

Chapter 5 – Nuclear Criticality Safety

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5.0 NUCLEAR CRITICALITY SAFETY

5.1 USE OF INDUSTRY STANDARDS

The Nuclear Criticality Safety (NCS) Program for the Facility is in accordance with U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 3.71 (NRC 2018), which provides guidance on complying with the applicable portions of NRC regulations. This includes 10 CFR Part 70, by describing procedures for preventing nuclear criticality accidents in operations involving handling, processing, storing, and transporting special nuclear material (SNM) at fuel and material facilities. OE commits to apply the ANSI/ANS criticality safety standards endorsed in Regulatory Guide 3.71 (NRC 2018) and to comply with all mandatory (“shall”) requirements contained therein:

- ANSI/ANS-8.1-2014_R2023 - NCS in Operations with Fissionable Materials Outside Reactors
- ANSI/ANS-8.3-1997_R2017 - Criticality Accident Alarm System
- ANSI/ANS-8.19-2014_R2024 - Administrative Practices for NCS
- ANSI/ANS-8.20-2025 - NCS Training for Fissionable Material Operations Outside Reactors
- ANSI/ANS-8.22-1997_R2021 - NCS Based on Limiting and Controlling Moderators
- ANSI/ANS-8.23-2019_R2024 - Nuclear Criticality Accident Emergency Planning and Response
- ANSI/ANS-8.24-2017 Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations
- ANSI/ANS 8.26 Nuclear Criticality Safety Engineer Training and Qualification Program

5.2 CRITICALITY ACCIDENT ALARM SYSTEM

The Facility is provided with a Criticality Accident Alarm System (CAAS) as required by 10 CFR 70.24. Each area where Special Nuclear Material (SNM) is handled, used, or stored are provided with CAAS coverage. The CAAS will be uniform throughout the Facility for the type of radiation detected and alarm signals. Documentation is maintained which demonstrates the CAAS meets the requirements of 10 CFR 70.24. Emergency management measures are covered in the Facility Emergency Plan.

The CAAS is designed, installed, and maintained in accordance with ANSI/ANS-8.3-1997 (R2017) (ANSI 2017) as modified by Regulatory Guide 3.71 (NRC 2018).

The CAAS is provided with emergency power and is designed to remain operational during credible events or conditions, including fire, explosion, corrosive atmosphere, or seismic shock.

Whenever the CAAS is not functional, compensatory measures, such as limiting access and restricting SNM movement, will be implemented. On-site guidance is provided and is based on process-specific considerations that consider applicable risk trade-off of the duration of reliance on compensatory measures versus the risk associated with process upset in shutdown.

5.3 EMERGENCY PLANNING AND RESPONSE

OE commits to ANSI/ANS-8.23-2019 (ANSI 2024b) for development of the Facility Emergency Plan (ANSI 2024b).

Pursuant to 10 CFR 70.24(b)(1), areas that require a CAAS are equipped with personnel accident dosimeters readily available with prompt onsite dosimeter readout.

Pursuant to 10 CFR 70.24(b)(2), the Facility includes arrangements for the on-site decontamination of personnel and the transport and medical treatment of exposed individuals outside the site boundary.

Additional information regarding nuclear accident planning and response is detailed in the Facility Emergency Plan.

5.4 SUBCRITICALITY AND DOUBLE-CONTINGENCY PRINCIPLE

The NCS criteria in Section 5.7 are used for managing criticality safety and include adherence to the double contingency principle as stated in the ANSI/ANS-8.1-2014 (ANSI 2023a). The adopted double contingency principle states:

“process design 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.”

Each process that has accident sequences that could result in an inadvertent nuclear criticality at the Facility meets the double contingency principle. To meet the double contingency principle, the Facility will incorporate into process designs sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

An accident sequence that could result in a criticality is not credible when the sequence is shown to be physically impossible or relies on conditions that cannot occur within the design, operation, or configuration of the system. A sequence is considered not credible if it requires multiple independent and unlikely failures, contradicts known physical or chemical behavior, depends on fissile configurations that cannot be achieved given fixed geometry or material limits, or is precluded by inherent or passive design features.

If double-contingency protection is lost through the failure of one or more of these controls, the process should be halted until the controls can be reestablished.

Using these NCS criteria, including the double contingency principle, low enriched uranium enrichment facilities have never had an accidental criticality. The Facility will produce no greater than 10 wt.% enrichment. However, as additional conservatism, the nuclear criticality safety analyses are performed assuming a ²³⁵U enrichment of 11 wt.%, except for 48Y product cylinders, which are limited to 8 wt.%, and the contingency dump system. The design of the contingency dump system controls enrichment to a limit of 1.5 wt.%. Both the 10 wt.% value and the 1.5 wt.% value include appropriate margins for safety for OE facility product equipment and systems, 48Y product cylinders in use on site are limited to 8 wt.%.

In accordance with 10 CFR 70.61(d), the general criticality safety philosophy is to prevent accidental uranium enrichment excesses, provide geometrical safety when practical, provide for moderation controls within the Uranium Hexafluoride (UF₆) processes and impose strict mass limits on containers of aqueous, solvent based, or acid solutions containing uranium.

All NCS controls are preventative in nature; there are no mitigative NCS controls. Interaction controls provide safe movement and storage of components. Facility and equipment features ensure the prevention of excessive enrichment.

As required by 10 CFR 70.64(a), by observing the double contingency principle throughout the Facility, a criticality accident is reduced.

OE commits to use a Monte Carlo computer code, SCALE 6.2.4, as the only method of analysis to calculate the effective multiplication factor k_{eff} . SCALE 6.2.4 is validated in accordance with ANSI/ANS-8.24-2017 (ANSI 2023b).

5.5 ORGANIZATION AND ADMINISTRATION OF THE NCS PROGRAM

The NCS organization is responsible for implementing the Nuclear Criticality Safety Program.

The Nuclear Criticality Safety Program Manager reports to the Environmental, Health, Safety, and Licensing (EHS&L) Manager in the organization, as described in Chapter 2. The designated responsibilities and minimum qualifications of the NCS Manager described in Chapter 2 Section 2.2.1 F are applicable to the incumbent manager.

The NCS Program Manager established the training and qualification program for NCS engineers. Training and qualification for the NCS Program Manager and NCS Engineers will be consistent with ANSI/ANS-8.26 (ANSI 2024c).

The NCS Engineers shall hold, as a minimum, a bachelor's degree in an engineering or scientific field and have successfully completed a training program applicable to the scope of operations, in the physics of criticality and in associated safety practices. The NCS Engineers shall have at least two years of experience performing criticality safety analyses.

The Nuclear Criticality Safety Program Manager implements the NCS Program to meet the regulatory requirements of 10 CFR 70.

The NCS Program commits to the following objectives:

- Performing and documenting Nuclear Criticality Safety Evaluations (NCSE) for new or changed processes and establishing safety limits and controls as necessary to ensure that processes will remain subcritical under normal and credible abnormal conditions
- Establishing, as practicable, double-contingency protection and defense-in-depth measures; ensuring sufficient margins of safety and subcriticality to provide additional assurance that the likelihood of criticality will be acceptably low
- Establishing and maintaining a CAAS system and emergency-response procedures to protect health and safety in the event criticality occurs
- Providing technical support to emergency response personnel in responding to and recovering from abnormal conditions and emergencies up to and including a criticality accident
- Verifying the adequacy of criticality controls through audits and assessments, including observation of operations and verification of equipment configuration
- Ensuring the adequacy of NCSEs through peer reviews, self-assessments, and validation and verification of calculational methods
- Training and otherwise supporting operations in procedures to ensure the safe handling of special nuclear material
- Supporting regulatory compliance with regard to event reporting (10 CFR 70.50 and Appendix A to 10 CFR 70), complying with the facility change process (10 CFR 70.72), and participating in the performance and documentation of the facility's ISA (10 CFR 70.61 through 70.66) insofar as they pertain to criticality safety.

The NCS Program structure follows guidance in ANSI/ANS-8.1-2014 (ANSI 2023a) and ANSI/ANS-8.19-2014 (ANSI 2024a) including definition of the responsibilities and authorities of key program personnel will be provided.

The NCS emergency procedure training will be provided. The NCS postings will be provided and maintained current.

The NCS program will be used to establish and maintain NCS safety limits and NCS operating limits for IROFS in nuclear processes and a commitment to maintain adequate management measures to ensure the availability and reliability of the IROFS.

The NCS methodologies and technical practices will be kept applicable to current configuration by means of the configuration management function. The NCS program will be upgraded, as necessary, to reflect changes in the ISA or NCS methodologies and to modify operating and maintenance procedures in ways that could reduce the likelihood of the occurrence of an inadvertent nuclear criticality.

The NCS program will be used to evaluate modifications to operations, to recommend process parameter changes necessary to maintain the safe operation of the Facility, and to select appropriate IROFS and management measures.

The NCS program will be used to promptly detect NCS deficiencies by means of operational inspections, audits, and investigations. Deficiencies will be entered into the corrective action program to prevent the recurrence of unacceptable performance deficiencies in IROFS, NCS function, or management measures.

NCS program records will be retained as described in Chapter 11 and the Quality Assurance Program Description.

5.6 MANAGEMENT MEASURES APPLIED TO NCS PROGRAM

Changes to NCS Program (e.g., processes, procedures, design criteria) will be screened through the Configuration Management Program. OE commits to following industry practices described in ANSI/ANS-8.19-2014 (ANSI 2024a) and ANSI/ANS-8.20-2025 (ANSI 2025) regarding training, procedures, and audits and assessments. In addition to the double contingency principle, effective management of the NCS Program including establishment of safety parameters and procedures. The NCS baseline design criteria requirements in 10 CFR 70.64(a) will be adhered to.

NCS Training will be provided to individuals who handle nuclear material at the Facility. The training is based upon the training program described in ANSI/ANS-8.20-2025 (ANSI 2025). The training program is developed and implemented with input from the criticality safety staff, training staff, and management. The training focuses on the following:

- Appreciation of the physics of nuclear criticality safety.
- Analysis of jobs and tasks to determine what a worker must know to perform tasks efficiently.
- Design and development of learning objectives based upon the analysis of jobs and tasks that reflect the knowledge, skills, and abilities needed by the worker.
- Implementation of revised or temporary operating procedures.
- Required response to the activation of the CAAS.
- Required response to NCS nonconformance.

With regard to audits and assessments, OE is committed to conducting and documenting periodic NCS program audits at least once every 2 years. The audit should be performed by staff independent of those who had performed the functions themselves, either internal or external to the NCS Organization. OE applies a graded approach, audit frequency may be increased for activities with higher safety significance, adverse performance trends, or elevated operational complexity. Conversely, reduced-significance activities may be evaluated using proportionally

scaled assessment methods, provided the minimum 2-year audit requirement is met. Any graded adjustments are supported by documented technical justification and managed through the organization's commitment-control process.

5.7 TECHNICAL PRACTICES FOR NCS

This section describes the methodologies and technical practices used to perform the NCSEs. The determination of the NCS controlled parameters and their application and the determination of the NCS limits on IROFS are also presented.

5.7.1 Calculational Method Validation

SCALE 6.2.4 (ORNL 2020) is a Monte Carlo tool for NCS analysis. The advanced geometry modeling capability and detailed continuous-energy and multigroup collision modeling treatments provide realistic 3-dimensional models for an accurate simulation of neutronic behavior to provide the best estimate neutron multiplication factor, k_{eff} . Complex models can be simply set up and verified. Additionally, SCALE 6.2.4 (ORNL 2020) has demonstrable accuracy over a wide range of applications and is distributed with a validation database comprising critical experiments covering uranium, plutonium and mixed systems over a wide range of moderation and reflection. The selected experiments are regarded as being representative of systems that are widely encountered in the nuclear industry, particularly with respect to chemical plant operations, transportation, and storage. The validation database is subject to on-going review and enhancement. A categorization option is available in SCALE 6.2.4 (ORNL 2020) to assist the criticality analyst in determining the type of system being assessed and provides a quick check that a calculation is adequately covered by validation cases.

The validation process establishes method bias by comparing measured results from laboratory critical experiments to method-calculated results for the same systems. The verification and validation processes are controlled and documented. The validation establishes a method bias by correlating the results of critical experiments with results calculated for the same systems by the method being validated. Critical experiments are selected to be representative of the systems to be evaluated in specific design applications. The range of experimental conditions encompassed by a selected set of benchmark experiments establishes the area of applicability over which the calculated method bias is applicable. Benchmark experiments are selected that resemble as closely as practical the systems being evaluated in the design application.

The extensive validation database contains a number of solution experiments applicable to this application involving both low and high-enriched uranium. The SCALE 6.2.4 (ORNL 2020) code with the V7-252 group ENDF/B-VII.1 cross section library is validated against the experiments (OE 2025) provided in the International Handbook of Evaluated Criticality Safety Benchmark Experiments (NEA 2023), NUREG/CR-1071 (NRC 1980), and benchmark catalogued in ORNL/CSD/TM-238 (ORNL 1986).

The validation process complies with the applicable requirements from ANSI/ANS-8.24-2017 (ANSI 2023b) as well as guidance from NUREG/CR-6698 (NRC 2001), NUREG/CR-7311 (NRC 2025) and NUREG-1520 (NRC 2015).

The criticality code validation methodology can be divided into four steps:

1. Identify general Facility design applications (Area of Interest, Aoi).
2. Select applicable benchmark experiments for the Area of Applicability (AoA) that bounds the Aoi.

3. Model and calculate k_{eff} values of selected critical benchmark experiments.
4. Perform statistical analysis of calculated results to determine computational bias and bias uncertainty associated with the computer code and cross section library to calculate an Upper Subcritical/Safety Limit, USL.

The AoA is established for the following four (4) parameters which are important for reactivity calculations: enrichment, moderation, epithermal fission fraction, calculation type and materials.

The area of applicability for enrichment includes benchmarks ranging from low-enriched material to mixtures of LEU and fissile material, up to systems containing 36.6 wt.% ^{235}U , which bound the Facility's maximum enrichment of 10 wt.% ^{235}U .

The area of applicability for moderation is supported by benchmarks spanning from low to high H/U-235 or C/U-235 atom-density ratio.

The area of applicability limits the contribution of epithermal-range fissions (0.8–10 eV, groups 145–209) to no more than 5% of total fissions for systems with moderation between $3 < \text{H/X} < 60$.

The area of applicability calculation type includes SCALE 6.2.4 (CSAS5 module) with the 252-group cross section library (v7-252). The following CSAS5 problem types are allowed: Infinite homogeneous, lattice cell, multi-region, and any geometry available in KENO V.a.

The area of applicability for material includes basic fissile material forms such as uranium oxide, uranium tetrafluoride, uranium metal, uranyl nitrate and uranyl fluoride.

A total of 601 experiments is selected for the validation effort; the description of all relevant experiments is provided in Appendix C of (OE 2025).

CSAS5 module of SCALE 6.2.4 is employed to model and calculate the k_{eff} ($k_{\text{eff}} = k_{\text{KENO}} + 2 \sigma_{\text{KENO}}$) of the selected critical experiments. Appendix B of (OE 2025) presents the calculated results.

The statistical analysis of the selected critical experiments k_{eff} is then performed to determine the bias and USL.

The formulation for the USL, (NRC 2025), is provided as:

$$\text{USL} = 1 + \beta - \sigma_{\beta} - M_A - M_D \text{ if } \beta \leq 0 \text{ or}$$

$$\text{USL} = 1 - \sigma_{\beta} - M_A - M_D \text{ if } \beta > 0$$

Where:

M_A is the administrative margin

M_D is the margin added to account for any deficiencies associated with the validation suite. M_D is referred to as Δ_{AoA} in (OE 2025). When no extrapolations are made to the AoA, then Δ_{AoA} is 0. This validation effort rigorously evaluates applicable benchmarks to ensure that the computational method is not extrapolated beyond its established AoA.

β is the bias and σ_{β} is the bias uncertainty.

When $\beta > 0$ then

$$\text{USL} = 1 - \sigma_{\beta} - M_A$$

NUREG-1520 Appendix B (NRC 2015) provides guidance in the development of adequate justification for the M_A (or Minimum Margin of Subcriticality, MMS, as referred to in NUREG-1520) based on the following approaches:

- Justification based on conservatism in calculational models
- Justification based on validation methodology and results
- Justification based on additional risk Informed considerations
- Justification based on statistical evaluation to establish MMS

NUREG-1520 (NRC 2015) Appendix B establishes that the first three approaches can provide an adequate technical basis for MMS selection, while the 4th method (based on statistical evaluation) is not considered by the NRC to be an appropriate basis for selection of a specific MMS.

Sections 2.3.1 through 2.3.3 of (OE 2025) address the 3 first justifications and provide the basis for using an MMS, or M_A , of 0.02 given the rigorous experiment selection and analysis and a conservative application of statistical methods to calculate bias, bias uncertainty, and the associated USL.

Table 5-1 summarizes the results of each USL calculation, for all cases and by subsets. The most limiting value is that of H/X: Epithermal ($3 < H/X < 60$) set which is 0.9525 (determined with 0.02 administrative margin).

SCALE 6.2.4, using the 252-group ENDF/B-VII cross-section library, has been validated for performing the criticality analyses at the Facility. Based on this validation, the calculated upper subcritical limit (USL) supports using a safety margin of 0.05 for nuclear criticality safety. This margin, expressed as $k_{\text{eff}} = k_{\text{KENO}} + 3\sigma_{\text{KENO}} < 0.95$, is sufficient to ensure that a system remains subcritical under normal and abnormal credible conditions. Therefore, the Facility will be designed based on:

$$k_{\text{eff}} = k_{\text{KENO}} + 3\sigma_{\text{KENO}} < 0.95$$

Table 5-1 SCALE-6.2.4 V7-252 Validation Summary
[Security-Related Information Withheld Under 10 CFR 2.390]

5.7.2 Nuclear Criticality Safety Evaluations

OE is committed to performing NCSE in accordance with documented and approved administrative procedures adhering to these following principles:

- a. NCS safety limits will be established based on analyses assuming optimum or the most reactive credible values of parameters unless specified controls are implemented to limit parameters to a particular range of values. When less than the optimum values are used, and corresponding controls are not identified, adequate justifications will be provided in the NCSE.
- b. NCS operating limits maybe established as appropriate to ensure that safety limits are unlikely to be exceeded. If separate operating limits are specified, process variability and uncertainty should be considered.
- c. The specific controls and management measures necessary to enforce NCSE safety limits and/or operating limits will be specified.

OE is committed to providing the technical basis that demonstrates subcriticality under normal and credible abnormal conditions and compliance with the double-contingency principle in the NCSE.

OE is committed to incorporating each CSE into its configuration-management program and its system of ISA documentation.

5.7.3 Evaluation and Implementation of Controlled Parameters

The use of a single criticality-safety control to maintain two or more parameters is credited as only one contingency for purposes of meeting the double-contingency principle.

OE commits to the preferred use of passive engineered controls; in particular, passive engineered geometry control.

OE commits to the following order of preference for NCS control: (1) passive engineered, (2) active engineered, (3) enhanced administrative, and (4) simple administrative controls.

OE commits to preference for designating explicit NCS controls over reliance on the natural and credible course of events.

OE commits to preference for control of two or more parameters over multiple controls on a single parameter. If relying on two or more controls on a single parameter. If relying on two or more controls on a single parameter, OE commits to preference for diverse over redundant means of control.

For the purposes of the NCSE, it is assumed that UF_6 comes in contact with water to produce aqueous solutions of UO_2F_2 . Most components that form part of the centrifuge plant or are connected to it reflect the assumption that any accumulation of uranium is taken to be in the form of a uranyl fluoride/water mixture at a maximum H/U atomic ratio of 7 (exceptions are discussed in the associated NCSE documentation). The ratio is based on the assumption that significant quantities of moderated uranium could only accumulate by reaction between UF_6 and moisture in air leaking into the plant process equipment. Due to the high vacuum requirements of a centrifuge plant, in-leakage is controlled at very low levels and thus the H/U ratio of 7 represents an abnormal condition. Gross water ingress into plant is not considered credible for the same reasons. In the case of oils, UF_6 pumps and vacuum pumps use a fully fluorinated PFPE (perfluorinated polyether) type lubricant. Mixtures of UF_6 and PFPE oil would be a less pessimistic case than the uranyl fluoride / water mixture considered since maximum HF solubility in PFPE is only approximately 0.1% by weight.

A maximum H/U ratio of 7 for the uranyl fluoride-water mixture is derived as follows. The stoichiometric reaction between UF_6 and water vapor in the presence of excess UF_6 can be represented by the equation:



Due to its hygroscopic nature, the resulting uranyl fluoride is likely to form a hydrate compound. Solid hydrates of compositions $\text{UO}_2\text{F}_2 \cdot 1.5\text{H}_2\text{O}$ and $\text{UO}_2\text{F}_2 \cdot 2\text{H}_2\text{O}$ can form in the presence of water vapor, the former composition being the stable form on exposure to atmosphere. Additionally, the HF produced by the UF_6 / water vapor reaction is also retained in the uranic breakdown to give an overall reaction represented by:



For the NCSE calculations, the composition of the breakdown product is simplified to $\text{UO}_2\text{F}_2 \cdot 3.5\text{H}_2\text{O}$ that represents the same H/U of 7 as above.

The determination of the Passive Engineered Control (PEC) values for the major controlling parameters used to control criticality in the facility is described below.

The following criticality analyses are performed assuming optimum moderation (i.e., various H/U ratios greater than and less than 7 are analyzed) and 30 cm water reflection to determine the critical and safe values for various parameters. Although the facility will be limited to 10.0 wt.% enrichment, as additional conservatism, the analysis is also performed at 11.0 wt.% enrichment and the PEC values for the buildings/systems/components are developed based on the analysis at 11.0 wt.% enrichment, for k_{eff} of 0.95, except for the Contingency Dump System upset events (both the dump traps and to tails cylinders) which are limited to 1.5 wt.% ^{235}U for the dumped material.

A uniform solution of UO_2F_2 , and a fixed enrichment, 10.0 wt.% and 11.0 wt.% enrichment, are conservatively modeled using SCALE 6.2.4 (ORNL 2020) and the ENDF/B-VII.1 cross-section library.

Table 5-2 shows the results of the criticality analyses for $k_{\text{eff}} = 1$, critical values, and for $k_{\text{eff}} = 0.95$, safe values.

Table 5-3 lists the safety criteria of Table 5-2 which are used as PEC control parameters to prevent a nuclear criticality event. The product cylinders are only safe under conditions of limited moderation and enrichment. In such cases, both design and operating procedures are used to ensure that these limits are not exceeded. Centrifuge array criticality is precluded by a probability argument with multiple operational procedure barriers.

**Table 5-2 Critical and Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2
[Security-Related Information Withheld Under 10 CFR 2.390]**

Table 5-3 Safety Criteria for Buildings / Systems / Components
[Security-Related Information Withheld Under 10 CFR 2.390]

The major controlling parameters used in the Facility are enrichment control, geometry control, moderation control, and/or limitations on the mass as a function of enrichment. In addition, reflection, interaction, and heterogeneous effects are important parameters considered and applied where appropriate in NCSEs. NCSEs are used to identify the significant parameters affected within a particular system. All assumptions relating to process, equipment, material function, and operation, including credible abnormal conditions, are justified, documented, and independently reviewed.

The NCSE captures the criticality calculations performed to determine the critical and safe limits of the Facility as well as the control parameters to prevent a nuclear criticality event.

Based on the criticality analyses, the control parameters applied to the Facility are as follows:

Enrichment

Enrichment is controlled to limit the percentage ^{235}U within any process, vessel, or container to a maximum enrichment of 10 wt.% except for the contingency dump system and for certain non-PEC tanks. The design of the contingency dump system controls enrichment to a limit of 1.5 % ^{235}U . The enrichment may further be restricted in non-PEC tanks to ≤ 1.0 wt.% ^{235}U .

Although the Facility is limited to a maximum enrichment of 10 wt.%, as added conservatism, nuclear criticality safety is analyzed using an enrichment of 11 wt.% ^{235}U .

Geometry/Volume

Geometry / volume control may be used to ensure criticality safety within specific process operations or vessels, and within storage containers.

Geometry control is implemented by limiting equipment dimensions in systems whose criticality safety relies on maintaining specific geometric configurations.

Prior to beginning operations, in response to changes to operations, and at periodic intervals, all dimensions relied on in demonstrating subcriticality are verified. Relevant dimensions and material properties are maintained in the facility's configuration-management program.

Credible mechanisms for loss of geometry control-such as corrosion, leakage, bulging, transfer into unfavorable geometry, or changes to a more reactive physicochemical form-are evaluated, and controls are established as necessary.

Neutron interaction with any equipment containing fissile material is considered in the demonstration of subcriticality unless the criteria for neutronic isolation can be met or justified.

The geometry / volume limits are chosen to ensure $k_{\text{eff}} < 0.95$.

The PEC values of geometry / volume define the characteristic dimension of importance for a single unit such that nuclear criticality safety is not dependent on any other parameter assuming 11 % ^{235}U for safety margin.

Volume limits are used as specified in NCSEs. PEC value of geometry / volume is evaluated assuming the most reactive credible geometry (sphere), optimum moderation, and full water reflection.

When volume control is used, the size of the containers is ensured through the Criticality Management Program and/or by procedural requirements specifying the containers permitted for fissile-material operations.

Moderation

Physical structures are designed to preclude the ingress of moderators.

Water and oil are the moderators considered in the Facility. Moderation control for product cylinders is established consistently with the guidelines of ANSI/ANS-8.1-2014 (ANSI 2023a) and incorporates the criteria below:

- Controls are established to limit the amount of moderation entering the cylinders.
- When moderation is the only parameter used for criticality control, the following additional criteria are applied. These controls ensure that at least two independent controls would have to fail before a criticality accident is possible.
 - Two independent controls are utilized to verify cylinder moderator content.
 - These controls are established to monitor and limit uncontrolled moderator prior to returning a cylinder to production thereby limiting the amount of uncontrolled moderator from entering system to an acceptable limit.
 - The evaluation of the cylinders under moderation control includes the establishment of limits for the ratio of maximum moderator-to-fissile material for both normal operating and credible abnormal conditions. This analysis has been supported by parametric studies.
- When moderation is not considered a control parameter, either optimum moderation or worst-case H/U ratio is assumed when performing criticality safety analysis.

Firefighting procedures for use in moderation-controlled areas are evaluated in CSEs. Restrictions on the use of moderating firefighting agents are included in procedures and training. The effects of a fire and activation of fire suppression are evaluated.

The centrifuge process equipment is a closed system designed to treat gaseous UF₆. The closed system minimizes the introduction of moderation.

Mass

Mass control may be utilized to limit the quantity of uranium within specific process operations, vessels, or storage containers. Mass control may be used on its own or in combination with other control methods. Weighing or physical measurements is employed to verify the mass of the material.

Conservative administrative limits for each operation are specified in the operating procedures.

When the dimensions of equipment or containers with a fixed geometry are used to limit the mass of the material, a conservative process density is used to calculate the resulting mass.

Whenever mass control is established for a container, records are maintained for mass transfers into and out of the container. Establishment of mass limits for a container involves consideration of potential moderation, reflection, geometry, spacing, and enrichment. The evaluation considers normal operations and credible abnormal conditions for determination of the operating mass limit for the container and for the definition of subsequent controls necessary to prevent reaching the safety limits. When only administrative controls are used for mass-controlled systems, double batching is conservatively assumed in the analysis.

In the absence of a dedicated control, an item containing enriched uranium is assumed to hold the maximum credible 235-U content based on its available volume.

Reflection

Normal and credible abnormal reflection is considered when performing NCSEs.

The possibility of full water reflection is considered when performing analyses, for example for the determination of PEC values when reflection is not controlled. It is recognized that concrete can be a more efficient reflector than water, and its potential presence is considered.

Minimum reflection conditions equivalent to a 1-inch tight-fitting water reflector are assumed to account for personnel and other transient incidental reflectors not explicitly included with fixed reflectors in the model.

Interspersed moderation is considered when evaluating subcriticality for an array of units.

Reflection controls are used to limit the potential reactivity of a fissile material operation.

Interaction

NCSEs consider the potential effects of interaction, including interaction effects of in-transit materials. Spacing requirements will be determined on a system-by-system basis. Individual unit multiplication, array interaction, and in-transit material interactions are evaluated in the NCSEs.

Passive engineered controls are used to the extent possible to ensure spacing requirements are maintained. When implemented, the structural integrity of spacers, storage racks, etc. is sufficient to ensure subcriticality under normal and credible abnormal conditions. Engineered devices that are moveable (e.g., birdcage drums, 55-gallon drums) are inspected periodically for deformation.

5.7.4 Safety Margins Against Criticality

The Facility UF₆ systems involve mostly gaseous operations. These operations are carried out under reduced atmospheric conditions (vacuum) or at slightly elevated pressures. The following contributes to the safety margins against criticality:

- It is highly unlikely that any size changes of process equipment under these conditions would lead to a criticality situation because a volume or mass limit may be exceeded.
- The Facility design has minimized the possibility of accidental moderation by eliminating direct water contact with accumulated UF₆.
- The Facility's stringent procedural controls for enriching UF₆ ensure that it does not become unacceptably hydrogen moderated while in process. The Facility's UF₆ systems operating procedures contain safeguards against loss of moderation control, ANSI/ANS-8.1-2014 (ANSI 2023a). No neutron poisons are relied upon to assure criticality safety.
- All the equipment and cylinders accumulating product and feed UF₆ are within enclosures protecting them from water ingress.

5.8 ADDITIONAL NCS PROGRAM COMMITMENTS

No additional NCS program commitments are made beyond the commitments already made previously.

5.9 REFERENCES

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