Attachment 3

Chapter 2
Organization and Administration

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)

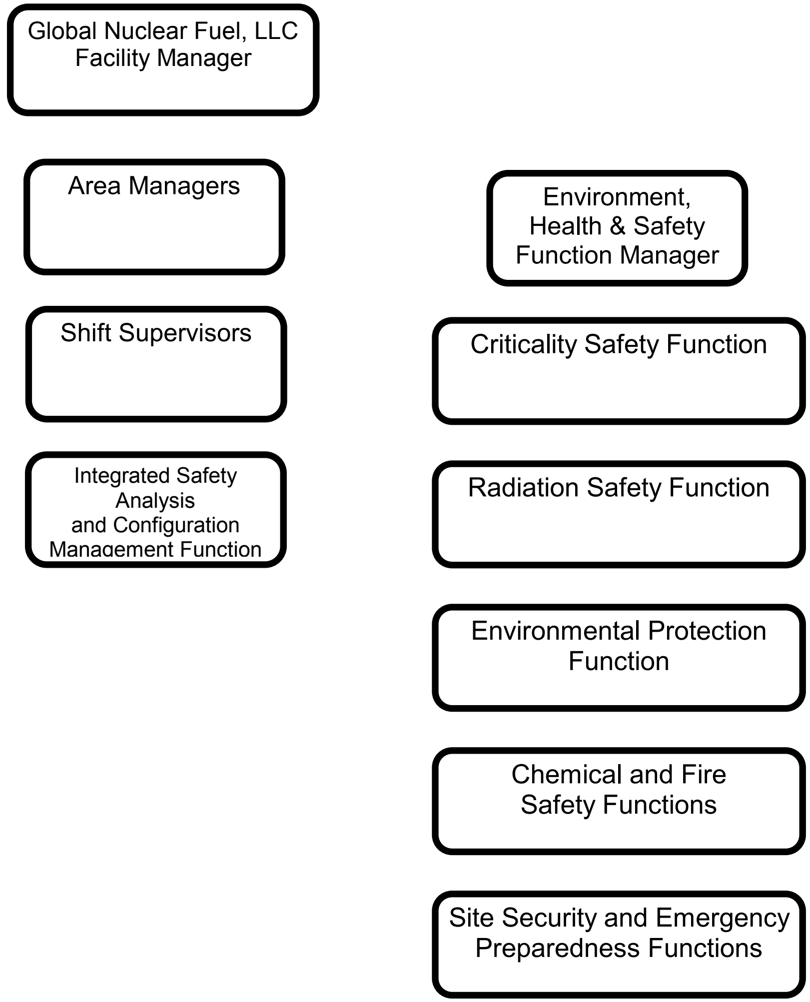
Responsibilities, authorities, and interrelationships among the GNF-A organizational functions with responsibilities important to safety are specified in approved position descriptions and in documented and approved practices.

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
(Typical)



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 are one of the following three options:

Option 1, a combination of:
- BS/BA degree in a technical field;
- Two years supervisory or technical experience in a nuclear, manufacturing or other technical field; and,
- One year of supervisory or technical experience in nuclear operations.

Option 2, a combination of:
- BA (non-technical) / AA degree;
- Three years supervisory or technical experience in a nuclear, manufacturing or other technical field; and,
- One year of supervisory or technical experience in nuclear operations.

Option 3, a combination of:

- High School diploma;
- Five years supervisory or technical experience in a nuclear, manufacturing or other technical field; and,
- Two years of supervisory or technical experience in nuclear operations.

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 are a High School diploma and one of the three qualifications outlined below.

- One year supervisory experience in a nuclear, manufacturing or technical field
- Two years of technical experience in nuclear or manufacturing operations, or
- Three years of operator experience in nuclear operations

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
- Oversight of the respiratory protection program

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 Functions

The chemical and fire safety functions are administratively independent of the production responsibilities and have the authority to shutdown operations with potentially hazardous health and safety conditions. These functions 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 managers of the chemical and fire safety functions are a BS or BA degree or equivalent and two years of experience in related assignments.

2.2.1.9 Site Security and Emergency Preparedness Functions

The site security and emergency preparedness functions are 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 of the manager of the site security and emergency preparedness functions are a BS or BA degree , or equivalent and 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 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.

Attachment 4

Chapter 4
Radiation Safety

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 maintain worker exposures ALARA by proper use of procedures, equipment, and process design.

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 and approved by radiation safety supervision prior to issuance.

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 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 instrumentation is 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 instrumentation 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. The radiation safety function makes a determination to issue personnel dosimetry to individuals based on work area surveys, occupancy time, or other exposure information such as area monitor results. 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, including written procedures for air sampling sufficient to identify the potential hazard, proper equipment selection, maintenance and testing, dose estimation; and surveys or bioassays, as necessary, to evaluate actual intakes. 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 and as documented in written procedures.

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 annually 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/hr - 2000 mR/hr	Area Dose Rate Survey, Shipment Survey
GM High Range	0.1 mR/hr - 1000 R/hr	Emergency Monitoring
Ion Chamber - Low Range	0.1 mR.hr - 10 R/hr	Area Dose Rate Survey, Shipment Survey
Ion Chamber - High Range	1 mR/hr - 1000 R/hr	Emergency Monitoring
ALPHA SURVEY METERS	50 cpm - 2 x 10^6 cpm	Direct Area Equipment Surveys
NEUTRON METERS	0.5 mR/hr - 5 R/hr	Special Dose Rate Surveys
PERSONAL CONTAMINATION MONITORS	N/A	Personal 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

Attachment 5

Chapter 5
Nuclear Criticality Safety

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 section 4.2.2 of the 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 process that has accident sequences that could result in an inadvertent nuclear criticality, a defense of one or more system parameters provided by at least two independent controls is documented in the criticality safety analysis (CSA), which is reviewed and enforced.

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

- all new and existing processes, facilities or equipment that process, store, transfer or otherwise handle fissile materials, and
- any change in existing 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 within 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

Information regarding General Organization and Administration is described in Chapter 2.

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 training 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.

Audits and assessments 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 Accident Alarm System (CAAS) Design and Performance Requirements

The criticality accident alarm system (CAAS) radiation monitoring unit detectors are uniform throughout the facility for the type of radiation detected, the mode of detection, the alarm signal, and the system dependability (e.g. concurrent response of two or more detectors to initiate the alarm). Also, individual 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 CAAS 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 CAAS 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.

In the event that CAAS coverage is lost in an area, compensatory measures such as limiting personnel access, halting special nuclear material movement or installing temporary detection equipment are used as an interim measure until the system is restored.

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. This regulatory & compliance tracking system is used by the Operations, Safety and Licensing 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 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. The 'moderation safety factor' is the ratio of the safety moderation limit to the process moderation limit. The moderation safety factor will normally be three or higher, but never less than two.

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.

When precipitating agents are used in systems where concentration is utilized as a criticality control parameter, controls are in place to ensure that the concentration level is maintained within defined limits for the system. Precautions are taken to protect against inadvertent introduction of precipitation agents in accordance with the configuration management program described in Chapter 11.

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.

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-sectional data set) used for the evaluation of nuclear criticality safety must be demonstrated and sufficiently documented in a validation report according to written procedures to allow understanding of the methodology by a qualified and knowledgeable individual. The validation of the computer code will be performed consistent with the guidance outlined in section 4.3 of ANSI/ANS-8.1-1998 and include the code 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 significant 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.

If parameters needed for anticipated applications is beyond the range of the critical benchmark experiments, the Area of Applicability (AOA) may be extended by extrapolation using the established trends in the bias. In general, if the extrapolation is too large, new factors that could affect the bias may be introduced as the physical phenomena in the system or process change. For conservatism, the extrapolation should be based on the following rules:

- The extrapolation should not result to a large underlying physics or neutronic behavior change in the anticipated application. If there is a rapid or non-conservative change in bias in the vicinity of the AOA range endpoints of a trending parameter, extra safety margin should be needed. Otherwise, critical experiments should be added for further justification.

- Statistical methods should be used to ensue that the extrapolation is not large. The leverage statistic, a measure of the distance between the extrapolation

point x for a predication and the mean of trending parameter values in the critical benchmark data set can be used to determine if an extrapolation using the regression model is acceptable when making predications at x.

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 management 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-validation 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.

Attachment 6

Chapter 7
Fire Safety

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 minimum 1.5-hour 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.

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.

Attachment 7

Chapter 8
Radiological Contingency and Emergency Plan

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 December 7, 2012 and was approved by NRC on January 18, 2013.

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.

Attachment 8

Chapter 10
Decommissioning

CHAPTER 10.0
DECOMMISSIONING

The current Decommissioning Funding Plan is dated December 14, 2012.

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.

Attachment 9

Chapter 11
Management Measures

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 (See Section 3.5).

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 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)
- Material Safety Data Sheets (MSDS)
- 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 11.8.

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)
Common Procedures (CPs) and Work Instructions (WIs)	Every 2 Years (2)	Radiation & Criticality Safety, Environmental Protection, 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

Fire protection audits and inspections include:

- Internal formal quarterly audits, supplemented by routine informal inspections.

- Independent auditors perform scheduled fire protection, prevention and inspections of the facility. Action plans are developed to address findings arising from such inspections.

These audits and inspections verify that ignition sources and combustibles are properly controlled.

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 and compliance with regulations or license conditions.
- All condition reports require investigation, a determination of root or most probable cause and the identification of required corrective action
- More significant condition reports 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 condition reports 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
- Condition report 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.