CHAPTER 5.0
NUCLEAR CRITICALITY SAFETY

5.1 NUCLEAR CRITICALITY SAFETY PROGRAM MANAGEMENT

5.1.1 CRITICALITY SAFETY DESIGN PHILOSOPHY

The Process analysis as discussed in section 4.1.2 and the Double-contingency principle as identified in section 4.2.2 of the nationally recognized American National Standard ANSI/ANS-8.1 (2014) are fundamental bases for design and operation of processes within the GNF-A fuel manufacturing operations using fissile materials. As such,

- "Before a new operation with fissionable material is begun, or before an existing operation is changed, it shall be determined that the entire process will be subcritical under both normal and credible abnormal conditions."
- "Process designs should incorporate sufficient factors 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 a nuclear criticality, a defense of one or more system parameters provided by at least two independent controls or process condition changes is documented in the criticality safety analysis (CSA), which is reviewed and enforced.

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 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 a 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 parameters

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 criticality safety 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, ISA documents are updated per section 11.2, 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 risk. 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 as described in internal procedures..

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 identified 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 (2014), 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. CSE qualifications shall be documented by 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. 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 documented and provided to the manager of the nuclear criticality safety function and facility management. Findings requiring corrective action 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 affected 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 installed 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 (R2012). 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 in accord with internal records management requirements outlined in Section 11.8.

5.4 METHODOLOGIES AND TECHNICAL PRACTICES

5.4.1 CONTROL PRACTICES

Criticality safety analyses identify specific limits and controls necessary for safe operation of a process. The Area Manager, with NCS support, implements the limits and controls documented in the CSA.

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 conduct preoperational audits (e.g., field verification of design features, functional test review) to verify that 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 other verification means. 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 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. This requires a combination of routine maintenance, calibration, 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 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. Beyond appropriate installation and management measures (e.g., periodic inspection, preventive maintenance), a passive engineered control requires no human action to perform its safety function. 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. Beyond appropriate installation and management measures (e.g., periodic functional testing), an active engineered control requires no human action to perform its safety function. Assurance is maintained through specific periodic calibration, functional testing, and preventive maintenance as appropriate. Active engineered controls that are designed to be fail-safe (i.e., meaning failure of the control results in a safe condition) are preferred.

5.4.2.3 Administrative Controls

- 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.

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

Use of administrative controls should be 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 **favorable geometry** values of Table 5.1 contain dimensions for sphere, cylinder, and slab which may be used for applicable operations 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 and the moderating material is not more effective than water. Acceptable safety margins for units listed in this table are documented in accordance with Section 5.4.5 analysis methods (keff $+ 3\sigma \leq$ USL).

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. When not applicable, the Criticality Safety Engineer may use approved validated stochastic or deterministic codes to determine explicit subcritical limits for the application in accordance with Section 5.4.5 analysis methods.

The **safe batch** values of Table 5.2 may be used for applicable operations at GNF-A. Application of these safe batch values is limited to situations where the neutron reflection present does not exceed that due to full water reflection and the moderating material is not more effective than water. Criticality safety may be based on U235 enrichment based mass limits in the following manner:

- 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 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:

kg X = (kg UO_2 • 0.88) / f

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

The safe concentration values below may be used for applicable operations at GNF-A subject to provision for adequate protection against precipitation or other circumstances which may increase concentration.

- 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 Favorable Geometry Values*

Homogeneous UO_2-H_2O Mixtures (ρ_{UO2} = 10.96 g/cc)	Weight Percent U235	Infinite Cylinder Diameters (cm)	Infinite Slab Thickness (cm)	Sphere Volume (Liters)
	5.0	23.26	8.54	21.14
	5.5	22.53	8.01	19.33
	6.0	21.90	7.62	17.97
	6.5	21.34	7.32	16.73
	7.0	20.87	7.04	15.80
	7.5	20.45	6.82	15.06
	8.0	20.08	6.61	14.35
Homogeneous Aqueous Uranyl Nitrate Solutions ($\rho_{UO2(NO3)2}$ = 2.203 g/cc)	**Weight Percent U235**	**Infinite Cylinder Diameters (cm)**	**Infinite Slab Thickness (cm)**	**Sphere Volume (Liters)**
	5.0	47.76	23.24	145.78
	5.5	44.50	20.87	122.66
	6.0	41.57	18.85	104.06
	6.5	39.79	17.42	92.73
	7.0	37.59	16.00	81.41
	7.5	36.15	14.92	72.98
	8.0	34.88	13.97	66.27
Heterogeneous UO2 – H_2O Mixtures (ρ_{UO2} = 10.96 g/cc)	**Weight Percent U235**	**Infinite Cylinder Diameters (cm)**	**Infinite Slab Thickness (cm)**	**Sphere Volume (Liters)**
	5.0	21.75	7.70	17.64
	5.5	21.10	7.40	16.37
	6.0	20.60	7.10	15.30
	6.5	20.10	6.85	14.42
	7.0	19.75	6.60	13.72
	7.5	19.40	6.40	13.10
	8.0	19.05	6.20	12.64

*** Values reported are fully reflected by 30.48 cm full density H2O (with exception of slab; the bottom is reflected by 60.96 cm Oak Ridge Concrete and the top with 30.48 cm full density H2O). 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 (kg UO_2)	Heterogeneous UO_2 Pellets & Water Mixtures (kg UO_2)
5.0	18.58	15.80
5.5	15.84	13.80
6.0	13.76	12.20
6.5	12.21	10.80
7.0	10.89	9.70
7.5	9.83	9.00
8.0	8.94	8.20

***NOTE: These values represent 45% of the calculated 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 worst credible conditions (e.g., reflection, moderation, heterogeneity, and enrichment) for the material 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 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 along with adequate measurement uncertainty may be employed to verify the mass measurements for a specific quantity of material.

Establishment of mass limits involves consideration of worst credible 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.

5.4.4.3 **Moderation** - Moderator control may be used for nuclear criticality safety control on its own or in combination with other control methods. Moderator control areas shall be defined in the process evaluation in which moderators are limited and controlled for nuclear criticality. For areas where moderation is used in conjunction with other control methods, the area is classified as a 'moderation control area (MCA)' and posted accordingly. When moderation is the single criticality safety controlled parameter, the area is classified as a 'moderation restricted area (MRA)' and posted accordingly.

Process evaluations for MCA/MRA designated areas shall explicitly identify the limits, controls, and engineered barriers for designated moderator control areas. Material properties, credible moderator present in, introduced to, or accumulated in an MCA/MRA shall be considered. Credible non-uniform distribution of moderators, moderator content measurement, and fire suppression methods shall also be considered.

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 at least two independent combinations of measurements and physical control, each physical control capable of preventing the concentration limit being exceeded. The preferred method of demonstrating double contingency being that one of the two combinations is a passive control (e.g., favorable geometry tanks) or at least one of the two is an active engineered (e.g., in-line density monitoring).

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 also be taken for neutron absorbers added to fuel, such as gadolinia.

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 (e.g., serial number, QA documentation, etc.), visual inspection or direct measurement, as appropriate for the application.

For crediting neutron absorbers added to the fuel, such as gadolinia, the following requirements apply:

- For in-process fuel (e.g., mechanical mixing of gadolinia powder with uranium oxide powder), the continued presence of the absorber in the fuel, its distribution, and its concertation is verified using an appropriate method. The system design should include factors such as process conditions, hazards, and human errors for potential degradation of the neutron absorber. Acquisition, storage, preparation and use of the neutron absorbers should conform to the established quality program.
- For fuel bundles, 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.

5.4.4.6 **Spacing (or Unit Interaction)** - Criticality safety controls may be based on isolation or interacting unit spacing. Unless a basis is explicitly documented in the criticality safety analysis, then units or arrays may be considered effectively non-interacting (isolated) when they are separated by either of the following:

- 12-inches of full density water equivalent,
- 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

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 may also be used. In either case, the criticality safety analysis must comply with the requirements of Sections 5.1.1 and 5.4.5.5.

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** – Worst credible reflection conditions will be considered in the development of all system controls and limits. Generally, systems are designed and operated with the assumption of 12-inch water or optimum reflection. However, subject to an approved analysis documenting controls or process conditions which limit reflection, certain system designs may be 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 factor (k_{eff}) of the system plus three (3) times the standard deviation of the Monte Carlo code must be 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 upset 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), and Monte Carlo codes (e.g., GEMER, SCALE/KENO-VI, MCNP) which utilize stochastic methods to approximate a solution to the 3-D neutron transport equation. Additionally, 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.

SCALE/KENO-VI is a multi-group Monte Carlo program which approximates a solution to the neutron transport equation in 3-dimensional space. KENO-VI is an extension of the KENO Monte Carlo criticality program developed by Oak Ridge National Laboratory (ORNL) for use in the SCALE[1] system and contains all features currently in KENO-V.a, plus an enhanced flexible geometry package known as the SCALE Generalized Geometry Package (SGGP). The CSAS6 module is used to process problem specific cross section sets from the ENDF/B-VII.0 library and to run KENO-VI. The SCALE Standard Composition Library is used for the theoretical densities and isotopic compositions of material used. KENO-VI calculations are performed in the continuous energy mode as opposed to the traditional multigroup approach.

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.

MCNP[2] is a general-purpose Monte Carlo N–Particle transport code developed by Los Alamos National Laboratory, can be used for neutron, photon, electron, or coupled neutron/photon/electron transport, including the capability to calculate eigenvalues for fissile medium systems. The code treats an arbitrary 3-dimensional configuration of materials in geometric cells bounded by first-and second-degree surfaces and fourth-degree elliptical tori. MCNP uses continuous-energy nuclear and atomic data libraries. For neutrons, all reactions given in a particular cross section evaluation (such as ENDF/B) are considered. Thermal neutrons are described by both the free gas and $S(\alpha,\beta)$ models. For photons, the code accounts for incoherent and coherent scattering, as well as fluorescent emission after photoelectric absorption, and absorption in electron-positron pair production. Electron and positron transport processes account for angular deflection through multiple Coulomb scattering, collisional energy loss with optional straggling, and the production of secondary particles.

[1] SCALE, “A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design,” ORNL/TM-2005/39, Version 6.1, June 2011 (as amended).

[2] Los Alamos National Laboratory, “MCNP – A General Monte Carlo N-Particle Transport Code, Version 6.2,” LA-UR-17-29981, 2017 (as amended).

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-8.1-2014 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 (2014), the criteria at GNF-A to establish subcriticality requires that for a system or process to be considered subcritical the calculated k_{eff} plus three (3) times the standard deviation for the Monte Carlo code 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 outlined in the validation report. The USL is defined as follows:

$$\text{USL} = 1 + \text{bias} - \text{bias uncertainty} - \text{MMS}$$

  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 upset conditions, must be satisfied when using k_{eff} calculations by Monte Carlo methods to establish subcritical limits for the GNF-A facility:

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

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

If parameters needed for anticipated applications are 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 in 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 included. Otherwise, critical experiments should be added for further justification.

- Statistical methods may be used to ensure that the extrapolation is not large. The SCALE/TSUNAMI code may be used to compare the application system to the benchmark experiments for similarity and USL penalty determination.

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. For codes developed or modified at GNF, 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)

A CSA is prepared or updated for each new or significantly modified unit or process system within GNF-A in accordance with section 5.1.2.1 and established configuration management control practices defined in Chapter 11.

The scope and content of any particular CSA reflects the needs and characteristics of the system being analyzed, as specified in internal procedures, and typically includes the following elements:

- Scope – Defines the extent and purpose of the analysis.

- General Process Description - 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 Hazards – A listing and evaluation of credible process upset conditions applicable to scope, and a discussion of how established nuclear criticality safety limits are addressed for each credible process upset condition. Independent controls and management measures that demonstrate compliance with the Double Contingency Principle are described with consideration for common mode failure.

- Methodology – A description of compute code(s) used, bias and bias uncertainty, area of applicability, bounding assumptions, calculational assumptions, and design features.

- Calculations and Results – A description of model constructs, how calculations were performed, what analytic tools or reference documents were used, and a summary of the calculational result and associated uncertainty (Keff + 3σ) as a function of key parameter(s). When applicable, the assigned bias and associated bias uncertainty is stated and compared to accident limit results. Limits derived are based on most reactive values of uncontrolled parameters or based on worst credible values of uncontrolled parameters with documented justifications.

- Specifications and Requirements for Safety – When applicable, this element presents both bounding design assumptions and the criticality safety requirements for correct implementation of established controls. The requirements are grouped according to passive, active, or administrative controls. Generic management measures and applicable elements of combustible material control programs may also be included in this element.

- Conclusions – This element concludes the analysis with pertinent summary statements and includes a statement regarding license compliance for process analysis.

- References – This element includes a listing of applicable references cited in the analysis.

- Attachments – This element includes appropriate attachments to support analysis content and may include (but not limited to) materials used, data trends, sample input file(s), tabulated (Keff + 3σ) results.

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.