ENCLOSURE 3
Mark-up Pages to the
Safety Analysis Report
(LAR 24-01)

SAFETY ANALYSIS REPORT

Markup pages for LAR 24-01

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then released through an exhaust stack. The exhaust stack flow is continuously monitored for alpha and HF. The stack exhaust is periodically sampled. The continuous monitoring and periodic sampling is in accordance with the guidance in Regulatory Guide 4.16.

3.4.34 Electrical Systems Design

The Electrical System design complies with the following codes and standards.

- IEEE C2, National Electrical Safety Code
- New Mexico Electric Code (based on the National Electric Code, NFPA 70)
- NFPA 70E, Standard for Electrical Safety in the Workplace

3.4.35 Batch Transfers

Batch transfers into and out of enriched uranium-bearing tanks or trains that are not favorable ~~“safe-by size’ or “safe by physical arrangement”~~ will be controlled in one of two ways based on the type of batch operation being performed. There are two types of batch operations that are considered. The first type is liquid transfer between tanks with enriched uranic material present in the liquid. The second is transferring a number of components into a tank or train with enriched uranic material contained within or on the components transferred in a batch operation.

- **Liquid Transfers into non-favorable~~-SBD~~ tanks**

  Liquid Transfers of solutions containing enriched uranium, above threshold limits, into non-favorable~~-SBD~~ tanks will utilize two independent IROFS for prevention of a nuclear criticality event. These IROFS provide for mass control and /or enrichment control (referred to as “sample and analyzed”), and controls referred to as “bookkeeping measures” control or combine them via a~~n~~ multiple of redundant IROFS.

  The “sampled and analyzed” is a process to calculate the potential mass of enriched uranium (^{235}U) that will be transferred into the receiving tank to ensure that the receiving tank will hold less than a safe mass of enriched uranium after the transfer and/or that the enrichment level in tank is maintained in accordance with the IROFS limits~~ specified in Table 5.1-2~~.

  Bookkeeping measures are a documented running inventory of the ^{235}U mass or uranium enrichment level in a particular tank. Readying the tank for receipt of enriched uranic material includes accounting for any enriched uranium heel or deposits that remain in the tank, as necessary.

  Prior to transfer, the proposed inventory of the receiving tank is calculated based on the “sampled and analyzed” value in the transferring tank and the current inventory of the receiving tank. The bookkeeping data is compared to the specified limit to determine if the IROFS acceptance criteria are met. This quantity is credited/debited to/from each tank as appropriate, depending on the amount of effluent transferred (bookkeeping). Bookkeeping data may be updated via additional sampling and analysis.

- **Transferring a number of components into a non-favorable ~~-SBD~~ tank with enriched uranic material contained within or on the components**

  Transfers of components containing uranic material, above threshold limits, into non-favorable~~-SBD~~ tanks are performed in the Decontamination Workshop. These transfers will utilize two independent IROFS for prevention of a nuclear criticality event. These IROFS provide for mass control and/or enrichment control (referred to as "sampled and analyzed") and controls referred to as "bookkeeping measures." These IROFS may separately apply the "sampled and analyzed" control and the "bookkeeping measures" control or combine them via an application of multiple IROFS. Additional IROFS controls are on the dimensional parameters of the Multi-Functional Decontamination Train.

  The "sampled and analyzed" is a process to calculate or conservatively estimate the potential mass of enriched uranium (^{235}U) and/or enrichment in a tank prior to introducing the batch of components to ensure the tank is maintained in accordance with the IROFS limits~~ specified in Table 5.1-2~~.

  Bookkeeping measures are a documented running inventory of the ^{235}U mass and/or enrichment in a particular tank that contains a solution of enriched uranium. Readying the tank for receipt of enriched uranic material includes accounting for any enriched uranium heel or deposits that remain in the tank.

Prior to transfer of a batch of components into a tank, the proposed inventory of the receiving tank is determined based on the "sampled and analyzed" values for the components and the current inventory of the receiving tank. The bookkeeping data is compared to the specified limit to determine if the IROFS acceptance criteria are met. Bookkeeping data may be updated via additional sampling and analysis.

3.4.36 UF$_6$ cylinders with Faulty Valves

UF$_6$ cylinders with faulty valves are serviced in the Ventilated Room. In the Ventilated Room, the faulty valve is removed and the threaded connection in the cylinder is inspected. A new valve is then installed in accordance with the requirements of ANSI N-14.1.

3.4.37 IROFS

IROFS will be designed, constructed, tested and maintained to QA Level 1, with the following exceptions,

3.4.38 IROFS27e

IROFS27e which will be designated and analyzed to QA Level 1, and will be constructed, tested, and maintained to QA Level 1 Graded.

- Fire protection features designated as IROFS which will be designed, procured, constructed, tested, and maintained to QA Level 1-Fire Protection (QL-1F)

3.4.43 Product Station Design

The Product Stations design will be based on ETC4069917-1 design drawings. The internal station design size of approximately 9’7” does not accommodate a 48-inch feed cylinder. Blending donor and receiver station designs do not accommodate 48-inch cylinders. Product cylinders, as designed, cannot physically connect to a feed station. Therefore, potential for re-feeding enriched materials does not exist. Future construction and design efforts will be consistent. Any modification to station designs or product cylinder connection points will be re-evaluated and revised consistent with overall ISA methodology including criticality reviews.

3.4.44 Assay Sampling Rig

The Assay Sampling Rig shall exhaust to a gaseous effluent vent system ~~with safe-by-design attributes~~. ~~At final design, this~~ This rig ~~will be~~ was evaluated for criticality concerns and IROFS or other controls will be identified in compliance with 10 CFR 70.61.

3.4.45 Administrative Control IROFS Support Equipment

Administrative Control IROFS Support Equipment contain attributes that are required by the worker to fulfill the Administrative Control IROFS unless the IROFS class is Redundant. The attributes are verified to ensure that the worker can perform the IROFS safety function. Support Equipment is in the Administrative Control IROFS boundary. Many of the actions are to prevent an event and upon failure of indication, actions would be implemented to stop continued operation or not start the operation. However, to enhance worker action and direction to prevent events, Support Equipment was identified and included in the boundary. The attributes of Support Equipment are controlled through the applicable management measures. For example, the attribute of “accurate and reliable indication” is controlled through the calibration and testing which is part of the Maintenance Function Testing Program.

Support Equipment is listed in Table 3.4-1, Administrative Control IROFS Support Equipment. This table contains Support Equipment and other equipment, other equipment is not inside the Administrative Control IROFS boundary; normally such equipment is QL-3. Equipment Attributes are in the Administrative Control IROFS boundary.

Management measures are applied to the attributes of Administrative Control IROFS Support Equipment and other equipment attributes. Management measures are also applied to Administrative Control IROFS Support Equipment as defined in the Quality Assurance Program Description for QL-2AC equipment.

3.4.46 IROFS Classes

For the three types of IROFS Administrative, Passive Engineered and Active Engineered there are three classes which the IROFS can fall into. An IROFS cannot fall into multiple classes, i.e., separate classes for different accident scenarios depending on the IROFS applied to each accident scenario.

Sole IROFS: A Sole IROFS is an IROFS that upon its failure there are no remaining IROFS capable of preventing or mitigating the consequences for the accident scenario for which the IROFS applies.

5.1 The Nuclear Criticality Safety (NCS) Program

The facility has been designed and will be constructed and operated such that a nuclear criticality event is prevented, and to meet the regulatory requirements of 10 CFR 70 (CFR, 2003a). Nuclear criticality safety at the facility is assured by designing the facility, systems and components with safety margins such that safe conditions are maintained under normal and abnormal process conditions and any credible accident. Items Relied On For Safety (IROFS) identified to ensure subcriticality are discussed in the UUSA Integrated Safety Analysis Summary.

5.1.1 Management of the Nuclear Criticality Safety (NCS) Program

The NCS criteria in Section 5.2, Methodologies and Technical Practices, are used for managing criticality safety and include adherence to the double contingency principle as stated in the ANSI/ANS-8.1, Nuclear Criticality Safety In Operations with Fissionable Materials Outside Reactors. 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 UUSA meets the double contingency principle. The UUSA meets the double contingency principle in that process design incorporates sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

The plant will produce uranium enriched in isotope ^{235}U no greater than the LES license limit. However, as additional conservatism, most nuclear criticality safety analyses for enriched material are performed assuming a ^{235}U enrichment of ~~6.0 $^{w}/_{o}$ and~~ 11.0 $^{w}/_{o}$ (for LEU+, except as noted), and include appropriate margins to safety. The exceptions to this are; the systems and components associated with a cascade dump, these include the Contingency Dump System equipment and piping on the 2nd floor of the Process Services Area and the Tails Take-off System, ~~-~~which are analyzed at various bounding enrichment levels for the specific system or component with the system and non-favorable~~-Safe-By-Design~~ tanks which may be limited to 1.0 $^{w}/_{o}$ ^{235}U. In accordance with 10 CFR 70.61(d) (CFR, 2003b), the general criticality safety philosophy is to prevent accidental uranium enrichment excesses, provide geometrical safety when practical, provide for moderation controls within the UF_6 processes and impose strict limits on containers of aqueous, solvent based, or acid solutions containing uranium with greater than established threshold values~~, where the limits are specified in Table 5.1-2~~. Interaction controls provide for safe movement and storage of components. Plant and equipment features assure prevention of excessive enrichment. The plant is divided into distinctly separate Assay Units (called Cascade Halls) with no common UF_6 piping. UF_6 blending is done in a physically separate portion of the plant. ~~Process piping, individual centrifuges and chemical traps other than the contingency dump chemical traps, are safe by limits placed on their diameters.~~ Product cylinders rely upon uranium enrichment, moderation control and mass limits to protect against the possibility of a criticality event. Each of the ~~liquid effluent collection~~ tanks that hold uranium in solution are controlled via enrichment, mass, geometry (e.g., diameter, slab thickness), neutron absorption, interaction or volume.~~one of the mechanisms specified in Table 5.1-2.~~ As required by 10 CFR 70.64(a) (CFR, 2003c), by observing the double contingency principle throughout the plant, a criticality accident is prevented. In addition to the double contingency principle, effective management of the NCS Program includes:

Moderation control is in accordance with ANSI/ANS-8.22, Nuclear Criticality Safety Based on Limiting and Controlling Moderators. However, for the purposes of the criticality analyses, it is assumed that UF_6 comes in contact with water to produce aqueous solutions of UO_2F_2 as described in Section 5.2.1.3.3, Uranium Accumulation and Moderation Assumption. A uniform aqueous solution of UO_2F_2, and a fixed enrichment are conservatively modeled using ~~MONK 8A and the JEF2.2 library. Criticality analyses were performed using Monk at 6 $^w/_o$ ^{235}U to determine the maximum value of a parameter to yield $k_{eff} = 1$. The criticality analyses were then repeated to determine the maximum value of the parameter to yield a $k_{eff} = 0.95$.~~

~~Similarly, Criticality analyses were performed using~~ MCNP at 11 $^w/_o$ ^{235}U to determine the maximum value of a parameter to yield $k_{eff} = 0.99180$. The criticality analyses were then repeated to determine the maximum value of the parameter to yield a $k_{eff} = 0.958$.

Table 5.1-1, Critical and Safe Values for Uniform Aqueous Solution of Enriched UO_2F_2 at 11 $^w/_o$, shows both the critical and safe limits ~~for 6.0 $^w/_o$ ^{235}U (based on Monk analysis) and~~ at 11.0 $^w/_o$ enrichment (based on MCNP analysis).

~~Table 5.1-2, Safety Criteria for Buildings/ Systems/Components, lists the safety criteria of Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2, which are used as control parameters to prevent a nuclear criticality event. Although UUSA is limited to 5.5 $^w/_o$ enrichment (for non-LEU+ operations), as additional conservatism, the values in the first half of Table 5.1-2, Safety Criteria for Buildings/Systems/ Components, represent the limits based on 6.0 $^w/_o$ enrichment except for the Contingency Dump System equipment and piping on the 2nd floor of the Process Services Corridor and the Tails Take-off System which are limited to 1.5 $^w/_o$ ^{235}U and non-Safe-By-Design tanks which may be limited to 1.0 $^w/_o$ ^{235}U.~~

~~Table 5.1-2 is not applicable to LEU+ systems and components.~~ The nuclear safety of LEU+ systems and components is not based on single item safety criteria, but rather by overall analysis of the configuration.

The values on Table 5.1-1 are chosen to be critically safe when optimum light water moderation exists and reflection is considered within isolated systems. The conservative modeling techniques provide for more conservative values than provided in ANSI/ANS-8.1. The product cylinders are only safe under conditions of limited moderation and enrichment. In such cases, both design and operating procedures are used to assure that these limits are not exceeded.

All Separation Plant components which handle enriched UF_6, including product cylinders and contingency dump chemical traps, are criticality safe based on analysis. Centrifuge array criticality is precluded by a probability argument with multiple operational procedure barriers. Total moderator or H/U ratio control as appropriate precludes product cylinder criticality.

In the Cylinder Receipt and Dispatch Building criticality safety for uranium loaded liquids is controlled via enrichment, mass, geometry (e.g., diameter, slab thickness), neutron absorption, interaction or volume.~~one of the mechanisms specified in Table 5.1-2.~~ Individual liquid storage bottles are safe by volume, unless exempted from NCS program requirements. Interaction in storage arrays is accounted for.

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

Enrichment

Enrichment is controlled to limit the percent ^{235}U within any process vessel or container to a maximum of the LES license limit except for the systems and components associated with a cascade dump and in certain non-favorable ~~-Safe-By-Design~~ tanks noted below. For added conservatism the systems controlled to the LES license limit in isotope ^{235}U are analyzed at ~~6 $^{w}/_{o}$ and~~ 11 $^{w}/_{o}$, except as previous noted. The enrichment level may further be restricted in non-favorable ~~-Safe-By-Design~~ tanks (e.g., Bulk Storage Tanks, Release Tanks, and Totes) to ≤ 1.0 $^{w}/_{o}$ ^{235}U.

For added conservatism, ~~for enrichments equal to or greater than 6 $^{w}/_{o}$ specific only to higher enrichment processes,~~ UUSA analyzes at an enrichment value of 1 $^{w}/_{o}$ higher (e.g., LES license limit of 10 $^{w}/_{o}$ - UUSA analyzes at 11 $^{w}/_{o}$) than the license imit. The exception is for systems where enrichment is the only control used for NCS (e.g., waste storage or off-site shipping from the LECTS[5] – bulk storage tanks, totes, drums, etc.).

Geometry/Volume

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

The geometry/volume limits are chosen to ensure ~~$k_{eff} = k_{calc} + 3\ \sigma_{calc} < 0.95$ for MONK 8A applications and~~ $k_{eff} = k_{calc} + 2\sigma_{calc} < 0.958$ for MCNP6 applications.

The safe values of geometry/volume in Table 5.1-1 define the characteristic dimension of importance for a single unit such that nuclear criticality safety is not dependent on any other parameter assuming ~~6~~ 11 $^{w}/_{o}$ ^{235}U for safety margin for UUSA operations.

Moderation

Water and oil are the moderators considered at UUSA. Moderation control for product cylinders is established consistent with the guidelines of ANSI/ANS-8.22 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 assure 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.

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[5] Other conservatisms (e.g., moderation, reflection, material) apply to LECTS.

Neutron Absorbers

Neutron Absorption is a factor in almost all of the materials at UUSA. The normal absorption of neutrons in standard materials used in the construction and processes ~~-~~(uranium, fluorine, water, steel, etc.) is not specifically excluded as a criticality control parameter.

Models incorporate conservative values (e.g., material compositions and equipment dimensions), which are validated at receipt, after installation or during surveillances. In particular analyses, the models include structural material that is credited for neutron absorption (i.e., Roots pump cases, Type A Chemical Trap walls, Slab Tanks shell (walls), and walls of SCDT baths, walls of UF_6 product cylinders, walls of A1 and 1S UF_6 sample cylinders).

Additional materials such as cadmium and boron for which the sole purpose would be to absorb neutrons are not incorporated in UUSA processes. Solutions of absorbers are not used as a criticality control mechanism.

Piece Count

Piece count, which refers to the number of uranic bearing components being modeled may be used as a control. When used as a control, the safe number of components can be established, for example, by dividing the safe mass of a single parameter or safe volume by the component's uranic mass or volume the safe mass of a single parameter or safe volume, respectively.

Concentration and Density

UUSA does not use either concentration or density as a criticality control parameter.

5.1.3 Safe Margins against Criticality

Process operations require establishment of criticality safety limits. The facility UF_6 systems involve mostly gaseous operations. These operations are carried out under reduced atmospheric conditions (vacuum) or at slightly elevated pressures not exceeding three atmospheres. It is highly unlikely that any size changes of process piping, cylinders, cold traps, or chemical traps under these conditions, would lead to a criticality situation because a volume or mass limit may be exceeded.

Within the Separations Building Modules, significant accumulations of enriched UF_6 reside only in the product cylinders and cold traps. The facility design minimizes the possibility of accidental moderation by eliminating water for automatic fire suppression. In addition, the facility's design assures that product cylinders and cold traps do not become unacceptably hydrogen moderated while in process. The plant's UF_6 systems operating procedures contain safeguards against loss of moderation control (ANSI/ANS 8.22).

5.1.4 Description of Safety Criteria

Each portion of the plant, system, or component that may possibly contain enriched uranium is designed with criticality safety as an objective. ~~Table 5.1-2, Safety Criteria for Buildings/ Systems/Components, shows how the safety criteria of Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2, are applied to the facility to prevent a nuclear criticality event. Although UUSA will be limited to Material License Condition 6B for ${}^{w}/_{o}$ enrichment, as additional conservatism, the values in Table 5.1-2, represent the limits based on 6.0 ${}^{w}/_{o}$ enrichment with the previously noted exceptions.~~ The safety criteria of Table 5.1-1, Critical and

Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2 at 11 $^w/_o$, are applied through analyses that calculate k_{eff} of the plant systems to prevent a criticality event.

Where there are significant in-process accumulations of enriched uranium as UF_6, the plant design includes multiple features to minimize the possibilities for breakdown of the moderation control limits. These features eliminate direct ingress of water to product cylinders while in process.

5.1.5 Organization and Administration

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

The Engineering and Projects Manager is accountable for overall criticality safety of the facility, is administratively independent of production responsibilities, and has the authority to shut down potentially unsafe operations.

Designated responsibilities of the Criticality Safety Organization include the following:

- Establish the Nuclear Criticality Safety Program, including design criteria, procedures, and training
- Assess normal and credible abnormal conditions
- Determine criticality safety limits for controlled parameters, with input from the Criticality Safety Engineers
- Develop and validate methods to support nuclear criticality safety evaluations (NCSEs) (i.e., non-calculation engineering judgments regarding whether existing criticality safety analyses bound the issue being evaluated or whether new or revised safety analyses are required)
- Specify criticality safety control requirements and functionality
- Provide advice and counsel on criticality safety control measures
- Support emergency response planning and events
- Evaluate the effectiveness of the Nuclear Criticality Safety Program using audits and assessments
- Provide criticality safety postings that identify administrative controls for operators in applicable work areas.

Criticality Safety Engineers will be provided in sufficient number to support the program technically. They are responsible for the following:

- Provide criticality safety support for integrated safety analyses and configuration control
- Perform NCS analyses (i.e., calculations), write NCS evaluations, and approve proposed changes in process conditions on equipment involving fissionable material

Qualified Criticality Safety Engineers may also perform tasks associated with Criticality Safety program implementation and assessment.

The minimum qualifications for the Criticality Safety Engineer are described in Section 2.2.3. The Criticality Safety Engineer training program is based on ANSI/ANS-8.26, Criticality Safety Engineer Training and Qualification Program. The Engineering and Projects Manager has the authority and responsibility to assign and direct activities for the Criticality Safety Program. The Engineering and Projects Manager is responsible for implementation of the NCS program.

Δ_{SM} is taken as 0.03 per justification provided in the UUSA MCNP6 Validation report (Sanders Engineering, 2022). Δ_{AOA} is set to zero, as the benchmark experiments encompass the range of actual applications at UUSA with the exception of the enrichment variable. The enrichment of the current Contingency Dump System in the previous NCSA is 1.5 $^{w}/_{o}$ ^{235}U (remains bounding of 1.6 $^{w}/_{o}$ value used in LEU+), while the lowest enrichment used in the benchmark calculations is 2 $^{w}/_{o}$. For enrichments between 0 and 2 $^{w}/_{o}$ ^{235}U, NUREG/CR-6698 Table 2.3 provides an allowable experimental range of ± 1.5 $^{w}/_{o}$ for the areas of applicability (NRC, 2001). The highest enrichment used in the benchmark calculations is 47 $^{w}/_{o}$ while future UUSA application may require enrichments up to 50 $^{w}/_{o}$. NUREG/CR-6698, Table 2.3 allows for a ±15 $^{w}/_{o}$ extension for benchmarks with enrichments between 20-80 $^{w}/_{o}$ ^{235}U (NRC, 2001). Accordingly, Δ_{AOA} with respect to an enrichment range of 0.5-50 $^{w}/_{o}$ ^{235}U is taken as 0.0.

The USL becomes:

$$USL = 0.98894 - 0.03 - 0.0 = 0.958 \text{ (for enrichments of 0.5 to } \leq 50\ ^{w}/_{o}\ ^{235}U)$$

NUREG/CR-6698 indicates that the following condition be demonstrated for all normal and credible abnormal operating conditions (NRC, 2001):

$$k_{calc} + 2\ \sigma_{calc} < USL$$

For the systems or components with enrichments of 0.5 up to ≤ 50 $^{w}/_{o}$, the nuclear criticality safety criterion for MCNP6 is given by:

$$k_{eff} = k_{calc} + 2\sigma_{calc} < 0.958$$

5.2.1.3 General Nuclear Criticality Safety Methodology

The NCS analyses results provide values of k-effective (k_{eff}) to conservatively meet the upper safety limit. The following sections provide a description of the major assumptions used in the NCS analyses.

5.2.1.3.1 Reflection Assumption

The layout of the NEF is a very open design and it is not considered credible that those vessels and plant components requiring criticality control could become flooded from a source of water within the plant. Full water reflection of vessels has therefore been discounted. However, where appropriate, spurious reflection due to walls, fixtures, personnel, etc. has been accounted for by assuming 2.5 cm (0.984 in) of water reflection around vessels.

5.2.1.3.2 Enrichment Assumption

Enrichment is controlled to limit the percent ^{235}U within any process vessel or container to the LES license limit. For added conservatism most systems controlled to the LES license limit in isotope ^{235}U are analyzed at ~~6 $^{w}/_{o}$ and~~ 11 $^{w}/_{o}$ (for LEU+), except as previously noted.

5.2.1.3.3 Uranium Accumulation and Moderation Assumption

Most components that form part of the centrifuge plant or are connected to it assume 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 nuclear criticality safety analyses 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. 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. The 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:

$$UF_6 + 2H_2O \rightarrow UO_2F_2 + 4HF$$

Due to its hygroscopic nature, the resulting uranyl fluoride is likely to form a hydrate compound. Experimental studies (Lychev, 1990) suggest that solid hydrates of compositions $UO_2F_2 \cdot 1.5H_2O$ and $UO_2F_2 \bullet 2H_2O$ can form in the presence of water vapor, the former composition being the stable form on exposure to atmosphere.

It is assumed that the hydrate $UO_2F_2 \bullet 1.5H_2O$ is formed and, additionally, that the HF produced by the UF_6/water vapor reaction is also retained in the uranic breakdown to give an overall reaction represented by:

$$UF_6 + 3.5H_2O \rightarrow UO_2F_2 \bullet \cdot 4HF \bullet 1.5H_2O$$

For the criticality safety calculations, the composition of the breakdown product was simplified to $UO_2F_2 \bullet 3.5H_2O$ that gives the same H/U ratio of 7 as above.

In the case of oils, UF_6 pumps and vacuum pumps use a fully fluorinated perfluorinated polyether (PFPE) type lubricant. Mixtures of UF_6 and PFPE oil would be a less conservative case than a uranyl fluoride/water mixture, since the maximum HF solubility in PFPE is only about 0.1 $^{w}/_{o}$. Therefore, the uranyl fluoride/water mixture assumption provides additional conservatism in this case.

5.2.1.3.4 Vessel Movement Assumption

The limits placed on movement of an individual vessel or a specified batch of vessels containing enriched uranium are specified in the facility procedures or work plans, both of which are reviewed by Nuclear Criticality Safety. Specified limits may not be required based on bounding or process/system-specific NCS evaluations or analysis.

Of the subset of individual vessels or groups of vessels that do not have specified controls, but are bounded by ~~a~~ the single-parameter ~~SBD limits~~NCS values in Table 5.1-1, separation must be maintained at least 60 cm (23.6 in) from any other enriched uranium.

Vessels or groups of vessels that do not comply with either of the statements above must not be moved without the written approval of the Criticality Safety Organization.

5.2.1.3.5 Pump Free Volume Assumption

There are various types of pumps used in the UF_6 process systems in the plant, such as rotary vane, Roots, molecular or ion, and diffusion pumps.

These pumps are subjected to Quality Control verifications, either at the respective factory or on-site at the NEF.

Any one of the following methods is used to establish the pump volumes for criticality safety analysis:

- A bounding internal free volume of the pump is established as determined by volumetric testing of the pump.
- Drawings and external measurements are used to determine the maximum bounding value for the volume.
- An explicit model of the pump may be created, with verification of specific criteria.

5.2.1.4 Nuclear Criticality Safety Analyses

Nuclear criticality safety is analyzed for the design features of the plant system or component and for the operating practices that relate to maintaining criticality safety. The analysis of individual systems or components and their interaction with other systems or components containing enriched uranium is performed to assure the criticality safety criteria are met. The nuclear criticality safety analyses and the safe values in Table 5.1-1, Critical and Safe Values for Uniform Aqueous Solution of Enriched UO_2F_2 at 11 $^{w}/_{o}$, provide a basis for the plant design and criticality hazards identification performed as part of the Integrated Safety Analysis.

Each portion of the plant, system, or component that may possibly contain enriched uranium is designed with criticality safety as an objective. ~~Table 5.1-2, Safety Criteria for Buildings/ Systems/Components, shows how the safe values of Table 5.1-1, are applied to the facility design to prevent a nuclear criticality event. The NEF is designed and operated in accordance with the parameters provided in Table 5.1-2.~~ The Integrated Safety Analysis reviewed the facility design and operation and identified Items Relied On For Safety to ensure that criticality does not pose an unacceptable risk.

Where there are significant in-process accumulations of enriched uranium as UF_6 the plant design includes multiple features to minimize the possibilities for breakdown of the moderation control limits. These features eliminate direct ingress of water to product cylinders while in process.

Each NCS analysis includes, as a minimum, the following information.

- A discussion of the scope of the analysis and a description of the system(s)/process(es) being analyzed.
- A discussion of the methodology used in the criticality calculations, which includes the validated computer codes and cross section library used and the k_{eff} limits used (0.95 for MONK 8A and 0.958 for MCNP6).
- A discussion of assumptions (e.g. reflection, enrichment, uranium accumulation, moderation, movement of vessels, component dimensions) and the details concerning the assumptions applicable to the analysis.

5.6 Chapter 5 Tables

Table 5.1-1 Critical and Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2 at ~~6.0 $^w/_o$ and~~ 11.0 $^w/_o$

~~Values for 6.0 $^w/_o$ enrichment~~			
~~Parameter~~	~~Critical Value~~ ~~k_{eff} = 1.0~~	~~Safe Value~~ ~~k_{eff} = 0.95~~	~~Safety Factor~~
~~Volume~~	~~25.3 L (6.7 gal)~~	~~19.3 L (5.1 gal)~~	~~0.76~~
~~Cylinder Diameter~~	~~24.8 cm (9.8 in)~~	~~22.4 cm (8.8 in)~~	~~0.90~~
~~Slab Thickness~~	~~11.6 cm (4.6 in)~~	~~10.1 cm (4.0 in)~~	~~0.87~~
~~Areal Density~~	~~9.4 g U/cm^2 (19.3 lb U/ft^2)~~	~~7.9 g U/cm^2 (16.2 lb U/ft^2)~~	~~0.84~~
~~Uranium Mass~~	~~27 kg U (59.5 lb U)~~		
~~- no double batching~~		~~20.1 kg U (29.7 kg UF_6)~~	~~0.74~~
~~- double batching~~		~~12.2 kg U (26.9 lb U)~~	~~0.45~~
Values for 11.0 $^w/_o$ enrichment			
Parameter	**Critical Value** **k_{eff} = 0.99339**	**Safe Value** **k_{eff} = 0.958**	**Safety Factor**
Volume	15.3 L (4.0 gal)	12.8 L (3.3 gal)	0.84
Cylinder Diameter	20.50 cm (8.0 in)	19.10 cm (7.5 in)	0.93
Slab Thickness	8.85 cm (3.4 in)	8.0 cm (3.1 in)	0.9
Areal Density	4.50 g U/cm^2 (9.2 lb U/ft^2)	4.00 g U/cm^2 (8.1 lb U/ft^2)	0.89
Uranium Mass	10.8 kg U (23.8 lb U)		
- no double batching		8.8 kg U (13.0 kg UF_6)	0.81
- double batching		4.86 kg U (7.19 lb U)	0.45

Table 5.1-2 ~~**Safety Criteria for Buildings/Systems/Components**~~ - Deleted

~~**Values for 6.0 $^{w}/_{o}$ enrichment**~~		
~~**Building/System/Component**~~	~~**Control Mechanism**~~	~~**Safety Criteria**~~
~~Enrichment~~	~~Enrichment~~	~~5.5 $^{w}/_{o}$ (6 $^{w}/_{o}$ ^{235}U used in NCS)~~
~~Product Cylinders~~	~~Moderation~~	~~H < 0.98 kg (2.16 lb)~~
~~UF_6 Piping~~	~~Diameter~~	~~< 22.4 cm (8.8 in)~~
~~Chemical Traps~~	~~Diameter~~	~~< 22.4 cm (8.8 in)~~
~~Product Cold Trap~~	~~Diameter~~	~~< 22.4 cm (8.8 in)~~
~~Contingency Dump System~~ ~~Tails System~~	~~Enrichment~~	~~1.5 $^{w}/_{o}$ ^{235}U (used in NCS)~~
~~Tanks~~ ~~(controlled by any one mechanism listed on the right)~~	~~Diameter~~	~~< 22.4 cm (8.8 in)~~
	~~Enrichment~~	~~≤ 1.0 $^{w}/_{o}$ ^{235}U (used in NCS for non-Safe-By-Design tanks)~~
	~~Mass~~	~~< 0.73 kg ^{235}U~~
	~~Slab Thickness~~	~~< 10.1 cm (4.0 in)~~
	~~Volume~~	~~< 19.3 L (5.1 gal)~~
~~Feed Cylinders~~	~~Enrichment~~	~~< 0.72 $^{w}/_{o}$ ^{235}U~~
~~Uranium Byproduct Cylinders~~	~~Enrichment~~	~~< 0.72 $^{w}/_{o}$ ^{235}U~~
~~UF_6 Pumps~~	~~Volume~~	~~< 19.3 L (5.1 gal)~~
~~Individual Uranic Liquid Containers, e.g., PFPE Oil Bottle, Laboratory Flask, Mop Bucket~~	~~Volume~~	~~< 19.3 L (5.1 gal)~~
~~Vacuum Cleaners~~ ~~Oil Containers~~	~~Volume~~	~~< 19.3 L (5.1 gal)~~