Global Nuclear Fuel

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Director, Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001
Attn: Document Control Desk

Subject: NRC License SNM-1097 – License Renewal Application Chapter Updates

Reference: 1) SNM-1097, Docket 70-1113

Dear Sir or Madam:

The Global Nuclear Fuel – Americas L.L.C. (GNF-A) facility in Wilmington, North Carolina hereby provides administrative updates to fthree Chapters of our approved SNM license renewal application (Reference 1).

The administrative updates to Chapters 3, 4, and 10 have been internally reviewed and documented as changes to license commitments that do not require prior NRC approval in accordance with SNM 1097 Section 1.3.1.2 and 10 CFR 70.72

A description and reason for the change in each Chapter is provided in Attachment 1 to this letter.

Attachments 2 through 4 provide updates to Chapter 3, "Integrated Safety Analysis", Chapter 4 "Radiation Safety" and Chapter 10, "Decommissioning". The revised sections are identified with a vertical line in the right hand margin on each revised page.

Please contact me on (910) 819-5950 if you have any questions or would like to discuss the request.

Sincerely,

Scott P. Murray, Manager
Facility Licensing

Attachments: 1) SNM-1097 Change Table Summary
2) SNM-1097 Chapter 3
3) SNM-1097 Chapter 4
4) SNM-1097 Chapter 10

cc: TD Naquin, USNRC NMSS
T. Grice, USNRC RII
SPM 19-011

Attachment 1

SNM-1097 License Application Change Table 4/16/19		
Section	**Description of Change**	**Reason for Change**
Chapter 3 -INTEGRATED SAFETY ANALYSIS		
Section 3.3.1- Selecting the Analysis Method	Last sentence add a space between “inputto” to show “input to”	Correct Typo
Section 3.4 Conducting the Quantitative Analysis	Next to last sentence of second paragraph replaced “…as described in Subsection 3.5.1…” with “….in accordance with internal procedures”.	Administrative Update
Section 3.5.1- ISA Change Management	Correct two spacing issues. Third paragraph change “required.Changes” to “required. Changes” and last paragraph remove extra space between “potential changes”	Correct Typo
Section 3.5.2.2 Quantitative Risk Analyst	Changed “approvers” to “authors”.	Administrative Update
Chapter 4 – RADIATION SAFETY		
Section 4.5.1 Surveys	In the survey frequency table, changed “Action Limit” to “Action Level” to align it with the text.	Clarification
Chapter 10 – DECOMMISSIONING	First sentence – added “…and as it may be further revised in accordance with 10 CFR 70.25(e).	Administrative Update to allow for future revisions

CHAPTER 3.0
INTEGRATED SAFETY ANALYSIS

3.0 Introduction

The Integrated Safety Analysis (ISA) identifies process hazards associated with the fuel manufacturing facility operated at Global Nuclear Fuel –Americas LLC (GNF-A), located in Wilmington, North Carolina.

The analysis determines potential accident sequences and provides reasonable assurance that adequate controls are in place to prevent and/or mitigate accidents in accordance with the performance requirements of 10 CFR Part 70.61. Items Relied On For Safety (IROFS) are identified for each accident sequence that could fail to meet the performance requirements of 10 CFR 70.61.

The primary scope of the analysis focuses on consideration of the effects of relevant hazards on radiological safety, prevention of nuclear criticality accidents, or chemical hazards directly associated with NRC-licensed radioactive material.

The ISA covers all major equipment associated with the fuel manufacturing facility. Utilities (e.g., cooling water, plant air) supporting the facility were considered only to the extent that (1) failure or improper operation of the utility systems could cause significant hazards in the facility or (2) upsets in the facility and manufacturing process systems could cause significant hazards in the utility systems.

Facility operating experience, including unusual event and incident reports, is considered in the process hazards analysis of the fuel manufacturing facility and its associated process systems. Consideration is also given to related nuclear operations at other fuel fabrication facilities. These allow the team to consider how additional problems might occur and whether similar incidents could occur again.

3.1 Integrated Safety Analysis

Integrated Safety Analysis is a systematic analysis to identify facility and external hazards and their potential for initiating accident sequences, the potential accident sequences, their likelihood and consequences, and the IROFS. Figure 3.1 provides an overall process flow diagram of the ISA methodology applied to licensed activities.

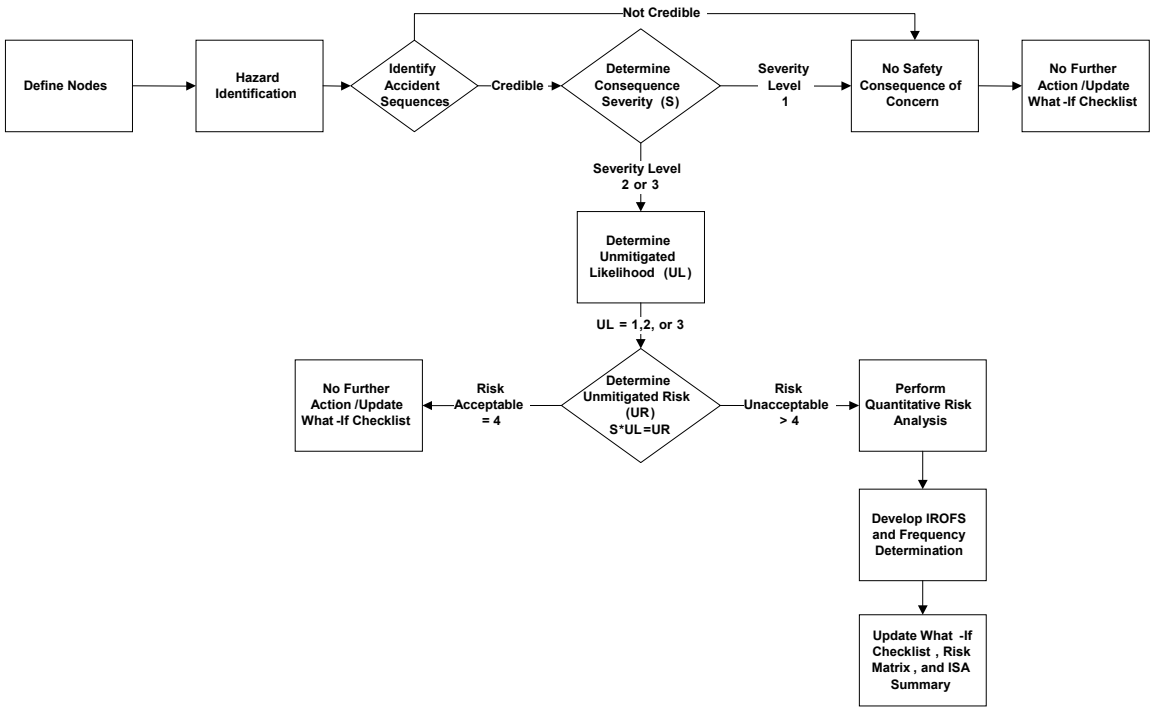

Figure 3.1 – ISA Process Flow Diagram (Typical)

3.2 Hazards and Risk Evaluation Methods used at GNF-A

To identify hazards and evaluate accident sequences, GNF-A in general uses methodologies identified in the following references: NUREG-1520 Rev. 1 (May 2010), *Standard Review Plan (SRP) for the Review of a License Application for a Fuel Cycle Facility,* NUREG-1513 (May 2001), *Integrated Safety Analysis Guidelines Document, Guidelines for Hazard Evaluation Procedures, Second Edition*, and *Layer of Protection Analysis, Simplified Process Risk Assessment.*

Several methods, which are routinely used in industry, are approved for use at GNF-A. Approved methods include the following:

- Checklist
- What-If Analysis
- Hazards and Operability Analysis (HAZOP)
- Failure Modes and Effects Analysis (FMEA)
- Fault Tree Analysis
- Event Tree Analysis
- Human Reliability Analysis
- Layer of Protection Analysis (LOPA)

One or more of these methods may be used to qualitatively analyze the hazards of the process or operation being studied. Methods such as HAZOP, what-if, checklist, or a combination of two or more of these methods are used to conduct the process hazard analyses.

Methods such as event tree analysis, fault tree analysis, human reliability analysis, and LOPA are approved for quantitatively determining the risks of a process or operation. Other methods consistent with industry or regulatory guidance, including semi-quantitative methods, may also be used. These methods can be used to determine the overall likelihood of an accident sequence previously identified during the process hazard analysis.

3.3 Conducting the Process Hazard Analysis

The focus of the process hazard analysis is to identify the hazards associated with the fuel manufacturing facility, identify credible accident sequences and their causes, and determine the unmitigated risks of these hazards. The results of the process hazard analysis are documented in the ISA reference report. GNF-A procedures require that the ISA Reference Report (also referred to as the process hazard analysis [PHA]) be maintained as a living document and supplemented with additional sections as changes are made to the facility and subsequent ISA studies are completed. Changes to the ISA PHA document are documented with an ISA Change Report and included in a Change Request.

3.3.1 Selecting the Analysis Method

GNF-A procedures require that the process hazard analysis method chosen be commensurate with the degree of complexity of the process or operation and the severity of hazards posed. Other factors to consider when selecting the analysis technique include the perceived risks associated with the process and the skill and knowledge of the personnel doing the analysis (which includes their process knowledge, experience, and knowledge of the process hazard analysis technique being used). The ISA leader selects an appropriate process hazard analysis technique, giving due consideration to these factors. Regardless of which method is used, the study must (1) include consideration of nuclear criticality, radiological, chemical/toxic, fire, and explosion hazards and (2) provide the required input to the ISA Reference Report.

HAZOP

The ISA teams used the HAZOP analysis approach to identify and evaluate process hazards for complex systems and processes such as the uranium hexafluoride (UF_6) feed and conversion processes. This technique is a systematic method for identifying ways the process equipment can malfunction or be improperly operated, leading to undesirable conditions. The HAZOP technique is typically used to analyze complex processes and operations. This technique focuses on both safety hazards and operability issues. It may be used both during and after the process design phase. It is applicable for both continuous and batch flow processes.

HAZOP uses the synergy of an interdisciplinary team and a systematic approach to identify hazards and operability problems resulting from deviations from the process's design intent that could lead to undesirable consequences. Typically a fixed set of guide words (e.g., no/not, more, less, as well as) are combined with process parameters (e.g., flow, temperature, pressure, level) to create deviations from the design intent, which are applied to the specified points (nodes) to evaluate potential outcomes.

What-if/Checklist Analysis

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

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

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

3.3.2 Define the Node/Area to Be Studied

The first step of the ISA, identifying the hazards, is initiated by systematically breaking down the process system or operation being studied into well-defined sections or nodes (e.g., major vessels, columns, interconnecting process piping) in which the ins, outs and internal activity/flows can be defined, in order to allow interactions to be studied. All licensed operations are treated in this manner so that the entire facility is evaluated in a logical flow approach. This approach is also used to (1) evaluate the hazards associated with a new process or operation and (2) identify any new hazards that may result from modifications made to an existing process or operation.

In defining the node it is necessary to identify the bounding assumptions and initial conditions that the analysis will be based on. These terms are defined as follows:

Initial Conditions – Important aspects of a process and associated equipment, process operating parameters (e.g., temperature, pressure, flow rate), material throughput, and characteristics of the facility in which the process resides (e.g., design features) that establish the normal operating conditions from which the process hazard analysis is performed.

Bounding Assumptions – Identified assumptions about a process or material characteristics that bound the credible conditions of the process. These assumptions are based on the process chemistry, applicable scientific principles, facility-specific experimental data, operational history, and/or facility construction requirements. In determining the bounding assumptions for process parameters or material characteristics, no credit may be taken for controls placed on those parameters.

The bounding assumptions and initial conditions considered in the analyses shall be documented.

Preparation for the process hazard analysis begins by gathering process safety information on the process system and/or operation to be studied. Information typically used for the analysis included, but is not limited to, the following:

- Piping and instrumentation diagrams (P&IDs)
- Process flow diagrams
- Equipment arrangement drawings with general equipment layout and elevations
- Design temperatures and pressures for major process equipment and interconnected piping

- Materials of construction for major process equipment and interconnected piping
- Operating procedures for normal operations, as well as procedures for startup, shutdown, sampling, emergency shutdown, and any on-line maintenance
- Material safety data sheets (MSDSs) for any chemicals involved in the process (including any intermediate chemical reaction products) and other pertinent data for the chemicals or process chemistry (e.g., chemical reactivity hazards)
- Nuclear Safety Release/Requirements (NSR/Rs),
- Data for process alarms, interlocks, or trips
- Incident reports for the specific area being studied

3.3.3 Identify Credible Accident Sequences

The goal is to identify credible accident sequences by analyzing single initiating events. Using one or more of the approved methods, the ISA team identifies accident sequences associated with a process or operation, including possible unmitigated consequences and causes. Consequences of interest included nuclear criticality accidents, radiological material releases, radiation exposures, chemical/toxic exposures from licensed material or hazardous chemicals produced from licensed material, fires, and explosions.

As required by 10 CFR 70.62, the ISA must consider credible external events, including natural phenomena, for the potential hazardous consequences that they can cause. Natural-phenomenon events, such as hurricanes, tornadoes / high winds, seismic events, and external events, such as aircraft crashes, are addressed separately in GNF-A ISA Summary.

In considering accident sequences at this facility, it is necessary to determine those that are considered not credible and those that are credible. When conducting the process hazard analysis, the ISA team considers each accident sequence as credible, unless it can be determined to be not credible. Accident sequences that do not meet the definition of *not credible* are therefore considered *credible* and treated in accordance with 10 CFR 70.61.

Any one of the following three independent criteria is used to define an event as not credible:

(1) An external event for which the frequency of occurrence can conservatively be estimated as less than once in a million years.

(2) A process deviation that consists of a sequence of many unlikely events or errors for which there is no reason or motive. In determining that there is no reason for such errors, a wide range of possible motives, short of intent to cause harm, must be considered. Necessarily, no such sequence of events can ever have actually happened in any fuel cycle facility.

(3) Process deviations for which there is a convincing argument, based on physical laws, that they are not possible, sound engineering or technical data that the deviations are not possible, or are unquestionably extremely unlikely. The validity of the argument must not depend on any feature of the design or materials controlled by the facility's system of IROFS or management measures.

The bounding assumptions and initial conditions for the node under evaluation may also be considered when identifying credible accident sequences and initiating events. Justification that an accident sequence is not credible shall be documented.

3.3.4 Identify Accident Causes

When analyzing accident sequences, the ISA team considers process deviations, human errors, internal facility events, and credible external events. The team evaluates common mode failure and systems interaction. The team documents postulated accident sequences considered not credible. In addition to normal conditions, the team considers abnormal conditions including start-up, shutdown, maintenance, and process upsets.

3.3.5 Determine the Unmitigated Consequence Severity

For each credible accident sequence identified, the ISA team assigns a severity rank for the unmitigated consequences using the consequence severity rankings shown in Table 3.1 and documents the assigned severity rank in the ISA Reference Report. Assigning a severity rank allows each accident sequence to be categorized in terms of the performance requirements set forth in 10 CFR 70.61 (b), (c), and (d). A severity rank of 3 corresponds to "high consequences"; a severity rank of 2 corresponds to "intermediate consequences." When estimating the possible unmitigated consequences of an accident sequence, the ISA team members use plant experience, guidance from NUREG/CR-6410, *Nuclear Fuel Cycle Accident Analysis Handbook*, and their best judgment. All credible criticality accident sequences are assigned a severity ranking of 3 "high consequences".

The quantitative standards used to assess the consequence severity from chemical exposures to licensed materials or chemicals produced by licensed materials are shown in Table 3.2. The levels-of-concern values shown are derived from the EPA Acute Exposure Guideline Levels (AEGLs), based on an exposure for up to one hour for each limit. The AEGL-1, -2, and -3 values are used as the threshold concentration levels for establishing a low, intermediate, or high severity consequences as shown in Table 3.1.

The uranium hexafluoride concentration in air is not directly equivalent to soluble uranium intake. GNF-A uses worker intake quantities consistent with NRC FCSE Interim Staff Guidance ISG-14, Rev. 0 "Acute Uranium Exposure Standards for Workers", dated June 15, 2015.

Table 3.1 – Facility Consequence Severity Categories

Severity Ranking	Consequence Description		
	Workers	**Off-site Public**	**Environment**
3	• Radiological dose greater than 1 Sv (100 rem) • 400 mg soluble uranium intake • Chemical exposure greater than AEGL-3 • A criticality accident	• Radiological dose greater than 0.25 Sv (25 rem) • 30 mg soluble uranium intake • Chemical exposure greater than AEGL-2 • A criticality accident	• A criticality accident
2	• Radiological dose greater than 0.25 Sv (25 rem) but less than or equal to 1 Sv (100 rem) • 150 mg soluble uranium intake • Chemical exposure greater than AEGL-2 but less than or equal to AEGL-3	• Radiological dose greater than 0.05 Sv (5 rem) but less than or equal to 0.25 Sv (25 rem • Chemical exposure greater than AEGL-1 but less than or equal to AEGL-2	• Radioactive release greater than 5,000 times Table 2 Appendix B of 10 CFR Part 20
1	Accidents with radiological and/or chemical exposures to workers less than those above	Accidents with radiological and/or chemical exposures to the public less than those above	Radioactive releases to the environment producing effects less than those specified above

*Where Sv = Sieverts; AEGL = acute exposure guideline level

Table 3.2 –Levels of Concern (AEGL)

Chemical	AEGL 1	AEGL 2	AEGL 3
Uranium hexafluoride (UF_6)	3.6 mg/m^3	9.6 mg/m^3	36 mg/m^3
Hydrogen fluoride (HF)	1 PPM	24 PPM	44 PPM

(Note: All values shown are for 60-minute exposures)

3.3.6 Determine the Unmitigated Likelihood

The unmitigated likelihood of an accident sequence occurring is required to be determined for all credible accident sequences assigned a consequence severity of "high" or "intermediate." Unmitigated likelihood is the likelihood or frequency that the initiating event or cause of the accident sequence occurs. The team assigns an unmitigated likelihood level for each accident sequence using the defined categories in Table 3.3 and documents the assigned level in the ISA Reference Report. When assigning a likelihood category, the team uses process knowledge, accident sequence information, operating history, and manufacturers/product information to determine which category of likelihood is appropriate. For accident sequences where multiple causes have been identified, the team estimates the likelihood for the most credible cause. This helps assure that the accident sequence is screened using the most conservative estimate of risk.

Table 3.3 – Unmitigated Likelihood Categories

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

* Default selection in absence of quantitative assessment.

3.3.7 Determine the Unmitigated Risk

Credible accident sequences identified for the facility, which have the capability of producing conditions that fail to meet the performance requirements of 10 CFR 70.61 (b), (c) or (d), require IROFS to be assigned to reduce the overall risk to an acceptable level. For each credible accident sequence, the ISA team uses the unmitigated severity category rank and unmitigated likelihood level to assign an unmitigated risk level. (The unmitigated risk is determined from the product of the severity ranking and the unmitigated-likelihood level.) The ISA teams use the risk matrix in Table 3.4 to determine the unmitigated risk and document the assigned risk in the ISA Reference Report.

Table 3.4 – Unmitigated Risk Assignment Matrix

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

3.4 Conducting the Quantitative Analysis

For each accident sequence having an unmitigated risk of unacceptable, IROFS must be assigned and the overall mitigated likelihood determined for each accident sequence. Approved quantification methods include event tree analysis, fault tree analysis, human reliability analysis, LOPA, and the semi-quantitative index method. Figure 3.2 presents the steps taken to quantify the mitigated likelihood of an accident sequence. Specific details for accomplishing these steps are included in this section, including identifying the initiating events, estimating the initiating event's frequency, identifying enabling conditions and conditional events, selection of IROFS, and estimating the failure probability of each credited IROFS.

Determination of the overall likelihood for an accident sequence is documented in a Quantitative Risk Assessment (QRA) report. The purpose of these reports is to provide sufficient background and operational information to understand and examine all accident sequences that result in unacceptable risks for each accident sequence. Each QRA report provides details concerning an accident sequence's quantification, including method used, initiating-event frequency determination, enabling or conditional event probabilities, the IROFS credited to prevent or mitigate the initiating event(s) being analyzed, the failure probabilities for the credited IROFS, and the overall likelihood estimates. The QRA reports are controlled by Configuration Management and are reviewed and approved when modified in accordance with internal procedures 3.5.2. The quantification results from each QRA are summarized in the GNF-A ISA Summary.

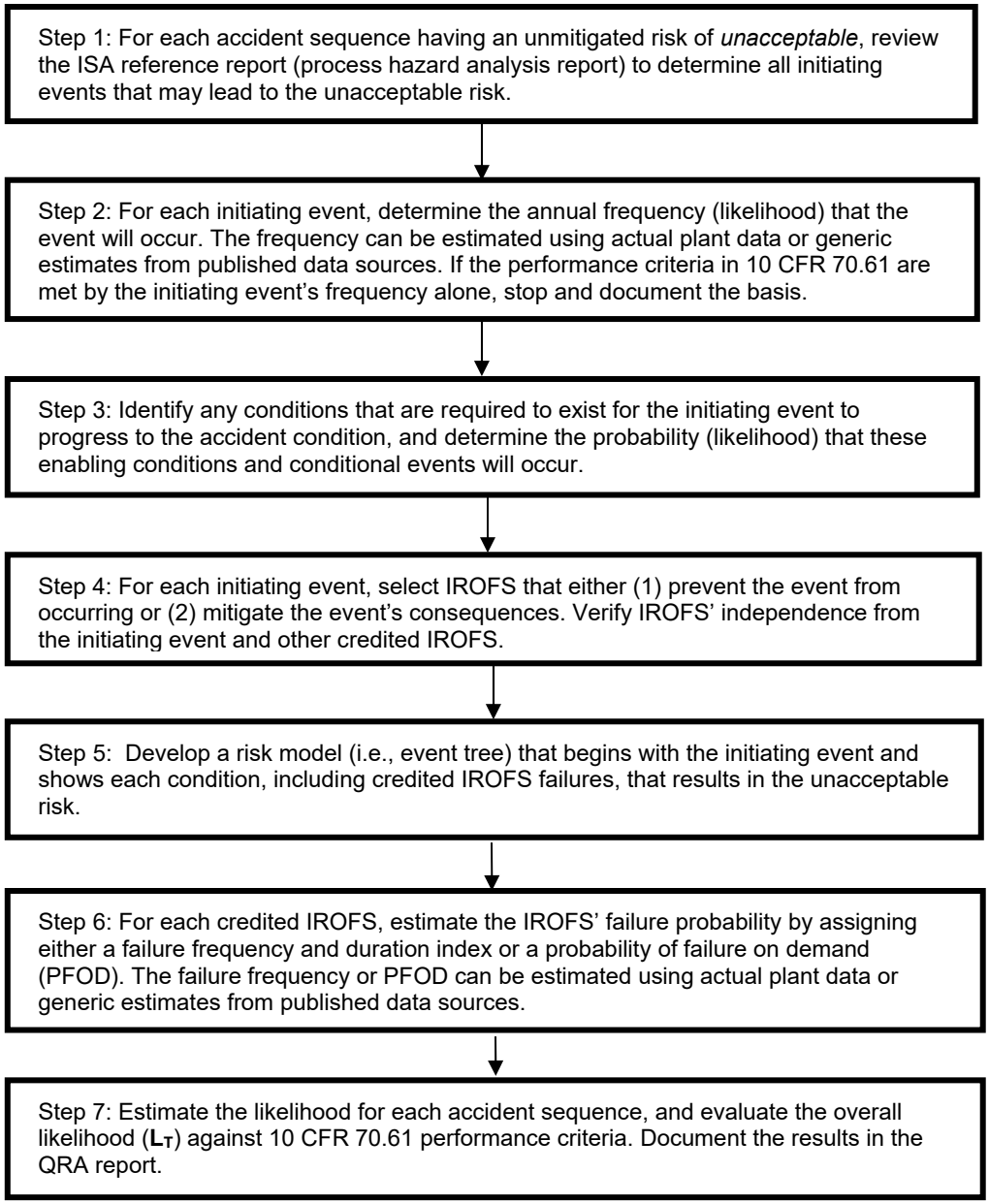

Figure 3.2 – Quantification Methodology

3.4.1 Initiating Events

For each accident sequence requiring quantification, the ISA team member responsible for quantifying the accident sequence first reviews the ISA reference report (process hazard analysis report) to determine all initiating events that may result in an unacceptable risk. The annual frequency of each initiating event is estimated using plant operational experience, industrial performance data, or index values supplied in the GNF-A ISA Summary.

3.4.2 Enabling Conditions and Conditional Events

For each accident sequence, enabling conditions and conditional events that affect the outcome of the accident sequence (i.e., conditions that affect the likelihood of the accident sequence or could mitigate the consequences to either workers or the public) are identified where appropriate.

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

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

3.4.3 IROFS Identification and Evaluation

IROFS are controls or control systems (eg. structures, systems, equipment, components, and activities of personnel) that are relied on to prevent potential accidents at a facility that could exceed the performance requirements of 10 CFR 70.61 or to mitigate their potential consequences. When selecting IROFS, the IROFS must be independent of the initiating event (i.e., occurrence of the initiating event does not cause failure of the IROFS) and other credited IROFS (i.e., failure of one IROFS does not cause failure of another IROFS).

For IROFS that use process control computer systems, such as distributed control systems (DCS) and programmable logic controllers (PLC), GNF-A uses design standards for these systems that result in IROFS with a high reliability and response capability. The architecture design standards for these control systems include the use of mechanically fail-safe final control elements where feasible, security procedures for access and changes to control-system software, separate final elements utilizing separate output modules, and independent control element sensors on separate input modules. When selecting IROFS, GNF-A follows the guidelines from LOPA using the type B methodology for IROFS that use process control computer systems. This methodology limits GNF-A to claiming no more than two IROFS in a single logic controller for any accident

sequence. All control-system IROFS are subject to the applicable management measures as described in the ISA Summary, including periodic verification of IROFS functionality.

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

When evaluating accident sequences, the overall likelihood of the accident sequence must be determined and the adequacy of IROFS to prevent or mitigate the accident sequence is clearly identified.

IROFS which are continuous controls may be evaluated by determination of failure frequency and duration. IROFS which are passive controls or only operate when demanded may be evaluated by determining the probability of failure on demand (PFOD). The duration term does not apply when PFOD is used.

3.4.4 Determining the Overall Likelihood

The *overall likelihood* for an accident sequence is the product of the frequency of the initiating event times the probability of any enabling conditions, times the probability of failure for each credited IROFS. Considerations include frequency of the initiating event, IROFS, enabling conditions, conditional events, time period (duration) of the IROFS failed condition prior to detection/response, IROFS testing or surveillance interval, and independence of IROFS which mitigate the progression of the accident sequence.

Several methods are approved for determining the overall likelihood for an accident sequence. Rigorous methods, such as event tree analysis, are used when the accident sequence is complex and issues such as employee evacuation, the size and location of the material release, and timing or order of IROFS failures needs to be considered. Standard quantitative risk assessment techniques were employed in assessing the overall likelihood for accident sequences using the event-tree analysis method. Overall likelihood is evaluated using limits defined in Table 3.5.

Simplified quantitative methods such as LOPA and an index method are approved for estimating an accident sequence's overall likelihood. The index value for the overall likelihood, L_T, can be determined using the following semi-quantitative equation. The index values are $\log_{10}$ values for each of the annual frequencies and probabilities, which are then summed to determine overall likelihood.

This method conforms to the GNF-A ISA methodology, the GNF-A proposed new overall likelihood methodology, and the additional refinements to the GNF-A overall likelihood methodology.

$$L_T = \lambda_{IE} + \sum_{k=1}^{k=M} P_{E,k} + [\sum_{i=1}^{i=N-1} (\lambda_{f,i} + \lambda_{(T/2+MTTR),i}) \bullet \lambda_{IND,i}] + [\lambda_{f,N} + \lambda_{T/2,N}]$$ Where,

L_T = Overall likelihood index value for the accident sequence being reviewed

Each summed index value term is the $\log_{10}$ representation of each probability or frequency where,

λ_{IE} = Index value for the probability of the initiating event occurring for the identified accident sequence (on a per-year basis, per 3.4.1).

$P_{E,k}$ = Index value for probability of the (k^{th}) enabling condition or conditional event, per 3.4.2. Enabling-condition probabilities are expressed on a per-year basis. These terms are optional.

$\lambda_{f,i}$ = Index value for the failure frequency (on a per-year basis) for an individual (i^{th}) IROFS considered in preventing or mitigating the accident sequence.

$\lambda_{(T/2+MTTR),i}$ = Index value for the duration for an individual (i^{th}) IROFS considered in preventing or mitigating the accident sequence. For functionally tested IROFS, use the sum of one-half the testing (or surveillance) interval and the mean time to repair (MTTR) or place the system in a safe configuration.

Note: For IROFS where the Probability of Failure on Demand (PFOD) is used, replace the term ($\lambda_{f,i} + \lambda_{(T/2+MTTR),i}$) with $\lambda_{PFOD,i}$, which represents the index value for the PFOD for the i^{th} IROFS.

$\lambda_{IND,i}$ = Independence factor for an individual (i^{th}) IROFS. If the failure of a particular IROFS in the identified accident sequence is not caused by, or made more likely to occur by, failure of another IROFS, independence is established, and a value of 1 is used; otherwise a value of 0 is used.

$\lambda_{f,N}$ = Index value for the failure frequency (on a per-year basis) for the final (N^{th}) IROFS considered in preventing or mitigating the accident sequence.

$\lambda_{T/2,N}$ = Index value for the duration for the N^{th} IROFS considered in preventing or mitigating the accident sequence. For functionally tested IROFS, use the sum of one-half the testing (or surveillance) interval. For the final (N^{th}) IROFS considered in preventing or mitigating the accident sequence, the mean time to repair term is excluded for *order-dependent* accident sequences, because this IROFS represents the final barrier in the accident sequence.

Qualitative indices are assigned to the initiating-event frequency, the IROFS failure frequencies and duration indices and then "combined" together with factors representing the immunity to common mode failure to assign a score to the overall (total) likelihood. The overall-likelihood index, L_T, is then evaluated against the applicable limit for the corresponding consequence category. The mitigated likelihood of the accident sequence occurring with the preventive or mitigating IROFS in-place must meet the requirements in 10 CFR 70.61, which requires that unacceptable consequences be limited (see Table 3.5 for mitigated overall likelihood limits).

Table 3.5 – Acceptance Criteria for Overall Likelihood

Index Value (L_T)*	Likelihood (per year)	Acceptance Criteria
-6.0	$\leq 1.0 \times 10^{-6}$	Acceptable for high (and intermediate) consequence accidents
-5.0	$\leq 1.0 \times 10^{-5}$	Acceptable for high (and intermediate) consequence accidents
-4.0	$\leq 1.0 \times 10^{-4}$	Acceptable for high (and intermediate) consequence accidents
-3.0	$\leq 1.0 \times 10^{-3}$	Acceptable for intermediate consequence accidents only; not acceptable for high consequence accidents
-2.0	$\leq 1.0 \times 10^{-2}$	Not acceptable for high or intermediate consequence accidents
-1.0	$\leq 1.0 \times 10^{-1}$	Not acceptable for high or intermediate consequence accidents

* L_T determined using the semi-quantitative equation in Subsection 3.4.4

3.5 ISA Management

3.5.1 ISA Change Management

As described in Chapter 11, Management Measures, a formal configuration management process, governed by written, approved practices, ensures that plant design changes do not adversely impact the ISA at GNF-A. Facility, documentation, and temporary changes are initially evaluated by a trained and approved safety reviewer to determine the potential effects to safety disciplines (criticality, radiation, chemical, industrial, fire and/or explosion), the site license and the ISA, and to assure safe implementation and operation of the change.

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

Changes that do not require NRC prior approval, but which may affect the ISA, require formal evaluation by the ISA team to determine the effects to any ISA documentation, including the ISA Reference Report, Quantitative Risk Assessment report(s), and the ISA Summary. ISA methods are utilized to evaluate the adequacy of existing IROFS and associated management measures, and to designate new or additional IROFS and appropriate management measures as required. Changes are evaluated to ensure they do not remove, without at least an equivalent replacement of safety function, an IROFS listed in the ISA Summary that is necessary for compliance with performance requirements.

Updates to the ISA, are issued in accordance with approved procedures. ISA updates are approved prior to operation of any change.

Unacceptable IROFS performance deficiencies will be corrected, and evaluated for potential changes that may be necessary to the ISA.

3.5.2 Training and Qualifications of ISA Teams

3.5.2.1 Process Hazard Analysis

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

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

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

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

3.5.2.2 Quantitative Risk Analyst

Technical or safety professionals may be assigned as authors of a Quantitative Risk Assessment (QRA) report, after they have completed fundamental training on Risk Assessment.

3.5.3 Management Measures

Management measures ensure that IROFS are designed, implemented, and maintained, as necessary, to be available and reliable to perform their function when needed. Management measures are applied to IROFS in a graded approach based on the type and robustness of the IROFS and the accident sequences the IROFS is preventing or mitigating. The ISA Summary provides a description of the management measures to be applied to each identified IROFS.

A minimum set of management measures are assigned to a particular grouping of IROFS by the ISA Team depending on whether the IROFS are classified as sole IROFS or if they are active engineered control (AEC), passive engineered control (PEC), augmented administrative control (AAC), or administrative control (AC) IROFS.

Within each of the five general classifications of IROFS (Sole, AEC, PEC, AAC, or AC), the IROFS are then assigned specific elements of the management measures. The selection of specific management measure elements is determined by the operational organization based on consideration of the selection criteria.

All IROFS will have management measures applied. The graded approach does not allow for the application of management measures to be waived, but rather allows for varying levels of the number and type of management measures to be applied, as well as the specific elements of management measures, to provide adequate assurance, commensurate with risk, that the IROFS safety function will be met.

The selection criteria used to identify the appropriate application of management measures (or elements of a specific management measure) includes the following:

- Type of IROFS (AEC, PEC, AAC, AC)
- Number of IROFS (e.g., sole IROFS)
- Failure probability of the IROFS as identified in the quantitative risk assessment
- Failure mechanisms
- Design attributes (redundancy, separation requirements, complexity)
- Applicable codes or standards applicable to the IROFS
- Failure history
- Consequence severity (from PHA)
- Worker, public, or environmental consequences (from PHA)
- Type of risk analysis performed (qualitative, semi-quantitative, quantitative)
- Safety function
- Preventative or mitigative IROFS

The use of the graded approach in assigning management measures to IROFS is documented and provided to the ISA Team performing the ISA review.

The management measures are described in Chapter 11 and in the ISA Summary. The ISA Summary specifies the management measures assigned to each IROFS.

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 from fuel manufacturing processes is exhausted as appropriate 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 stack discharge.

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 Level (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 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

CHAPTER 10.0
DECOMMISSIONING

The current Decommissioning Funding Plan is dated September 7, 2016 and as it may be further revised in accordance with 10 CFR 70.25(e)

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.