceability (i.e. serial number, QA documentation, etc.), visual inspection or direct measurement.

5.4.4.6 **Spacing (or Unit Interaction)** - Criticality safety controls based on isolation or interacting unit spacing. Units may be considered effectively non-interacting (isolated) when they are separated by either of the following:

- 12-inches of full density water equivalent, or
- the larger of 12-foot air distance or the greatest distance across an orthographic projection of the largest of the fissile accumulations on a plane perpendicular to the line joining their centers.

For Solid Angle interaction analyses, a unit where the contribution to the total solid angle in the array is less than 0.005 steradians is also considered non-interacting (provided the total of all such solid angles neglected is less than one half of the total solid angle for the system). Transfer pipes of 2 inches or less in diameter may be excluded from interaction consideration, provided they are not grouped in close arrays.

Techniques which produce a calculated effective multiplication factor of the entire system (e.g., validated Monte Carlo or S_n Discrete Ordinates codes) may be used. Techniques which do not produce a calculated effective multiplication factor for the entire system but instead compare the system to accepted empirical criteria, (e.g., Solid Angle methods) may also be used. In either case, the criticality safety analysis must comply with the requirements of Sections 5.1.1 and 5.3.

5.4.4.7 **Material Composition (or Heterogeneity**) - The criticality safety analysis for each process determines the effects of material composition (e.g., type, chemical form, physical form) within the process being analyzed and identifies the basis for selection of compositions used in subsequent system modeling activities.

It is important to distinguish between homogeneous and heterogeneous system conditions. Heterogeneous effects within a system can be significant and therefore must be considered within the criticality safety analysis when appropriate. Evaluation of systems where the particle size varies take into consideration effects of heterogeneity appropriate for the process being analyzed.

5.4.4.8 **Reflection** - Most systems are designed and operated with the assumption of 12-inch water or optimum reflection. However, subject to approved controls which limit reflection, certain system designs may be analyzed, approved, and operated in situations where the analyzed reflection is less than optimum.

In criticality safety analysis, the neutron reflection properties of the credible process environment are considered. For example, reflectors more effective than water (e.g., concrete) are considered when appropriate.

5.4.4.9 **Enrichment** - Enrichment control may be utilized to limit the percent U-235 within a process, vessel, or container, thus providing a method for nuclear criticality safety control. Active engineered or administrative controls are required to verify enrichment and to prevent the introduction of uranium at unacceptable enrichment levels within a defined subsystem within the same area. In cases where enrichment control is not utilized, the maximum credible area enrichment is utilized in the criticality safety analysis.

5.4.4.10 **Process Characteristics** - Within certain manufacturing operations, credit may be taken for physical and chemical properties of the process and/or materials as nuclear criticality safety controls. Use of process characteristics is predicated upon the following requirements:

- The bounding conditions and operational limits are specifically identified in the criticality safety analysis and, are specifically communicated, through training and procedures, to appropriate operations personnel.
- Bounding conditions for such process and/or material characteristics are based on established physical or chemical reactions, known scientific principles, and/or facility-specific experimental data supported by operational history.
- The devices and/or procedures which maintain the limiting conditions must have the reliability, independence, and other characteristics required of a criticality safety control.

Examples of process characteristics which may be used as controls include:

- Conversion and oxidation processes that produce dry powder as a product of high temperature reactions.
- Experimental data demonstrating low moisture pickup in or on uranium materials that have been conditioned by room air ventilation equipment.
- Experimental/historical process data demonstrating uranium oxide powder flow characteristics to be directly proportional to the quantity of moisture present.

5.4.5 ANALYSIS METHODS

5.4.5.1 Keff Limit

Validated computer analytical methods may be used to evaluate individual system units or potential system interaction. When these analytical methods are used, it is required that the effective neutron multiplication factors, including applicable bias and bias uncertainty corrections, for credible process upset (accident) conditions are less than or equal to the established Upper Subcritical Limit (USL), that is:

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

Normal operating conditions include maximum credible conditions expected to be encountered when the criticality control systems function properly. Credible process upsets include anticipated off-normal or credible accident conditions and must be demonstrated to be critically safe in all cases in accordance with Section 5.1.1. The sensitivity of key parameters with respect to the effect on Keff are evaluated for each system such that adequate criticality safety controls are defined for the analyzed system.

5.4.5.2 Analytical Methods

Methodologies currently employed by the criticality safety function include hand calculations utilizing published experimental data (e.g., ARH-600 handbook), Solid Angle methods (e.g., SAC code), and Monte Carlo codes (e.g., GEMER, GEKENO) which utilize stochastic methods to approximate a solution to the 3-D neutron transport equation. Additional Monte Carlo codes (e.g., Keno-Va. and MCNP) or S_n Discrete Ordinates codes (e.g., ANISN, DORT, TORT or the DANTSYS code package) may be used after validation as described in Section 5.4.5.3 below has been performed.

GEMER (Geometry Enhanced MERIT) is a multi-group Monte Carlo program which approximates a solution to the neutron transport equation in 3-dimensional space. The GEMER criticality program is based on 190-energy group structure to represent the neutron energy spectrum. In addition, GEMER treats resolved resonances explicitly by tracking the neutron energy and solving the single-level Breit-Wigner equation at each collision in the resolved resonance range in regions containing

materials whose resolve resonances are explicitly represented. The cross-section treatment in GEMER is especially important for heterogeneous systems since the multi-group treatment does not accurately account for resonance self-shielding.

GEKENO (Geometry Enhanced KENO) is a multi-group Monte Carlo program which approximates a solution to the neutron transport equation in 3-dimensional space. The GEKENO criticality program utilizes the 16-energy group Knight-Modified Hansen Roach cross-section data set, and a potential scattering σ_p resonance correction to compensate for flux depression at resonance peaks. GEKENO is normally used for homogeneous systems. For infinite systems, K_∞ can be calculated directly from the Hansen Roach cross-sections using the program KINF.

5.4.5.3 Validation Techniques

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

The following definitions apply to the documented validation report(s):

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

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

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

Consistent with the requirements of ANSI/ANS-8.1 (1998), the criteria at GNF-A to establish subcriticality requires that for a system or process to be considered subcritical the calculated k_{eff} must be less than or equal to an established Upper Subcritical Limit (USL) as presented in the validation reports.

The validation of the calculational method and cross-sections considers a diverse set of parameters which include, but are not limited to:

- Fuel enrichment, composition and form of associated uranium materials;
- Geometry configuration of the system(e.g., shape, size, spacing, reflector, lattice pattern);
- Degree of neutron moderation in the system (e.g., H/fissile atom ratio)
- Homogeneity or heterogeneity of the system; and
- Characterization of the neutron energy spectra.

The selection of critical experiments for the GNF-A's criticality safety computer code validation for each identified area of applicability incorporates the following considerations:

- Critical experiments are assessed for completeness, accuracy, and applicability to the GNF-A nuclear fuel fabrication facility prior to its selection and use as a critical benchmark.
- Critical experiments are selected to cover the spectrum of parameters spanning the range of normal and credible abnormal conditions anticipated for past, current, and future analyzed uranium systems for GNF-A modeled systems.
- Critical experiments are drawn from multiple series and sources of critical experiments to minimize systematic error. The range of parameters characterized by selected critical experiments is used to define the area of applicability for the code.

The calculational bias, bias uncertainty and USL over the defined area of applicability are determined by statistical methods as follows:

- The normality of calculated k_{eff} values based on a set of critical experiments similar in the system configuration and nuclear characteristics is verified prior to the estimation of the bias and bias uncertainty.
- The calculational bias is determined either as a constant, if no trends exist or as a smooth and well-behaved function of selected characteristic parameters (e.g., hydrogen-to-fissile ratio, etc.) by regression analysis if trends exist

with parameters statistically important over the area of applicability. The bias is applied over its negative range and assigned a value of zero over its positive range.

- The bias uncertainty is estimated by a confidence interval of uniform width that ensures that there is at least a 95% level of confidence that a future k_{eff} value for a critical system will be above the lower confidence limit.
- The USL is established based on confidence interval with MMS for the area of applicability as follows:

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

At GNF-A, a minimum MMS = 0.03 shall be used to establish the acceptance criteria for criticality calculations.

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

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

where σ is the standard deviation of the k_{eff} value obtained with Monte Carlo calculation.

5.4.5.4 Computer Software & Hardware Configuration Control

The software and hardware used within the criticality safety calculational system is configured and controlled in accordance with internal software configuration procedures. Software changes are conducted in accordance with an approved configuration control program described in Chapter 11 that addresses both hardware and software qualification.

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

Each individual workstation is verified to produce results identical to the development workstation prior to use of the software for criticality safety calculations demonstrations on the production workstation.

Modifications to software and nuclear data that may affect the calculational logic require re-validation of the software. Modifications to hardware or software that do not affect the calculational logic are followed by code operability verification, in which case, selected calculations are performed to verify identical results from previous analyses. Deviations noted in code verification that might alter the bias or uncertainty requires re-qualification of the code prior to release for use.

5.4.5.5 Criticality Safety Analysis (CSA)

The scope and content of any particular CSA reflects the needs and characteristics of the system being analyzed and includes applicable information requirements as follows:

- **Scope** - This element defines the stated purpose of the analysis.
- **General Discussion** - This element presents an overview of the process that is affected by the proposed change. This section includes as appropriate; process description, flow diagrams, normal operating conditions, system interfaces, and other important to design considerations.
- **Criticality Safety Controls/Bounding Assumptions** - This element defines a minimum of two criticality safety controls that are imposed as a result of the analysis. This section also clearly presents a summary of the bounding assumptions used in the analysis. Bounding assumptions include; worst credible contents (e.g., material composition, density, enrichment, and moderation), boundary conditions, inter-unit water, and a statement on assumed structure. In addition, this section includes a statement which summarizes the interface considerations with other units, subareas and/or areas.
- **Model Description** - This element presents a narrative description of the actual model used in the analysis. An identification of both normal and credible upset (accident condition) model filenaming convention is provided. Key input listings and corresponding geometry plot(s) for both normal and credible upset cases are also provided.

- **Calculational Results** - This element identifies how the calculations were performed, what tools or reference documents were used, and when appropriate, presents a tabular listing of the calculational result and associated uncertainty (e.g., Keff + 3σ) results as a function of the key parameter(s) (e.g., wt. fraction H_2O). When applicable, the assigned bias of the calculation is also clearly stated and incorporated into both normal and/or accident limit comparisons
- **Safety During Upset Conditions** - This element presents a concise summary of the upset conditions considered credible for the defined unit or process system. This section include a discussion as to how the established nuclear criticality safety limits are addressed for each credible process upset (accident condition) pathway.
- **Specifications and Requirements for Safety** - When applicable, this element presents both the design specifications and the criticality safety requirements for correct implementation of the established controls. These requirements are incorporated into operating procedures, training, maintenance, quality assurance as appropriate to implement the specifications and requirements.
- **Compliance** - This element concludes the analysis with pertinent summary statements and includes a statement regarding license compliance.
- **Verification** - Each criticality safety analysis is verified in accordance with Section 5.3.2.5 by a senior engineer approved by the criticality safety function and who was not involved in the analysis.
- **Appendices** - Where necessary, a summary of information ancillary to calculations such as parametric sensitivity studies, references, key inputs, model geometry plots, equipment sketches, useful data, etc., for each defined system is included.

5.4.5.6 Technical Reviews

Independent technical reviews of proposed criticality safety control limits specified in criticality safety analyses are performed. A senior engineer within the criticality safety function is required to perform the independent technical review.

The independent technical review consists of a verification that the neutronics geometry model and configuration used adequately represent the system being analyzed. In addition, the reviewer verifies that the proposed material characterizations such as density, concentration, etc., adequately represent the system. The reviewer also verifies that the proposed criticality safety controls are adequate.

The independent technical review of the specific calculations and computer models is performed using one of the following methods:

- Verify the calculations with an alternate computational method.
- Verify methods with an independent analytic approach based on fundamental laws of nuclear physics.
- Verify the calculations by performing a comparison to results from a similar design or to similar previously performed calculations.
- Verify the calculations using specific checks of the computer codes used, as well as, evaluations of code input and output.
- Based on one of these prescribed methods, the independent technical review provides a reasonable measure of assurance that the chosen analysis methodology and results are correct.

CHAPTER 6.0
CHEMICAL SAFETY

6.1 CHEMICAL SAFETY PROGRAM

It is the policy of GNF-A to provide a safe and healthy work place by minimizing the risk of chemical exposure from licensed material to employees and members of the general public. The chemical safety program is applicable to the chemicals associated with the authorized activities in Chapter 1 and include UF_6 and hydrofluoric acid as well as any other hazardous chemicals associated with licensed material activities. GNF-A's chemical safety program is documented in written, approved practices that are followed, and ensures that processes and operations comply with applicable federal and state regulations pertaining to chemical safety.

Hazard evaluations are performed on nuclear and non-nuclear operations within the nuclear manufacturing operations where the potential exists for hazardous chemicals to be used in such a manner that they could affect the integrated safety program. This ensures appropriate controls are in place for adequate protection of the general public and safe use by employees, and that the use of chemicals does not create potential conditions that adversely affect the handling of licensed nuclear materials.

Employees using hazardous materials are trained to ensure safe handling, use, and disposal.

6.2 CONTENTS OF CHEMICAL SAFETY PROGRAM

The following management control elements are incorporated into GNF-A's chemical safety program:

6.2.1 CHEMICAL SAFETY IN INTEGRATED SAFETY ANALYSIS

Considerations of chemical safety for hazardous materials as described in this Chapter are incorporated in GNF-A's Integrated Safety Analysis program. This program includes UF_6 and hydrofluoric acid as well as any other hazardous chemicals associated with licensed material activities. GNF-A's Integrated Safety Analysis Program is explained in detail within Chapter 3. The ISA change

management process assures that internal process descriptions are sufficiently detailed to allow an understanding of the chemical process hazards. Management assurances on identified chemical safety related IROFS are described in Chapter 11.

6.2.2 CHEMICAL APPROVAL / EVALUATION

Prior to new hazardous materials being brought on-site or used in a process, they are approved through the environmental health and safety function and the chemical and fire safety function. The formal approval process consists of evaluations of the following potential hazards:

- Physical Hazards
- Health Hazards
- Fire / Explosive Hazards
- Potential Impact on handling of licensed nuclear material

The conclusions of this approval process may dictate the following assurance of chemical process safety:

- New procedures or changes in existing procedures
- Maintenance programs for control related equipment
- Configuration management
- Emergency Planning
- Training

6.2.3 LABELING & IDENTIFICATION

Hazardous materials or conveyance systems are labeled or identified to meet applicable regulations. The proper identification of hazardous materials decreases the likelihood of improper use, handling and disposal reducing potential negative consequences.

6.2.4 EMPLOYEE TRAINING & AWARENESS

Radiation workers receive nuclear safety training and other job related training (Chapter 11) which includes safety information related to chemicals associated with nuclear material and chemicals in the area which could impact the safety of the process.

6.2.5 INCIDENT CLASSIFICATION & INVESTIGATION

GNF-A's incident classification and investigation program is discussed in Chapter 11.

6.2.6 CONDUCT OF OPERATIONS

Other elements of the chemical safety program are included in Chapter 11, "Management Measures".

CHAPTER 7.0
FIRE SAFETY

GNF-A's fire protection is achieved by appropriate combinations of fire prevention measures and response systems. Such measures and systems are designed and maintained in accordance with federal, state, and local codes, appropriate industry standards and prudent practices. The National Fire Protection Association (NFPA) is the most common standard and practice used as guidance.

7.1 FIRE PROTECTION PROGRAM RESPONSIBILITY

The Emergency Organization is comprised of functional groups capable of assisting and/or advising in the prevention, response to and controlling of emergency situation. The structure of the Emergency Organization is detailed within the Radiological Contingency and Emergency Plan for GNF-A.

7.2 FIRE PROTECTION PROGRAM

Fire hazard analysis is incorporated into the GNF-A's Integrated Safety Analysis (ISA) program and/or site process reviews. The ISA program includes a provision for fire safety review as described in Chapter 3.

Routine inspection and testing of the fire protection system are conducted by GNF-A personnel and/or contract personnel under the direction of the manager of the site security & emergency preparedness function. Responsibility for maintenance, operation, and engineering of the fire protection system and equipment is specified in written, approved GNF-A procedures.

The fire protection program equipment is maintained as part of the formal, planned preventative maintenance program at GNF-A.

Review and control of modifications of the facility or processes to minimize fire hazards is part of configuration management described in Chapter 11.

An approved cutting and welding procedure known as a hot work permit is provided to control welding and torch cutting activities as a means of fire prevention.

Basic fire protection training is provided as needed. Additionally new employees and contractors are trained during orientation programs. The emergency response

team is given documented training as part of the emergency preparedness program described in Chapter 8.

A system is provided to enable reporting of fire incidents to the emergency response organization. Fire alarm pull stations are strategically located throughout the facility. Areas with potential fire hazards are equipped with appropriate fire detection and/or suppression systems.

In order to ensure emergency response readiness a comprehensive emergency exercise is conducted on an annual basis.

7.3 ADMINISTRATIVE CONTROLS

(See Chapter 11, Section 11.6.4)

7.4 BUILDING CONSTRUCTION

7.4.1 EXISTING BUILDING

The existing building's original design is in accordance with the local, state, federal and national codes, standards and/or regulations in effect at the time of construction. The building and appurtenances used to process and store hazardous materials are designed to provide containment of such material under the conditions of fire and explosion.

7.4.2 DRY CONVERSION PROCESS FACILITY (DCP)

The building's design is in accordance with the local, state, federal and national codes, standards and/or regulations in effect at the time of construction. The building and appurtenances used to process and store hazardous materials are designed to provide containment of such material under the conditions of fire and explosion. Recognizing the requirement for moderation restriction, the DCP facility is compartmentalized with fire walls to control the spread of fire using appropriate techniques.

7.5 VENTILATION SYSTEMS

Ventilation systems are designed to perform the following functions in the event of a fire:

- Air supply closed and air exhaust will continue
- Automatic closing of fire dampers and doors

7.6 PROCESS FIRE SAFETY

Potential fire hazards are determined, evaluated, and controlled by internal and external personnel using industry accepted methods, analysis, and procedures.

7.7 FIRE DETECTION AND ALARM SYSTEMS

7.7.1 DETECTION DEVICES

Areas where fire or explosion hazards are present, automatic detection equipment is installed. Equipment such as the following is utilized:

- Smoke Detectors
- Heat Detectors
- Hydrogen Detectors (DCP only)

7.7.2 ALARMS

- Audible fire alarms are installed in specified locations throughout the facility. Such alarms are monitored by a continuously manned, central control station that monitors fire detection system and zone status.
- Manual fire alarm actuators (pull-boxes) are installed in appropriate locations throughout the facility and serve to activate a coded fire alarm.

7.8 FIRE SUPPRESSION EQUIPMENT

GNF-A's fire protection system is designed in accordance with the applicable NFPA.

Selection of equipment for suppression of fire takes into account the severity of the hazard, the type of activity to be performed, the potential consequences of a fire, and the potential consequences of use of the suppression equipment (including, risk of accidental criticality).

Automatic sprinkler systems are specifically excluded from areas where moderation control is the primary nuclear criticality safety controlled parameter.

Portable fire extinguishers, of sufficient capacity, quantity and type of suppression agent used, are available and maintained throughout the facility.

7.9 FIRE PROTECTION WATER SYSTEM

- The fire protection water system is supplied by site water wells.
- Prime components of the fire protection system are as follows:
- Elevated tank capable of supplying dedicated water to the fire protection system.
- Ground level fire protection reservoir with a dry hydrant connection.
- Pump back up system with automatic startup capabilities for supplying the fire protection loop from the retention basin with water at adequate pressure.
- A jockey pump to maintain sufficient pressure on the fire protection system.
- A pump under the water tower with automatic startup and manual stop.
- A fire main loop around the prime production facilities.
- A series of branch headers supplying fire protection water to sectionalized sprinkler system in each building.
- A supervised alarm and warning system providing full time coverage of prime fire protection safety auxiliaries such as sprinkler system supply valve closing, sprinkler system water flow, fire pump operations, smoke detection operation, etc.
- Fire hose on reels connected to the primary fire protection system.

7.10 RADIOLOGICAL CONTINGENCY AND EMERGENCY PLAN (RC&EP)

GNF-A maintains plans that provide information needed by fire-fighting personnel responding to an emergency. This plan is described in Chapter 8.

7.11 EMERGENCY RESPONSE TEAM

Fire training of the Emergency Response Team is conducted for the response to incipient stage fires in accordance with emergency planning requirements. Outside agency fire departments are contacted for more serious fires which include structural fires.

CHAPTER 8.0

RADIOLOGICAL CONTINGENCY AND EMERGENCY PLAN

GNF-A shall maintain and execute the response measure in the Radiological Contingency and Emergency Plan as specified in Safety License conditions of Materials License SNM-1097; or as further revised by the licensee consistent with 10 CFR 70.32(i). The Radiological Contingency and Emergency Plan incorporates the requirements established by the Emergency Planning and Community Right-to-Know Act of 1986, Title III, Publication L 99-499.

The current Radiological Contingency and Emergency Plan is dated January 8, 2007 and was submitted January 18, 2007.

GNF-A will make no changes to the Radiological Contingency and Emergency Plan which would decrease its effectiveness without prior approval of the NRC.

Changes that do not decrease the effectiveness of the Radiological Contingency and Emergency Plan, will be reported within six months of the change to the Director, Division of Nuclear Security, Office of Nuclear Security and Incident Response, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

The requirements of the Radiological Contingency and Emergency Plan are implemented through approved documented procedures maintained by GNF-A.

CHAPTER 9.0
ENVIRONMENTAL PROTECTION

9.1 ENVIRONMENTAL PROTECTION PROGRAM & REPORTS

The GNF-A nuclear fuel fabrication facility maintains an Environmental Protection Program for the site. A primary purpose of the Environmental Protection Program is to assure that exposure of the public and environment, to hazardous materials used in facility operations, is kept As Low As Reasonably Achievable (ALARA).

GNF-A's licensed activity prepared an Environmental Report dated January 1, 1974, revised July 1983, revised May 1989, and supplemented in March 2007. Future Environmental Report updates will be prepared and submitted to the NRC Licensing Staff on a schedule contingent upon the operating term of the license. The review and updating will be concurrent with each renewal application.

As part of the design of new facilites or significant additions or changes in existing facilities, environmental considerations are assessed in accord with established facility practices and conform to ISA change management process described in Chapter 3.

Change requests are processed in accordance with configuration management requirements described in Chapter 11. Change requests which establish or involve a change in existing environmental controls require environmental staff review and disposition of the proposed change with respect to impact on established environmental protection programs.

9.2 AIR EFFLUENT CONTROLS AND MONITORING

Air effluent control systems are designed and operated to assure compliance with regulatory requirements. Operations that could potentially exhaust radioactive materials have air effluent controls that are monitored by representative stack sampling to demonstrate compliance with regulations. Samples are collected and analyzed so as to be representative of the discharges during production operations. Adequate controls and evaluations are in place to monitor, assess and take necessary

protective actions that may be needed for circumstances not explicitly treated. The ventilation and exhaust systems are described in Chapter 4.

Radioactivity in releases of radioactive materials in gaseous effluents from the facility will be reported to the NRC on a semi-annual basis.

9.3 LIQUID TREATMENT FACILITIES

A treatment facility, with sufficient capacity and capability to enable treatment, sampling, analysis, and discharge of liquids in accordance with the regulations, is provided and maintained in proper working order during the operation of the plant.

Compliance with NRC 10 CFR 20 effluent radioactivity limits for discharge of liquids to the unrestricted area is assured by on-line gamma energy monitoring or other appropriate controls. Quarantine tanks, diversion tanks, and filtration operations are provided to assure the liquid is below regulatory-driven limits. Process RadWaste and laundry streams are released to the final process basins. The discharges from these operations are controlled to assure uranium concentration in the final process effluent is less than 5 ppm in one day and less than 0.2 ppm daily average for a month. Assurance is provided that uranium levels are in compliance with 10 CFR 20.1301 and 1302, thereby meeting the unrestricted release limit.

A continuous proportional sample of process liquid effluent release to the Northeast Cape Fear River is collected. The sampling program design is such that, typically, a daily composite is analyzed for uranium content; a weekly composite of this sample is analyzed for gross alpha activity and gross beta activity; and the determination of technetium 99 is performed on a composite sample which is a collection of weekly samples over a six month period.

In the dry process for converting UF_6 to UO_2, hydrogen fluoride dissolved in water is generated. This hydrofluoric acid is collected in a bulk storage tank facility to await shipment (see Chapter 1, Section 1.3.3.2).

Radioactivity in releases of radioactive materials in liquid effluents from the facility will be reported to the NRC on a semi-annual basis.

9.4 SOLID WASTE MANAGEMENT FACILITIES

Solid waste management facilities, with sufficient capability to enable preparation, packaging, storage, and transfers to licensed disposal sites in accordance with the regulations, is provided and maintained in proper operating condition as required to support the operation of the plant. Combustible wastes may be incinerated on site.

9.5 PROGRAM IMPLEMENTATION

GNF-A's environmental monitoring program includes the elements illustrated in Figure 9.1. Analytical sensitivities (minimum detection levels) are illustrated in Figure 9.1. Action levels will be included in documented procedures for environmental monitoring parameters as appropriate so that internal review and other actions are initiated. Such action levels provide guidance in assuring compliance within 10 CFR 20 limits. Locations of (a) air sampling sites; (b) vegetation and soil sampling points; (c) surface water monitoring points; and (d) monitoring wells are illustrated in Figures 9.2, 9.3, 9.4, and 9.5, respectively. For monitoring wells found not to contain water at time of sampling, an evaluation is performed by the EHS function to determine if alternate well sampling data may be used or other assessments will be used. These program elements, analytical sensitivities, and/or locations may be changed without prior NRC Licensing Staff approval, provided: (1) a documented evaluation by the EHS function demonstrates that the changes will not decrease the overall effectiveness of the environmental monitoring program; and, (2) the documented evaluation is maintained on file at the facility and the changes are submitted to the NRC Licensing Staff in the subsequent Environmental Report update.

9.6 EVALUATIONS

The EHS function performs a periodic evaluation of vendors contracted to analyze environmental samples. The evaluations consider applicable methods such as "spike" and "replicate sample" submittals.

9.7 OFF-SITE DOSE

Compliance with NRC 10 CFR 20, Subpart D, and EPA 40 CFR 190 regulations for off-site dose requirements to the maximally exposed individual is demonstrated by

assuring that the off-site annual dose does not exceed 25 mRem. Additionally, compliance with 10 CFR 20.1101(d) regulations for off-site dose projections due to air emissions is demonstrated by assuring that off-site annual dose (due to air emissions of radioactivity) does not exceed 10 mRem.

9.8 ALARA

Compliance and the ALARA concept are inherent in the Environmental Program in terms of comprehensive monitoring, analysis, and evaluation of air emissions, liquid effluents and disposition of solid waste. Management controls, quality assurance and program implementation provide (1) representative measurements of radioactivity in the highest potential exposure pathways and (2) verification of the accuracy of the effluent monitoring program of those environmental exposure pathways. Trends are assessed using monitoring results to evaluate plant operations, in terms of "control-at-the-source" of contamination and the containment of radioactivity; the projections of potential dose to off-site populations; and the detection of any unanticipated pathways for the transport of radionuclides within the environment. Monitoring with periodic evaluations are summarized and presented to senior management on an annual basis.

FIGURE 9.1
GNF-A'S ENVIRONMENTAL MONITORING PARAMETERS

TYPE OF SAMPLE	ANALYSES	TYPICAL SAMPLING FREQUENCY	ROUTINE MINIMUM DETECTION LEVEL	
Air Particulates - Point Sources	Alpha	Continuous (Collection Weekly)	1.0E-12 microcuries per milliliter	
Ambient Air - On-Site	Alpha	Continuous (Collection Weekly)	0.5E-15 microcuries per milliliter	
Process Liquid At On-Site Discharge Point	Uranium Content Alpha; Beta	Daily; Weekly	0.02 parts per million - uranium	
			3.0E-8 microcuries per milliliter alpha	5E-8 microcuries per milliliter beta
Ground Water - On-Site	Uranium Content Alpha; Beta	Monthly; Quarterly	0.02 parts per million - uranium	
			5 picocuries per liter - alpha	20 picocuries per liter - beta
River Water - Upstream and Downstream of Site Discharge	Alpha; Beta	Monthly	5 picocuries per liter alpha	20 picocuries per liter beta
Sediment - Above Site Dam	Uranium	Annually	0.02 parts per million - uranium	
Soil - On-Site	Uranium	Semi-Annually	0.02 parts per million - uranium	
Vegetation - On-Site	Fluoride	Semi-Annually	1.0 parts per million - fluoride	

Locations of Ambient Air
Sampling Sites
(Typical)

FIGURE 9.2

Ambient Air Samples		
1	AANE	On Site NE of FMO
2	AASE	On Site SE of FMO
*3	AASS	On Site South of FMO
4	AASW	On Site SW of FMO
*5	AADK	On Site GE dock on NE Cape Fear River
6	AAFE	On Site NE of FET

*State split sampling

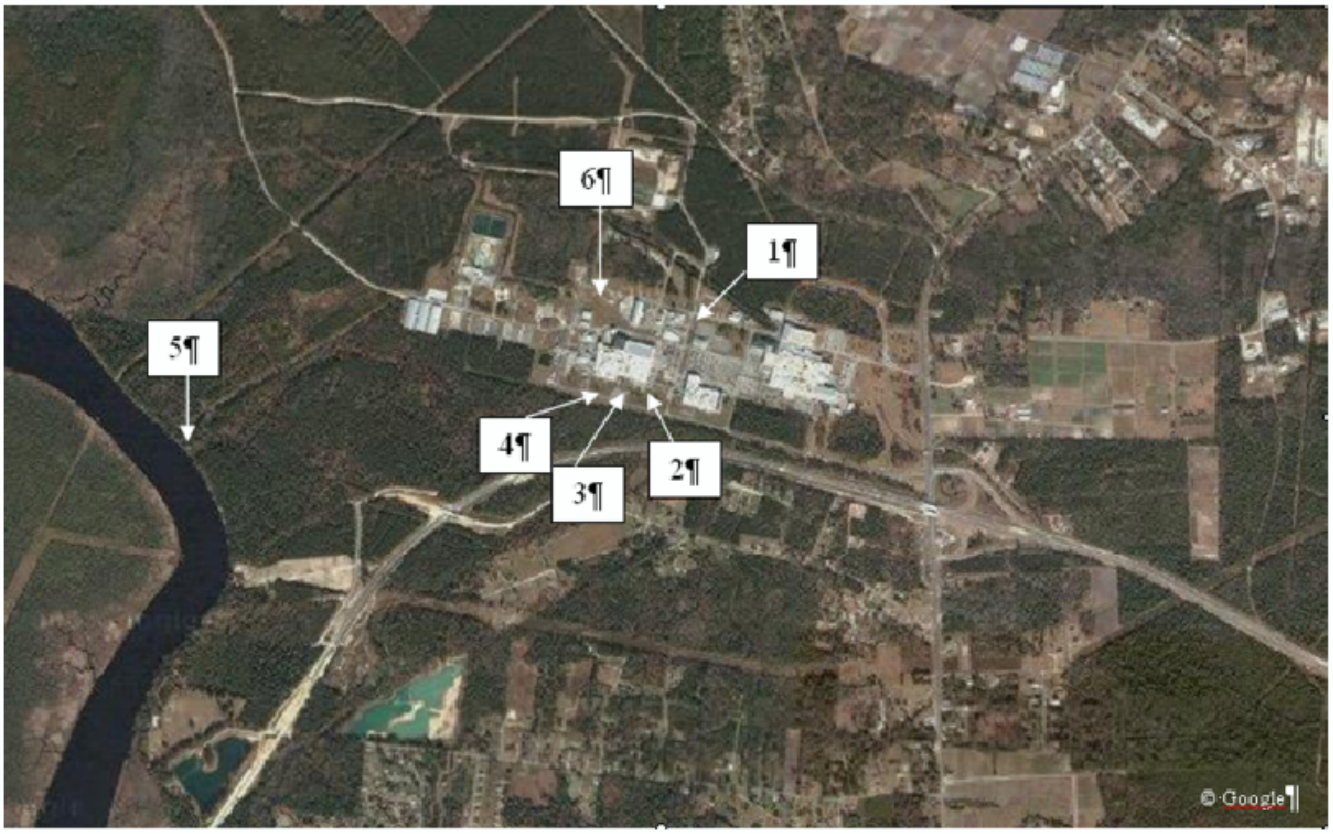


FIGURE 9.3a

Locations of Environmental Soil Sampling Sites (Typical)

Soil Sampling Locations

Soil Samples	
SL01	On Site. South Side of FMO Building. Sample Between Parking Lot and Building.
SL01A	On Site. Approximately 100' east of CP&L substation and 10' north of perimeter fence. Collect sample from bottom of stormwater ditch at a location within reaching distance from concrete retaining wall.
SL02	On Site. Entering the plant from the north highway entrance, the first road on the right. Sample east of the road near gate at the scrap yard. General vicinity of well #5.
SL03	On Site. East side of unpaved roadway to process lagoons at #1 water well. Sample near intersection with Thomas Edison.
SL04	Off-Site. Across NC Hwy 133 from GE Plant. North Carolina Horticulture Research Farm, edge of field east of office building.
*SL06	Off-Site. Go to end of Marathon Ave. that intersects with NC Hwy 133 at the location of St. Stainislaus Catholic Church, approximately 2.5 miles north of GE. Sample at the end of the paved road.
*SL11	Off-Site. South From GE take Rock Hill Road to Walnut Hill Development. Go to intersection of Rockhill Road and Reminisce Road. Turn left on Reminisce Road. Sample approximately 50 feet from the intersection on the east side of Reminisce Road

* State split samples

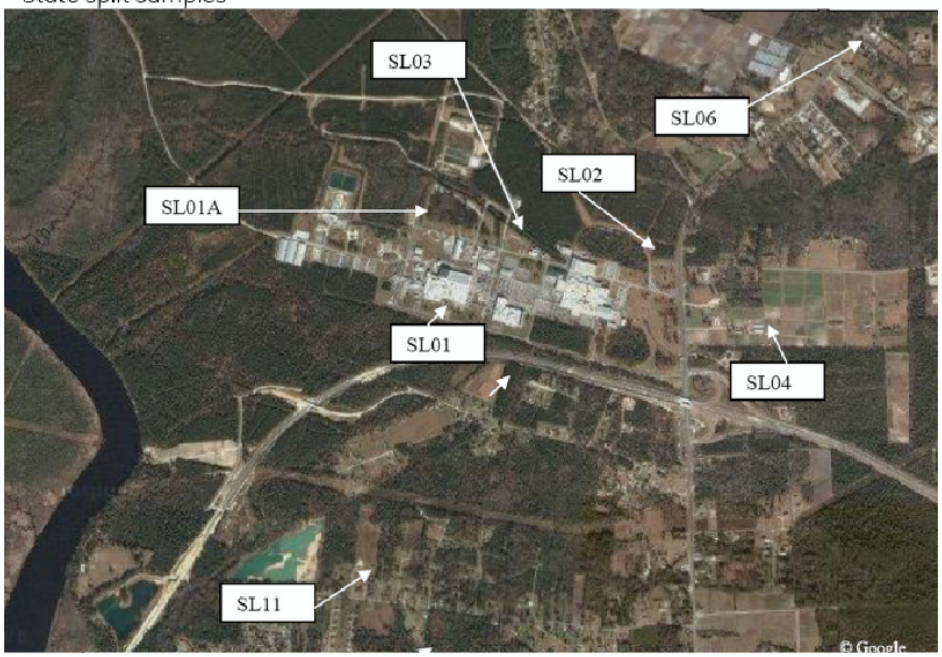


Figure 9.3b
Locations of Ditch & Vegetation Sampling Sites (Typical)

Ditch Samples	
DD05	On Site 10' Upstream of Site Dam, South Side
DD07	On Site Entrance to tidal swamp at bottom of stairs
Vegetation Samples	
*VG06	Off Site Go to end of Marathon Ave. that intersects with NC Hwy 133 at the location of St. Stainislaus Catholic Church, ~2.5 mi. north of GE. Sample at end of paved road.
*VG07	North end of Blue Clay Rd. area
*VG11	Off Site South from GE take Rockhill Road to Walnut Hill Development. Go to intersection of Rockhill Road and Reminisce Road. Turn left on Reminisce. Sample ~50 feet from the intersection on east side of Reminisce Road.
VG98	On Site At GE Ambient Air Station – AAHE
VG99	On Site At GE Ambient Air Station – AASW
Surface Water Samples	
*UPST	Off Site Castle Hayne Boat Landing
*SDAM	On Site Downstream ~10ft of the Site Dam
*GEDK	On Site GE Dock –NE Cape Fear River
Sediment Samples	
*SD02	On Site 10' Upstream of Site Dam, South Side
*SD03	On Site GE Dock – NE Cape Fear River
*SD04	Off Site Castle Hayne Boat Landing

*State split sampling

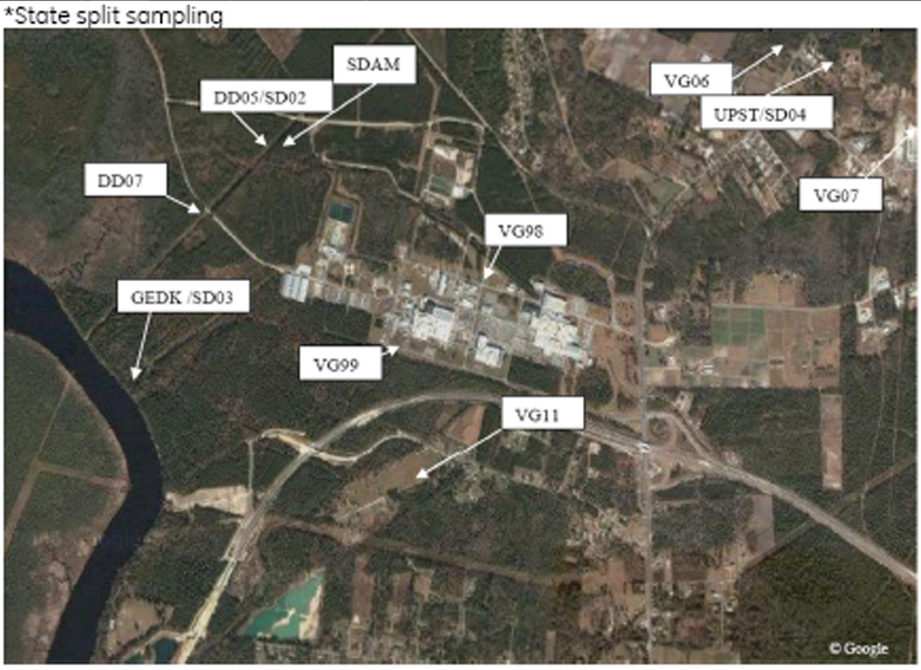


Figure 9.4
Location of Site Dam Sampling

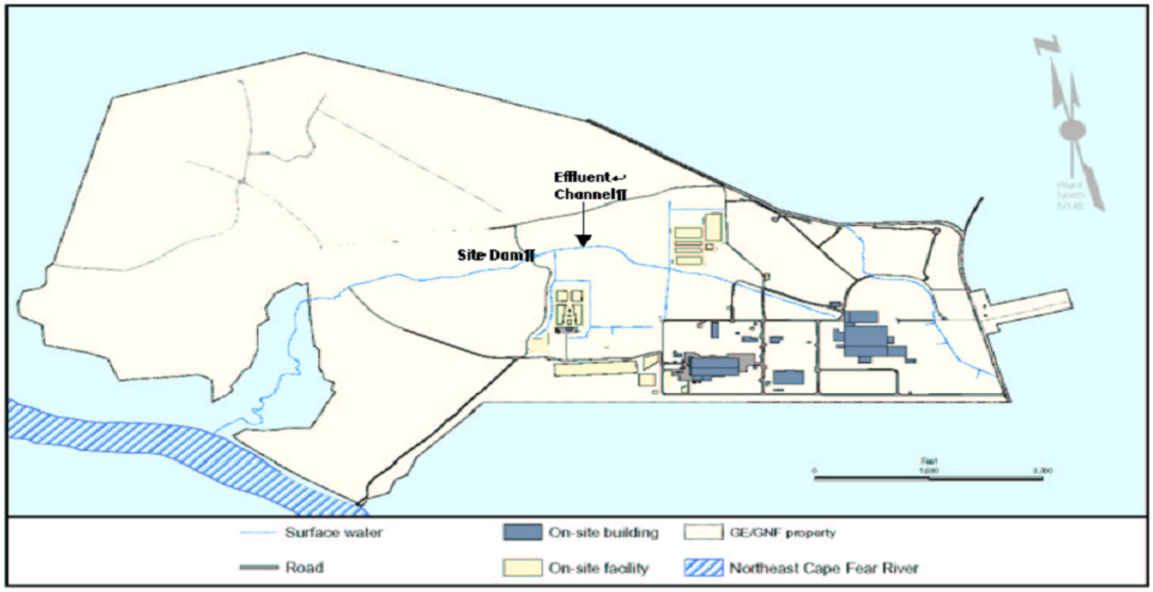


Figure 9.5a
Locations of MW Series Wells Around
Waste Treatment Facility
(Typical)

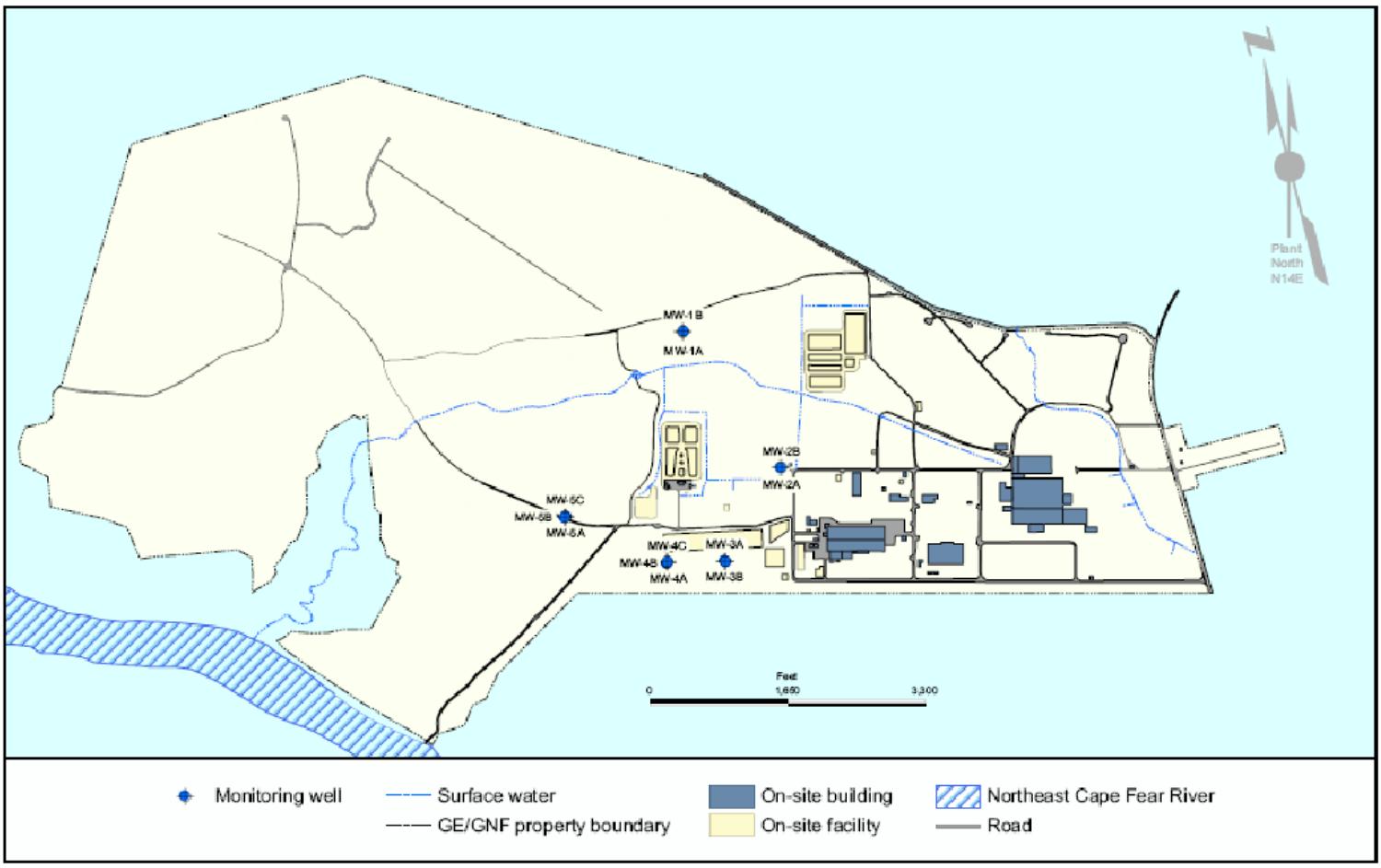
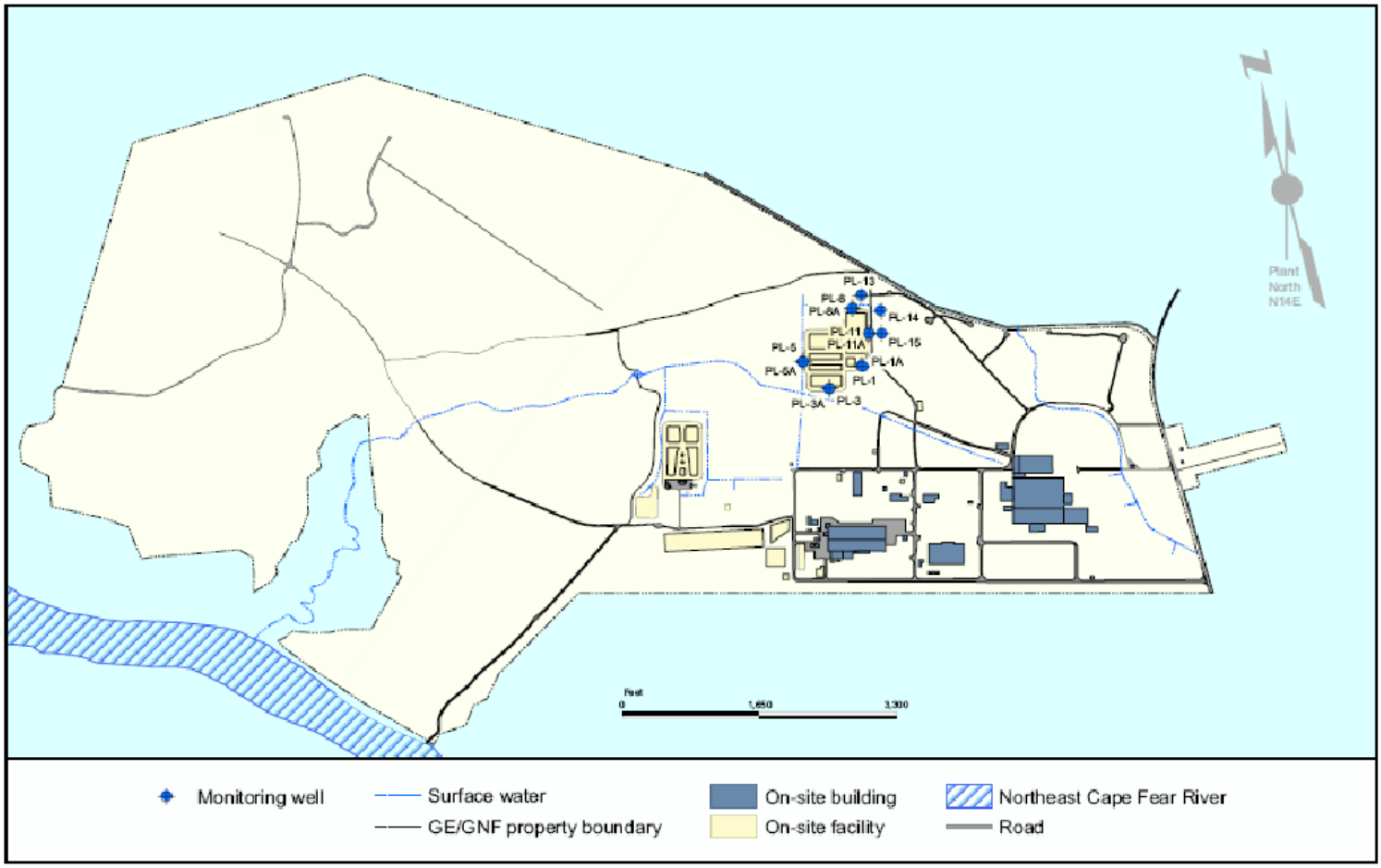
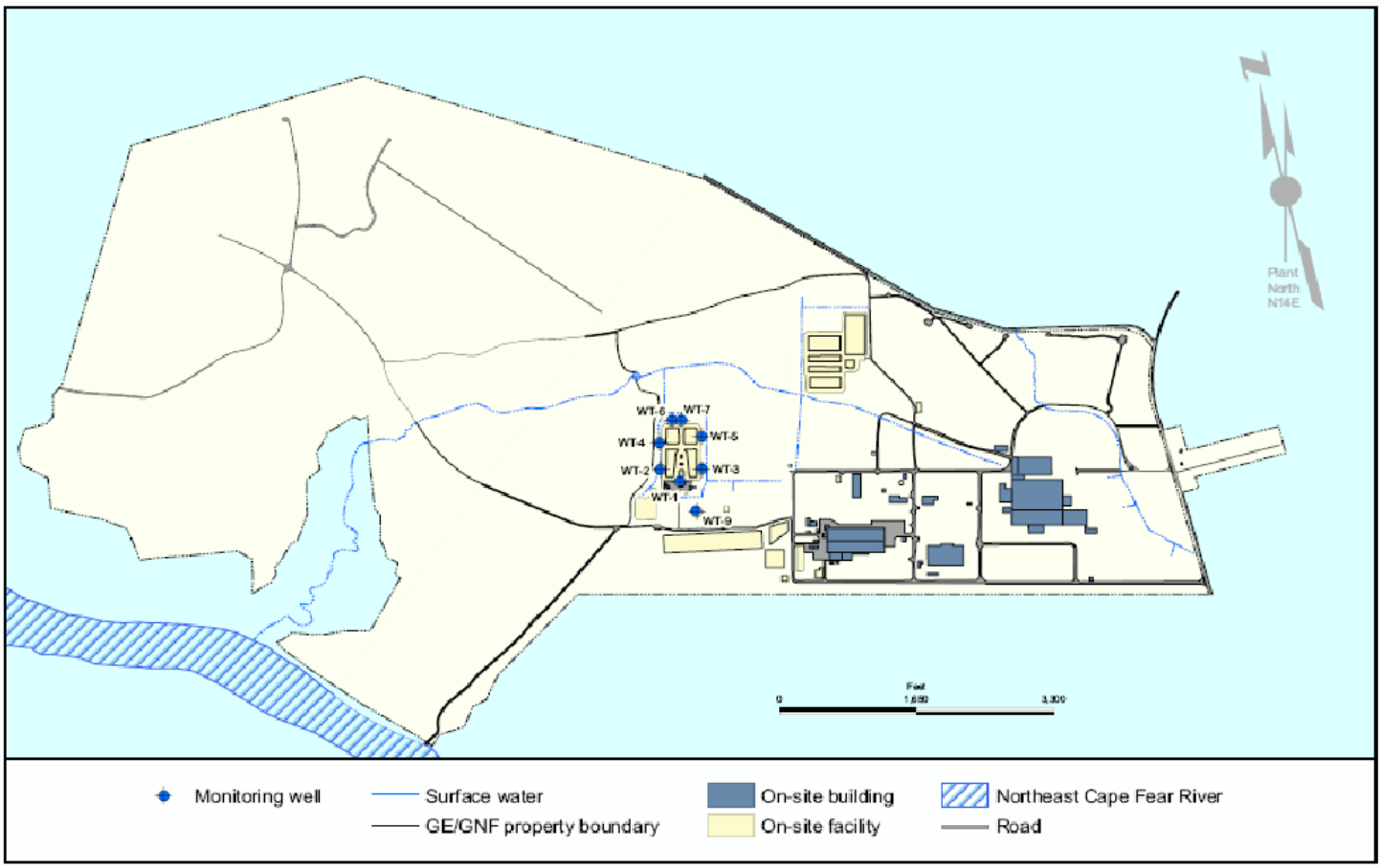


Figure 9.5b
Locations of PL Series Wells Around
Final Process Waste Treatment Facility
(Typical)



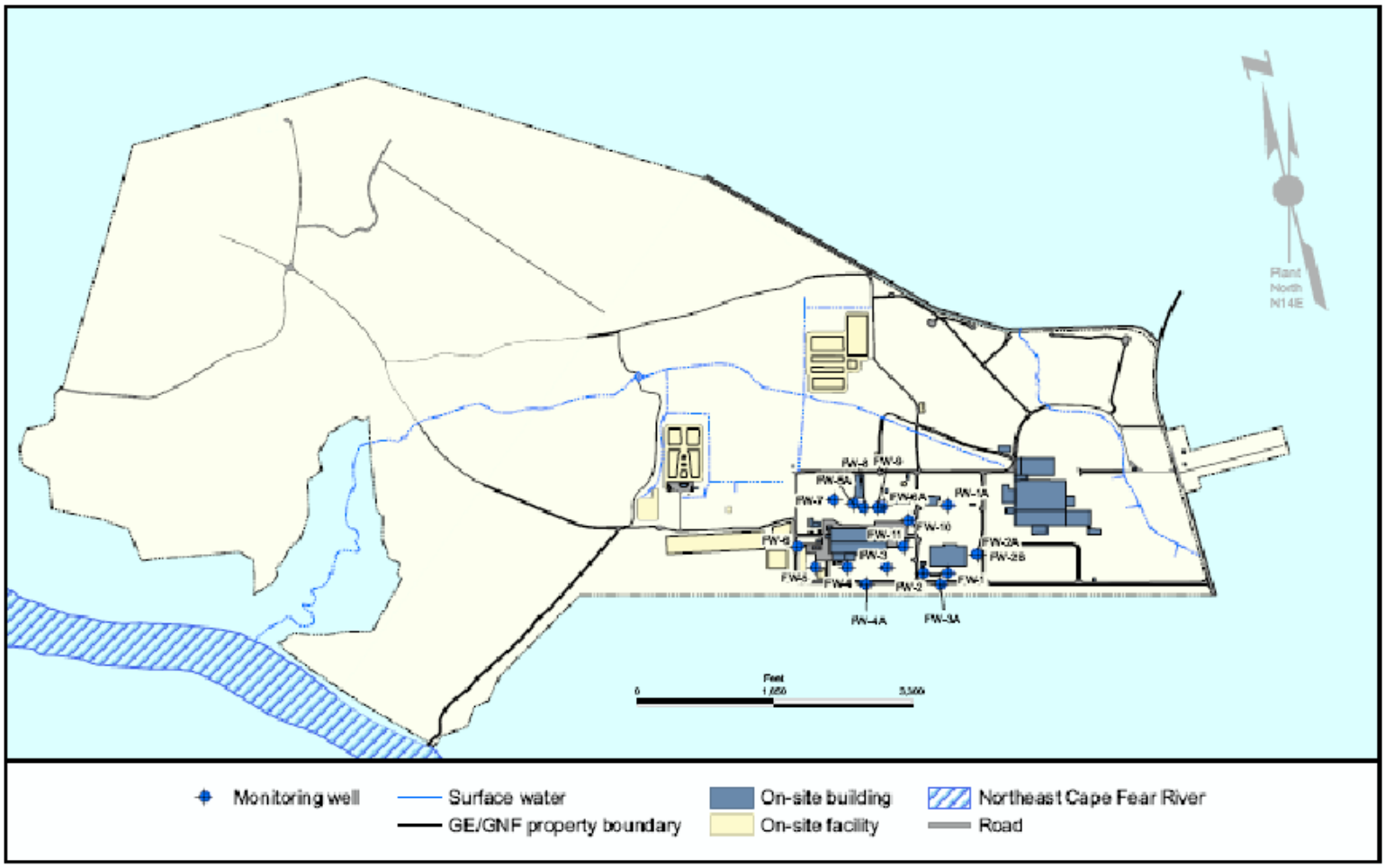
Subject: Sample Collection From Monitoring Wells

Figure 9.5c
Locations of WT Series Wells Around
Waste Treatment Facility
(Typical)



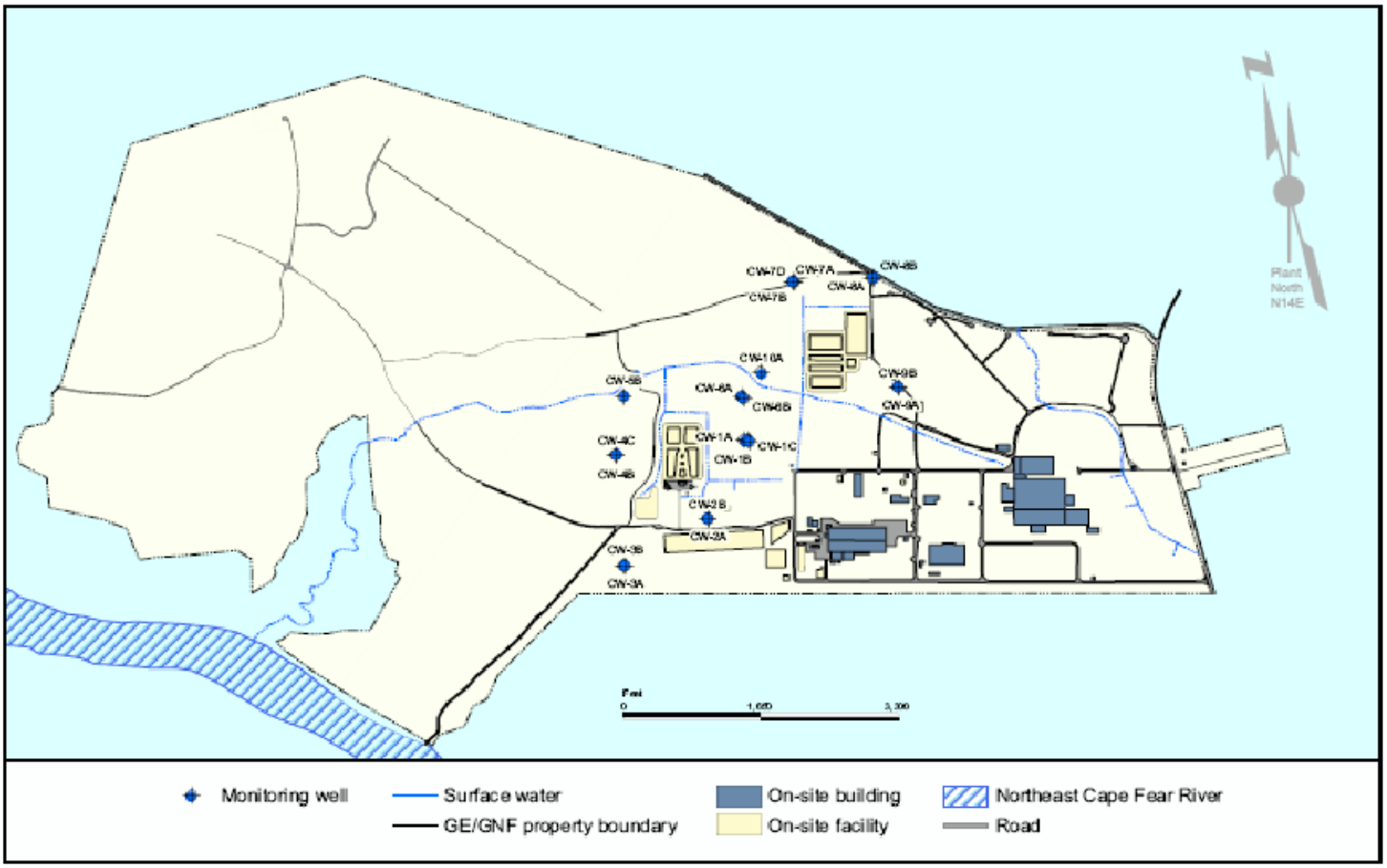
Subject: Sample Collection From Monitoring Wells

Figure 9.5d
Locations of FW Series Wells Around
FMO/FMOX & FCO Areas
(Typical)



Subject: Sample Collection From Monitoring Wells

Figure 9.5e
Locations of CW Series Wells
Around Waste Treatment Facilities
(Typical)
Compliance wells Approximately 500 feet from Facilities



Subject: Sample Collection From Monitoring Well

Figure 9.5f
Locations of FX Series Wells Along Western
Perimeter of FMO/FMOX Buildings
(Typical)

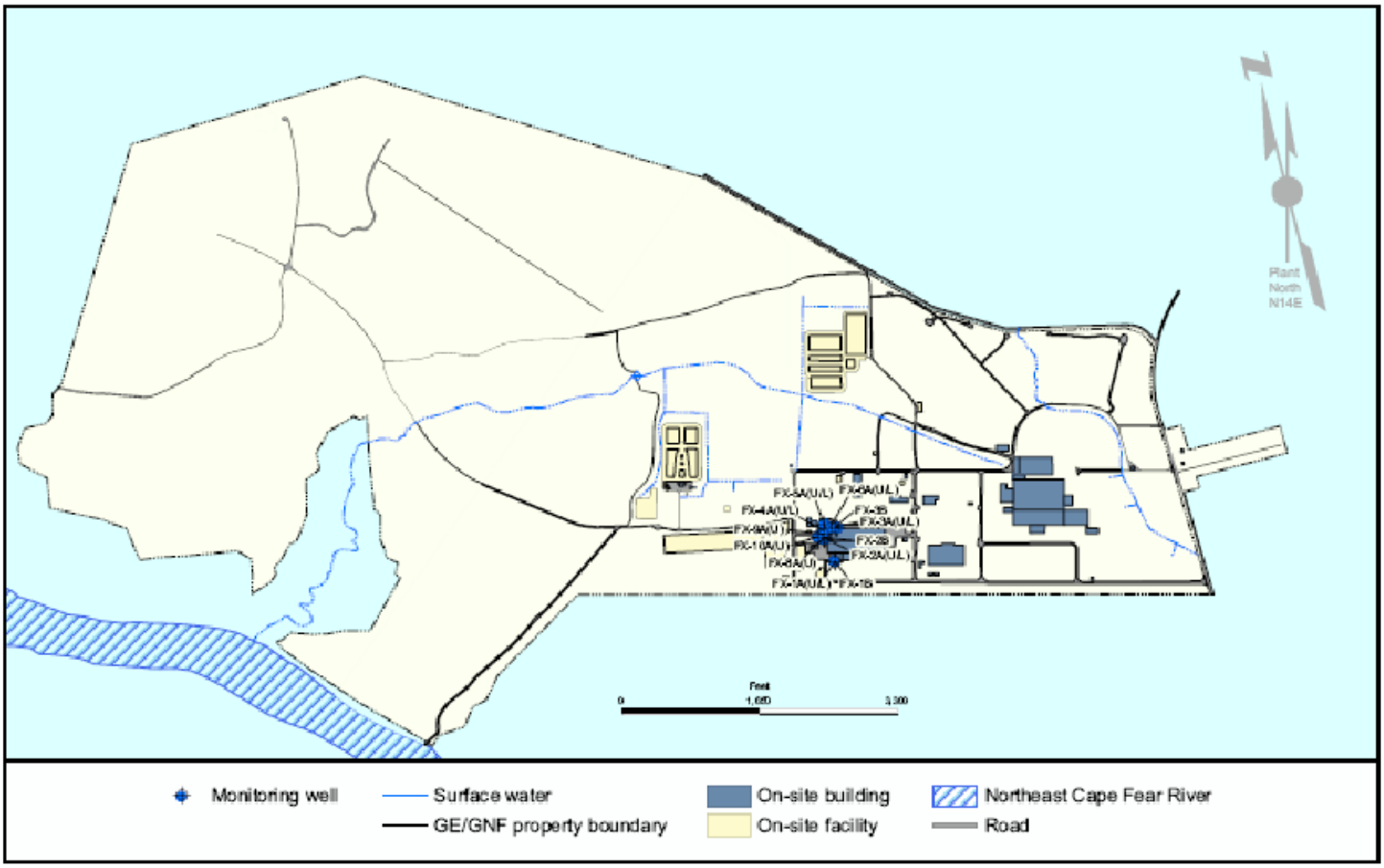
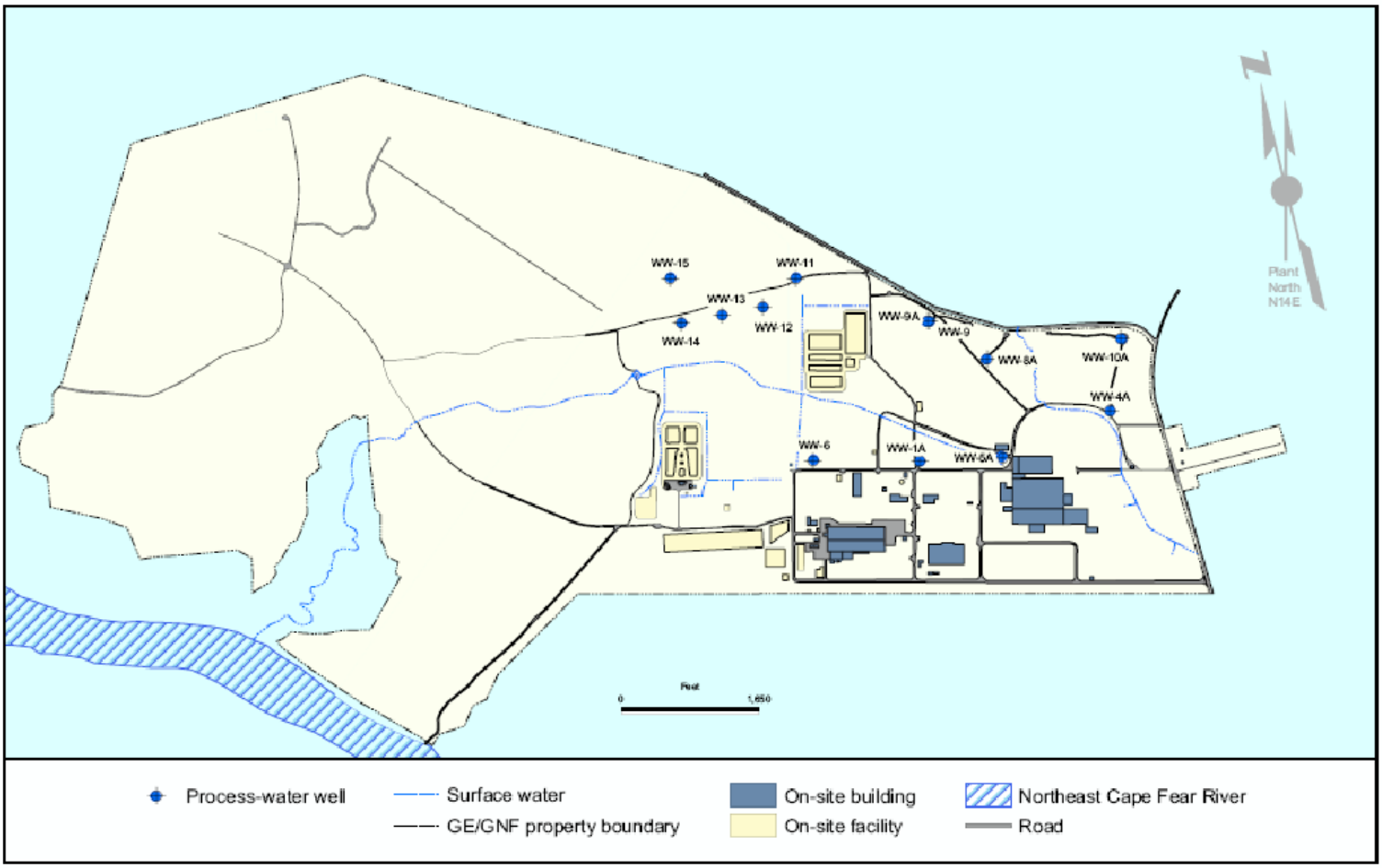


Figure 9.5g
Locations of WW
Series Process Water Supply Wells
(Typical)



CHAPTER 10.0
DECOMMISSIONING

The current Decommissioning and Closure Plan is dated February 2, 2007.

The Decommissioning and Closure Plan for the facility was originally approved by the NRC on December 11, 1981.

At the end of plant life, GNF-A, through a parent company guarantee, shall decommission the facilities and site in accordance with the then current Decommissioning and Closure Plan.

CHAPTER 11.0
MANAGEMENT MEASURES

11.1 MANAGEMENT MEASURES

11.1.1 REASONABLE ASSURANCE

GNF-A commits to apply *Management Measures* on a continuing basis to IROFS for the purpose of providing reasonable assurance that the IROFS are available and able to perform their function when needed.

11.1.2 GRADED APPLICATION OF MANAGEMENT MEASURES TO IROFS

GNF-A applies *Management Measures* in a graded approach based on unmitigated risk as described in Chapter 3 (in particular see Sections 3.5, 3.6 and 3.10).

11.2 CONFIGURATION MANAGEMENT (CM)

11.2.1 CONFIGURATION MANAGEMENT POLICY

GNF-A commits to maintain a formal configuration management process, governed by written, approved practices, and ensures that plant design changes do not adversely impact safety, health, or environmental protection programs at GNF-A. The following items are addressed prior to implementing a change:

- The technical basis for the change
- The impact of the change on safety, health and control of licensed material
- Modifications to existing operating procedures including any necessary training or retraining before operation
- Authorization requirements for the change
- For temporary changes, the approved duration (expiration date) of the change

- The impacts or modifications to the ISA, ISA Summary and any other component of the overall safety program

The configuration management (CM) program ensures that the information used to operate and maintain safety controls is kept current. Safety controls are systems, structures, components and procedures that prevent and/or mitigate the risk of accidents.

The CM program includes the following activities:

- Maintenance of the design information for the plant
- Identification of all IROFS
- Control of information used to operate and maintain the plant
- Documentation of changes
- Assurance of adequate safety reviews for changes
- Periodic comparison assessment of the conformance of specific safety controls to the documentation of plant design basis

11.2.2 DESIGN REQUIREMENTS

Written plant practices define the development, application, and maintenance of the design specifications and requirements. Plant design specifications and requirements are maintained as controlled information. The specific content of the information depends on the age of the design and the requirements in place at the time of design. As a minimum, the information required for safe operation of the facility is available.

11.2.3 DOCUMENT CONTROL

Documented plant practices define the control system, including creation, revision, storage, tracking, distribution and retrieval of applicable information including:

- Hazards Analysis (ISA reference report), ISA Summary including a listing of IROFS
- Operating procedures

- Drawings for safety related systems, structures and components
- Technical specifications and requirements
- Software for safety controls
- Calibration instructions
- Functional test instructions

The documented plant practices describe the responsibilities and activities that maintain consistency between the facility design, the physical facility, and the documentation. They also describe how the latest approved revisions are made available for operations.

11.2.4 CHANGE CONTROL

GNF-A maintains written plant practices describing the configuration management program for controlling design change, including approval to install and operate facility, process, or equipment design changes. These practices stipulate that a trained and approved safety reviewer determine if the applicable ISA is impacted by the facility change. If there is an impact to the ISA, it is identified and the change is flagged for review and approval by an ISA team in accordance with the process described in Chapter 3.

The written plant practices also prescribe controls and define the distinction between types of changes, ranging from replacement with identical designs that are authorized as part of normal maintenance, to new or different designs that require specified review and approval.

11.2.5 ASSESSMENTS

Planned and scheduled internal and independent audits are performed to evaluate the application and effectiveness of management controls and implementation of programs related to activities significant to plant safety. Audits are performed to assure that operations are conducted in accordance with the operating procedures, and to assure that safety programs reflected in the operating procedures are maintained.

11.2.6 DESIGN RECONSTITUTION

The current plant design was reconstituted in accordance with the requirements specified in 10 CFR 70.62.

GNF-A submitted a plan as required by 10 CFR 70.62 (c) (3) (i) and this plan was approved by the NRC on June 11, 2002 (TAC NO. L31607).

GNF-A performed the design reconstitution in accordance with their approved plan and submitted the completed summary required by 10 CFR 70.62 (c) (3) (ii) on October 12, 2004. Periodic updates as required by the regulations (10 CFR 70.72 (d) (2&3)) are submitted to the NRC.

11.3 MAINTENANCE

The purpose of planned and scheduled maintenance of safety controls is to assure that systems are kept in a condition of readiness to perform the planned and designed functions when required.

Area Managers are responsible for assuring the operational readiness of safety controls in their assigned facility areas.

The maintenance function utilizes a systems-based program to plan, schedule, track and maintain records for maintenance activities. Maintenance instructions are an integral part of the maintenance system for maintenance activities. Key maintenance requirements for safety controls such as calibration, functional testing, and replacement of specified components are derived from integrated safety analyses described in Chapter 3.

Maintenance activities generally fall into the categories described in the following sections.

11.3.1 CORRECTIVE MAINTENANCE

Corrective Maintenance refers to situations where repairs, replacements or major adjustments such as re-calibration take place.

GNF-A commits to promptly perform corrective actions to remediate unacceptable performance deficiencies in IROFS.

The maintenance planning and control system provides documentation and records of systems and components that have been repaired or replaced.

When a component of a specified safety control is repaired or replaced, the component is functionally verified via post maintenance testing to assure that it has the capability to perform its planned and designed function when called upon to do so.

If the performance of a repaired or replaced safety control could be different from that of the original component, the change to the safety control is specifically approved under the configuration management program and pre-operationally tested to assure it is likely to perform its desired function when called upon to do so.

11.3.2 PREVENTATIVE MAINTENANCE

Preventative Maintenance refers to activities that are performed as precautions to help ensure that systems remain operational and avoid unexpected failures. Examples of safety controls included for scheduled preventive maintenance are:

- Radiation Measurement Instruments
- Criticality Detection Devices
- Effluent Measurement & Control Devices
- Emergency Power Generators
- Fire Detection and Control Systems
- Pressure Relief Valves
- Air Compressors
- Steam Boilers

11.3.3 SURVEILLANCE/MONITORING

GNF-A utilizes active engineered controls that are integrated into the routine plant operations to the degree practical. In these systems the IROFS are near continuously monitored by the digital control system as a routine part of the operating process. Degradations or failures in these cases result in immediate safe shutdown of the operations.

IROFS associated with passive engineered systems are typically fixed physical design features to maintain safe process conditions. Assurance is maintained through pre-operational audit and periodic verification of effectiveness as prescribed in the ISA process described in Chapter 3, Table 3.7 and includes consideration of the importance of the IROFS as well as quality and reliability information.

IROFS relying on geometry-based controls, where the geometry is subject to undetected change in routine operation, are periodically verified on a schedule commensurate with the potential for change in the parameters of interest.

- Examples of active engineered controls that are integrated into routine plant operations include all IROFS managed by the distributed control system (e.g. PROVOX) or hardwired interlocks.
- Examples of passive engineered IROFS would include process equipment design features such as physical separation of storage fixtures (floor storage fixtures, installed can-conveyor separation); or other process design characteristic (air breaks, overflows, orifice sizing, restricting vessel feeds, hood physical restraints, etc.).
- Examples of geometry-based IROFS would include design control of process equipment physical dimensions (pellet tray dimensions, boat size, container volume, pipe tank ID, annular tank thickness, slab tank thickness) and/or use of neutron absorbers.

11.3.4 FUNCTIONAL TESTING

GNF-A commits to perform post-maintenance testing to verify that the maintenance activity did not adversely affect the functionality of the IROFS associated with the maintenance work.

GNF-A commits to perform functional tests in accordance with written instructions that define the method for the test and the required acceptable results. The results of the tests are also recorded and maintained.

11.4 TRAINING AND QUALIFICATIONS

11.4.1 ORGANIZATION AND MANAGEMENT OF THE TRAINING FUNCTION

Training programs at the GNF-A facility for personnel who perform activities relied on for safety, and are provided through shared responsibility between EHS safety disciplines, Operations and Human Resources functional organizations. Area Managers are responsible for the content and effective conduct of training for operations personnel. Records are maintained on each employee's qualifications, experience, training, and retraining.

Facility administrative procedures establish the requirements for indoctrination and training of personnel performing activities relied on for safety and to ensure that the training program is conducted in a reliable and consistent manner throughout all training areas.

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

11.4.2 FUNCTIONAL AREAS REQUIRING TRAINING

Training is provided for each individual at GNF-A, commensurate with assigned duties (or roles). Training and qualification requirements are met prior to personnel fully assuming the duties of safety-significant positions, and before assigned tasks are independently performed.

Functional areas requiring training may be grouped into one of three broad categories:

- General Employee Training
- Technical Training
- Developmental Training

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

Continuing or periodic retraining courses shall be established when applicable to ensure that personnel remain proficient. Periodic training generally is conducted to ensure retention of knowledge and skills important to facility operations. The training may consist of periodic retraining exercises, instructions, or review of subjects as appropriate to maintain the proficiency of all personnel assigned to the facility.

Chapter 8, Radiological Contingency and Emergency Plan, provides additional information on personnel training for emergency response tasks.

11.4.2.1 General Employee Training

General Employee Training (GET) encompasses those quality assurance, radiation protection, industrial safety, environmental protection, emergency response, and administrative procedures established by facility management and applicable regulations. The industrial safety training for GNF-A complies with applicable section of the Occupational Safety and Health Administration (OSHA) regulations such as 29 CFR 1910 and with 10 CFR 19 (Notices, Instructions, and Reports to Workers: Inspection and Investigations). Continuing training is conducted in these areas as necessary to maintain employee proficiency. All persons under the supervision of facility management (including contractors) must participate in GET; however, certain facility support personnel, depending on normal work assignment, may not participate in all topics of this training. Temporary maintenance and service personnel receive GET to the extent necessary to assure safe execution of their duties. Certain portions of GET may be included in new employee orientation program implementation.

GET topics are listed below:

- General administrative controls and procedures and their use
- Quality Assurance policies and procedures
- Nuclear Safety (Criticality/Radiological)
- Industrial, Chemical, Fire, Health and First Aid
- Emergency Plan and implementing procedures
- Fire protection and fire brigade

- New Employee Orientation
- Environmental Protection

11.4.2.2 Nuclear Safety Training

Training programs are established for the various job functions (e.g., operations, radiation protection technicians, contractor personnel) commensurate with criticality safety and radiation safety responsibilities. Visitors to the airborne radioactivity controlled area are trained in the formal training program or are escorted by trained personnel.

Formal Nuclear Safety training includes information about radiation and radioactive materials, risks involved in receiving low level radiation exposure in accordance with 10 CFR 19.12, basic criteria and practices for radiation protection, nuclear criticality safety principles not verbatim, but in general conformance with applicable objectives contained in ANSI/ANS 8.19 and ANSI/ANS 8.20 national consensus standard guidance.

Training policy requires that employees must complete nuclear safety training prior to unescorted access in the airborne radioactivity controlled area. Methods for evaluating the understanding and effectiveness of the training includes passing an initial examination covering formal training contents and observations of operational activities during scheduled audits and inspections.

Such training is typically performed using computer based training, but may be performed by authorized instructors. Training program contents are reviewed on a scheduled basis by the manager of the criticality safety and radiation safety functions to ensure that training program contents are current and adequate.

Previously trained employees who are allowed unescorted access to the airborne radioactivity controlled area are retrained at least every two years. The effectiveness of the training program is evaluated by either initial training exam or re-training exam. Visitors are trained commensurate with the scope of their visit and/or escorted by trained employees.

11.4.2.3 Industrial, Chemical, Fire, Health and First Aid

Industrial, Chemical, Fire Safety, Health and First Aid safety orientation of new or transferred employees is an important part of establishing the proper safety attitude among plant employees and insuring that they are aware of safety procedures, rules and hazards involving assigned duties. New employee orientation in performance of duties may include, as appropriate, the review of:

- OSHA General Duty Clause
- Employee Responsibilities
- Employer Responsibilities
- General Site Safety Rules
- Hazard Communication Training
- Fire Extinguisher Training
- Emergency Evacuation Procedure
- Job Hazards Analysis (JHA)
- Chemical Job Hazards Analysis (CJHA)
- Lock-Out-Tag-Out Awareness

11.4.2.4 Technical Training

Technical training is designed, developed and implemented to assist facility operations and maintenance personnel in gaining an understanding of the applicable fundamentals, procedures, and technical practices common to a nuclear fuel conversion and fabrication facility. Technical training consists of initial training, on-the-job training, continuing training, and special training, as applicable to assigned technical duties of the job function (or role). This may include, but is not limited to, the following topics:

- On-the-Job Training
- Process Specific Training
- Mechanical Maintenance

- Controls, Instrumentation, Electrical Maintenance
- Chemistry

11.4.2.5 Development Training

Developmental Training is a broad category implemented to assist facility operations supervisory, and management personnel in gaining additional understanding of fundamentals and technical practices common to assigned job duties (or roles). Developmental training typically utilizes internal/external professionals via formal workshop, tutorials, and select training programs.

11.4.3 POSITION TRAINING REQUIREMENTS

Operator training is performance based, and incorporates the structured elements of analysis, design, development, implementation, and evaluation commensurate with assigned duties.

Minimum training requirements are developed for positions whose activities are relied on for safety. Initial identification of job-specific training requirement is based on individual employee experience. Entry-level criteria (e.g., education, technical background, and/or experience) for these positions are contained in position descriptions.

Job-specific training is performance based and established with relevant technical EHS safety discipline and operations leadership to develop a list of qualifications for assigned duties (or roles). Changes to facilities, processes, equipment, or job duties are incorporated into revised lists of qualifications.

11.4.4 BASIS OF TRAINING AND OBJECTIVES

The training program is designed to prepare initial and replacement personnel for safe, reliable, and efficient operation of the facility. Emphasis is placed on safety requirements where human actions are important to safety.

11.4.5 EVALUATION OF TRAINEE LEARNING

Trainee understanding and proficiency is evaluated through observation/ demonstration or oral or written examinations, as appropriate. Such evaluations measure the trainee's skill and knowledge of job performance requirements.

Operator training and qualification requirements are met prior to process safety-related tasks being independently performed or before startup following significant changes to safety controls.

11.4.6 CONDUCT OF ON-THE-JOB TRAINING

On-the-Job training (OJT) is a systematic method of providing the required job related skills and knowledge for a position. This training is conducted in the work environment. Applicable tasks and related procedures make up the OJT/qualifications program for each technical area which is designed to supplement and complement training received through formal classroom, laboratory, and/or simulator training. The object of the program is to assure the trainee's ability to proficiently perform job duties as required for the assigned role. Refer to Section 11.4.3.

Completion of on-the-job training is demonstrated through actual task actions using the conditions encountered during the performance of assigned duties (or roles) including references, tools, and equipment conditions reflecting the actual task to the extent practical.

11.4.7 EVALUATION OF TRAINING EFFECTIVENESS

Periodic evaluations of training program content and requirements are performed to assess program effectiveness. The trainees provide feedback after completion of classroom or computer based training session to provide data for this evaluation. These evaluations identify program strengths and weaknesses, determine whether training content matches current job needs, and determines if corrective actions are needed to improve program effectiveness.

Independent audits of EHS safety disciplines may also be used to provide independent evaluations of overall training program effectiveness (see Section

11.6.5 of this Chapter) as it relates to the ISA program, IROFS implementation, protection of the public, worker, and environment.

Evaluation objectives applicable to the overall organization and management of the GNF-A training programs may include, but are not limited to:

- Management and administration of training and qualification programs
- Development and qualification of the matrix organization
- Design and development of training programs, content, and conduct of training, and trainee examinations / evaluations.
- Training program interface with facility configuration management practices
- Training program assessments and evaluations

11.4.8 PERSONNEL QUALIFICATION

The qualification requirements for key management positions are described in Chapter 2, Organization and Administration. Education, experience, training and qualifications are specified in this chapter.

Qualification and training requirements for operations personnel shall be established and implemented in accordance with internal plant procedures (e.g, Human Resource).

11.4.9 RECORDS

The system established for maintaining records of training and retraining of personnel who perform activities relied on for safety is described in Section 3.8.

11.5 PROCEDURES

Licensed material processing or activities will be conducted in accordance with properly issued and approved management control procedures.

11.5.1 OPERATING PROCEDURES

Area Managers are responsible to assure preparation of written, approved and issued operating procedures incorporating control and limitation requirements established by the criticality safety function, the radiation safety function, the environmental protection function and the chemical and fire safety function. Integrated safety analysis results as described in Chapter 3 are used to identify procedures necessary for human actions important to safety. Operating procedures are initiated and controlled by a configuration management system. Area Managers ensure that operating procedures are made readily available in the work area and that operators are trained to the requirements of the procedures and that conformance is mandatory. Operators are trained to report inadequate procedures, and/or the inability to follow procedures.

Nuclear safety control procedure requirements for workers in uranium processing areas are incorporated into the appropriate operating, maintenance and test procedures in place for uranium processing operations.

The safety program design requires the establishment and maintenance of documented procedures for environmental, health and safety limitations and requirements to govern the safety aspects of operations. Requirements for procedure control and approval authorities are documented. Procedure review for updating frequencies are as follows:

Document	Review Frequency	Reviewing & Approving Functional Manager
Operating Procedures (OPs) {Note: Nuclear Safety Release/Requirement (NSR/R) limitations and requirements are incorporated into OPs}	When changed (1)	Area Manager and Affected EHS Discipline (Radiation, Criticality, Environmental, Industrial (4), or MC&A)
Operating Procedures (OPs)	Every 3 Years (3)	Area Manager and Affected EHS Discipline (Radiation, Criticality, Environmental, Industrial (4), or MC&A)
Nuclear Safety Instructions (NSIs)	Every 2 Years (2)	Radiation & Criticality Safety
Environmental Protection Instructions (EPIs)	Every 2 Years (2)	Environmental Protection

1) The safety awareness portions of these OPs are reviewed and updated by the appropriate environment, health, and safety (EHS) discipline when warranted based on process related facility change requests.

2) Every 2 years means a maximum interval of 26 months.
3) Every 3 years means a maximum interval of 39 months
4) EHS Discipline - Industrial means normal worker safety, chemical safety, and fire and explosion protection.

11.5.2 MANAGEMENT CONTROL PROCEDURES

Licensed material activities are conducted in accordance with management control programs described in administrative and general plant practices approved and issued by cognizant management at a level appropriate to the scope of the practice. These documented practices direct and control activities across the manufacturing functions, and assign functional responsibilities and requirements for these activities. These practices are reviewed for updating at least every two years (26 months).

11.6 AUDITS AND ASSESSMENTS

11.6.1 CRITICALITY, RADIATION, CHEMICAL AND FIRE SAFETY AUDITS

Representatives of the criticality safety function, the radiological safety function, and the chemical and fire safety function conduct formal, scheduled safety audits of fuel manufacturing and support areas in accordance with documented, approved practices. These audits are performed to determine that operations conform to criticality, radiation, and chemical and fire safety requirements.

Criticality and radiological audits are performed quarterly (at intervals not to exceed 110 days) under the direction of the manager of the criticality safety function and the manager of the radiation safety function. Chemical and fire safety audits are performed under the direction of the chemical and fire safety function manager. Personnel performing audits do not report to the production organization and have no direct responsibility for the function and area being audited.

Audit results are communicated in writing to the cognizant Area Manager and to the manager of the environment, health & safety function. Required corrective actions are documented and approved by the Area Manager, and tracked to completion by the environment, health & safety function.

Radiation protection personnel within the radiation safety function conduct weekly nuclear safety inspections of fuel manufacturing and support areas in accordance

with documented procedures. Inspection findings are documented and sent to the affected Area Manager for resolution.

Records of the audit or inspection, instructions and procedures, persons conducting the audits or inspections, audit or inspection results, and corrective actions for identified violations of license conditions are maintained in accordance with procedural requirements for a minimum period of three years.

11.6.2 ENVIRONMENTAL PROTECTION AUDITS

An audit schedule of the environmental protection program is developed by the environmental protection function on an annual basis. Audits are conducted in accordance with documented practices to ensure that operational activities conform to documented environmental requirements.

Personnel under the direction of the manager of the environmental protection function perform the environmental protection audits. Personnel performing the audits do not report to the production organization and have no direct responsibility for the function and area being audited.

Audit findings are communicated to the cognizant Area Manager, who is responsible for nonconformance corrective action commitments in accordance with documented practices. The manager of the environmental protection function or delegate is responsible for resolution follow-up for identified nonconformance. Audit results in the form of corrective action items are reported to the GNF-A Facility Manager and staff for monitoring of closure status.

11.6.3 INDEPENDENT AUDITS

GNF-A commits to perform triennial independent audits of its safety program elements (radiation protection, criticality safety, chemical safety, fire and explosion protection, industrial safety and environmental protection). The audit team will consist of appropriately trained and experienced individuals who are not involved in the routine performance of the work or program being audited. The audit scope includes compliance to procedures, conformance to regulations and the overall adequacy of the safety program.

Audit results are reported in writing to GNF-A's Facility Manager, the Area Managers, the manager of the radiation safety function, and the manager of the criticality safety function, as appropriate. The findings of the audit are assigned to

the appropriate safety function or Area Managers. The assigned responsible individual takes the necessary steps to investigate the finding and identify appropriate corrective actions to address and correct the finding.

The corrective actions resulting from the audit are entered into the management tracking system and reported and tracked to completion by the Facility Manager.

11.6.4 FIRE SAFETY

11.6.5 WORKER CONCERNS

GNF-A commits to maintain a safety conscious work environment. All workers are encouraged to report potentially unsafe conditions to their supervisor, management or the safety organization. Reported concerns are promptly investigated, assessed and resolved.

11.7 INCIDENT INVESTIGATIONS

GNF-A commits to maintain a system to identify, track, investigate and implement corrective action for abnormal events (unusual incidents). The system includes the following requirements and features:

- The system operates in accordance with written procedures
- Abnormal events are documented, tracked and reported to the Area Managers, the safety functions and facility management
- Abnormal events associated with IROFS or their associated management measures are specifically identified
- Each event is considered in terms of regulatory reporting criteria
- Events are considered in terms of severity where precursor events are considered unusual events (UIR) and events concerning compliance with regulations or license conditions are considered potential non-compliances (PNC)
- All UIRs require investigation, a determination of root or most probable cause and the identification of required corrective action
- More significant UIRs and PNCs require a formal, systematic determination of root cause (typically using an independent, qualified team), definition of

corrective actions and a higher level management review and approval of the investigation and corrective actions

- Monthly reports covering UIRs and PNCs and their status are issued to the Facility Manager, Area Managers and the safety functional managers
- Events are graded for the purpose of an ongoing management evaluation of facility performance and used as one element in driving safety culture focus
- Records of the events and the documented evidence of closure are maintained for a minimum of three years
- UIR and PNC information is used where appropriate when performing ISAs

11.8 RECORDS MANAGEMENT

Records appropriate for integrated safety analyses, IROFS, the application of management measures to IROFS, criticality and radiation safety activities, training/retraining, occupational exposure of personnel to radiation, releases of radioactive materials to the environment, and other pertinent safety activities are maintained in such a manner as to demonstrate compliance with license conditions and regulations.

Records of integrated safety analyses and the identification of IROFS are retained during the conduct of the activities analyzed and for six months following cessation of such activities to which they apply or for a minimum of three years.

Records of criticality safety analyses are maintained in sufficient detail and form to permit independent review and audit of the method of calculation and results. Such records are retained during the conduct of the activities and for six months following cessation of such activities to which they apply or for a minimum of three years.

Records associated with personnel radiation exposures are generated and retained in such a manner as to comply with the relevant requirements of 10 CFR 20. The following additional radiation protection records will be maintained for at least three years:

- Records of the safety review committee meetings
- Surveys of equipment for release to unrestricted areas
- Instrument calibrations

- Safety audits
- Personnel training and retraining
- Radiation work permits
- Surface contamination surveys
- Concentrations of airborne radioactive material in the facility
- Radiological safety analyses

Records associated with the environmental protection activities described in Chapter 10 are generated and retained in such a manner as to comply with the relevant requirements of 10 CFR 20 and this license.

11.9 OTHER QA ELEMENTS

GNF-A performs a broad spectrum of work that requires the application of QA measures. This includes work-requiring conformance to 10 CFR 50, Appendix B, 10 CFR 71, Subpart H as well as certain aspects of 10 CFR 70. As a result of these over arching quality requirements, GNF-A's management system is structured to provide a full scope of QA elements and apply them as appropriate.

With regard to 10 CFR 70, particularly the identification and maintenance of IROFS and the management measures (discussed in this Chapter) that assure the availability of the IROFS to perform their intended function when required, the following information outlines the classic QA Elements and summarizes the manner in which they are applied for the operations. The following assurance elements are applied to IROFS and the management measurements at GNF-A:

- Organization – GNF-A operates to a documented organizational structure in which responsibility and authority is clearly identified
- Program – GNF-A operates to written policies, procedures and instructions.
- Design Control – GNF-A policies and procedures outline a program to provide design control for IROFS including the management measures necessary to assure their successful operation (see CM program Section 11.2).
- Procurement Documentation Control – GNF-A policies and procedures require the definition of procurement specifications, review and approval of procurement to assure they are compatible with regulatory requirements

- Instructions, Procedures, and Drawings – GNF-A uses instructions, written procedures and drawings to document configuration, processes and methods for doing work
- Document Control – GNF-A implements document control as described here in Chapter (11.5).
- Control of Purchased Materials, Equipment, and Services – GNF-A procedures require that purchased materials, equipment or services be secured from appropriately qualified vendors and that as appropriate vendor certifications or in-house dedication of the items or work are provided
- Identification and Control of Materials, Parts, and Components
- Control of Special Processes – GNF-A procures materials from qualified vendors to documented specifications that include where necessary control of special processes. Internally the change control process, Production Tests, Engineering Evaluation Tests, Radiation Work Permit and Temporary Operating Procedure routines control special situations.
- Internal Inspections – GNF-A uses pre-operational audits for IROFS to verify that parts, configuration and operations are as intended.
- Test Control – GNF-A implements a functional test program for IROFS as defined in this Chapter.
- Control of Measuring and Test Equipment – GNF-A maintains measuring and test equipment in accordance with procedures.
- Handling, Storage, and Shipping Controls –GNF-A process for procuring materials include where appropriate handling and shipping controls to ensure the validity of the items received. In addition where shelf life is important controls are implemented to ensure these limits are implemented for the item.
- Inspection, Test, and Operating Status – Where the ISA and associated IROFS require this type of marking; items are so marked and maintained.
- Control of Nonconforming Materials, Parts, or Components - GNF-A maintains a non-conforming materials program.
- Corrective Action – GNF-A procedures for investigating the failure of IROFS require the definition of root cause and corrective action.
- Records – Where specific actions are required, GNF-A maintains records to demonstrate the action has been completed.
- Audits – GNF-A provides audits as defined in this Chapter.