NATIONAL
ENRICHMENT
FACILITY

10 CFR 30.6
10 CFR 40.5
10 CFR 70.5

January 18, 2005

NEF#05-003

ATTN: Document Control Desk
Director
*Office of Nuclear Material Safety and Safeguards*
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Louisiana Energy Services, L. P.
National Enrichment Facility
NRC Docket No. 70-3103

Subject: National Enrichment Facility Safety Analysis Report Chapter 5, "Nuclear Criticality Safety," Updated through Revision 4

References:
1. Letter NEF#03-003 dated December 12, 2003, from E. J. Ferland (Louisiana Energy Services, L. P.) to Directors, Office of Nuclear Material Safety and Safeguards and the Division of Facilities and Security (NRC) regarding "Applications for a Material License Under 10 CFR 70, Domestic licensing of special nuclear material, 10 CFR 40, Domestic licensing of source material, and 10 CFR 30, Rules of general applicability to domestic licensing of byproduct material, and for a Facility Clearance Under 10 CFR 95, Facility security clearance and safeguarding of national security information and restricted data"

2. Letter NEF#04-002 dated February 27, 2004, from R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC) regarding "Revision 1 to Applications for a Material License Under 10 CFR 70, "Domestic licensing of special nuclear material," 10 CFR 40, "Domestic licensing of source material," and 10 CFR 30, "Rules of general applicability to domestic licensing of byproduct material"

3. Letter NEF#04-029 dated July 30, 2004, from R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC) regarding "Revision to Applications for a Material License Under 10 CFR 70, "Domestic licensing of special nuclear material," 10 CFR 40, "Domestic licensing of source material," and 10 CFR 30, "Rules of general applicability to domestic licensing of byproduct material"

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4. Letter NEF#04-037 dated September 30, 2004, from R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC) regarding "Revision to Applications for a Material License Under 10 CFR 70, "Domestic licensing of special nuclear material," 10 CFR 40, "Domestic licensing of source material," and 10 CFR 30, "Rules of general applicability to domestic licensing of byproduct material"

5. Letter NEF#04-040 dated September 30, 2004, R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC) regarding "Clarifying Information Related to Criticality Safety"

6. Letter NEF#04-049 dated November 30, 2004, R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC) regarding "Additional Clarifying Information Related to Criticality Safety

By letter dated December 12, 2003 (Reference 1), E. J. Ferland of Louisiana Energy Services (LES), L. P., submitted to the NRC applications for the licenses necessary to authorize construction and operation of a gas centrifuge uranium enrichment facility. Revision 1 to these applications was submitted to the NRC by letter dated February 27, 2004 (Reference 2). Subsequent revisions (i.e., revision 2 and revision 3) to these applications were submitted to the NRC by letters dated July 30, 2004 (Reference 3) and September 30, 2004 (Reference 4), respectively.

As a result of conference calls concerning criticality safety, held between LES and NRC representatives, additional clarifying information was provided to the NRC by letters dated September 30, 2004 (Reference 5) and November 30, 2004 (Reference 5). These clarifications were provided in the form of revised Safety Analysis Report (SAR) pages. In order to facilitate completion of the review of changes to SAR Chapter 5, "Nuclear Criticality Safety," a complete copy of SAR Chapter 5, updated to include the associated clarifications provided in References 3 and 4, is enclosed.

If you have any questions or need additional information, please contact me at 630-657-2813.

Respectfully,

Daniel G. Green for

R. M. Krich
Vice President – Licensing, Safety, and Nuclear Engineering

Enclosure:
National Enrichment Facility Safety Analysis Report Chapter 5, Updated through Revision 4

cc: T.C. Johnson, NRC Project Manager

**ENCLOSURE**

**National Enrichment Facility**
**Safety Analysis Report**
**Chapter 5**
**Updated through Revision 4**

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# 5.0 NUCLEAR CRITICALITY SAFETY

The Nuclear Criticality Safety Program for the National Enrichment Facility (NEF) is in accordance with U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 3.71, Nuclear Criticality Safety Standards for Fuels and Material Facilities (NRC, 1998). Regulatory Guide 3.71 (NRC, 1998) provides guidance on complying with the applicable portions of NRC regulations, including 10 CFR 70 (CFR, 2003a), 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. The facility is committed to following the guidelines in this regulatory guide for specific ANSI/ANS criticality safety standards with the exception of ANSI/ANS-8.9-1987, "Nuclear Criticality Safety Criteria for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Material." Piping configurations containing aqueous solutions of fissile material will be evaluated in accordance with ANSI/ANS-8.1-1998 (ANSI, 1998a), using validated methods to determine subcritical limits.

The information provided in this chapter, the corresponding regulatory requirements, and the section of NUREG-1520 (NRC, 2002), Chapter 5 in which the NRC acceptance criteria are presented is summarized below.

| Information Category and Requirement | 10 CFR 70 Citation | NUREG-1520 Chapter 5 Reference |
|---|---|---|
| **Section 5.1 Nuclear Criticality Safety (NCS) Program** | | |
| Management of the NCS Program | 70.61(d)<br>70.64(a) | 5.4.3.1 |
| Control Methods for Prevention of Criticality | 70.61 | 5.4.3.4.2 |
| Safe Margins Against Criticality | 70.61 | 5.4.3.4.2 |
| Description of Safety Criteria | 70.61 | 5.4.3.4.2 |
| Organization and Administration | 70.61 | 5.4.3.2 |
| **Section 5.2 Methodologies and Technical Practices** | | |
| Methodology | 70.61 | 5.4.3.4.1<br>5.4.3.4.4<br>5.4.3.4.6 |
| **Section 5.3 Criticality Accident Alarm System (CAAS)** | | |
| Criticality Accident Alarm System | 70.24 | 5.4.3.4.3 |
| **Section 5.4 Reporting** | | |
| Reporting Requirements | Appendix A | 5.4.3.4.7 (7) |

- Safety parameters and procedures will be established.
- The NCS program structure, including definition of the responsibilities and authorities of key program personnel will be provided.
- 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 occurrence of an inadvertent nuclear criticality.
- 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.
- NCS postings will be provided and maintained current.
- NCS emergency procedure training will be provided.
- The NCS baseline design criteria requirements in 10 CFR 70.64(a) (CFR, 2003c) will be adhered to.
- 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 so as to prevent recurrence of unacceptable performance deficiencies in IROFS, NCS function or management measures.
- NCS program records will be retained as described in Section 11.7, Records Management.

Training will be provided to individuals who handle nuclear material at the facility in criticality safety. The training is based upon the training program described in ANSI/ANS-8.20-1991, Nuclear Criticality Safety Training (ANSI, 1991). 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.

Additional discussion of management measures is provided in Chapter 11, Management Measures.

## 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 NEF 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-8.1-1998, Nuclear Criticality Safety In Operations with Fissionable Materials Outside Reactors (ANSI, 1998a). 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 NEF meets the double contingency principle. The NEF 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.

Using these NCS criteria, including the double contingency principle, low enriched uranium enrichment facilities have never had an accidental criticality. The plant will produce no greater than 5.0 $^{w}/_{o}$ enrichment. However, as additional conservatism, the nuclear criticality safety analyses are performed assuming a $^{235}U$ enrichment of 6.0 $^{w}/_{o}$, except for Contingency Dump System traps which are analyzed assuming a $^{235}U$ enrichment of 1.5 $^{w}/_{o}$, and include appropriate margins to safety. 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 mass limits on containers of aqueous, solvent based, or acid solutions containing uranium. Interaction controls provide for safe movement and storage of components. Plant and equipment features assure prevention of excessive enrichment. The plant is divided into six 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 is mass controlled, as none are geometrically safe. 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:

- An NCS program to meet the regulatory requirements of 10 CFR 70 (CFR, 2003a) will be developed, implemented, and maintained.

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### 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 the NEF will be limited to 5.0 $^w/_o$ enrichment, as additional conservatism, the values in Table 5.1-2, represent the limits based on 6.0 $^w/_o$ enrichment.

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. During the design phase, the criticality safety function is performed within the design engineering organization. The criticality safety function for operations is described in the following section.

The criticality safety organization reports to the Health, Safety, and Environment (HS&E) Manager as described in Chapter 2, Organization and Administration. The HS&E 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 staff include the following:

- Establish the Nuclear Criticality Safety Program, including design criteria, procedures, and training
- Provide criticality safety support for integrated safety analyses and configuration control
- Assess normal and credible abnormal conditions
- Determine criticality safety limits for controlled parameters
- 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)

- Perform NCS analyses (i.e., calculations), write NCS evaluations, and approve proposed changes in process conditions on equipment involving fissionable material
- Specify criticality safety control requirements and functionality
- Provide advice and counsel on criticality safety control measures, including review and approval of operating procedures
- 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.

The minimum qualifications for a criticality safety engineer are a Bachelor of Science (BS) or Bachelor of Arts (BA) degree in science or engineering with at least two years of nuclear industry experience in criticality safety. A criticality safety engineer must understand and have experience in the application and direction of criticality safety programs. The HS&E Manager has the authority and responsibility to assign and direct activities for the criticality safety staff. The criticality safety engineer is responsible for implementation of the NCS program. Criticality safety engineers will be provided in sufficient numbers to implement and support the operation of the NCS program.

The NEF implements the intent of the administrative practices for criticality safety, as contained in Section 4.1.1 of American National Standards Institute/American Nuclear Society (ANSI/ANS)-8.1-1998, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors (ANSI, 1998a). A policy will be established whereby personnel shall report defective NCS conditions and perform actions only in accordance with written, approved procedures. Unless a specific procedure deals with the situation, personnel shall report defective NCS conditions and take no action until the situation has been evaluated and recovery procedures provided.

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## 5.2 METHODOLOGIES AND TECHNICAL PRACTICES

This section describes the methodologies and technical practices used to perform the Nuclear Criticality Safety (NCS) analyses and NCS evaluations. The determination of the NCS controlled parameters and their application and the determination of the NCS limits on IROFS are also presented.

### 5.2.1 Methodology

MONK8A (SA, 2001) is a powerful Monte Carlo tool for nuclear criticality safety analysis. The advanced geometry modeling capability and detailed continuous energy collision modeling treatments provide realistic 3-dimensional models for an accurate simulation of neutronic behavior to provide the best estimate neutron multiplication factor, k-effective. Complex models can be simply set up and verified. Additionally, MONK8A (SA, 2001) 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 experiments selected 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 MONK8A (SA, 2001) 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.

#### 5.2.1.1 Methods Validation

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 MONK8A (SA, 2001) code with the JEF2.2 library was validated against these experiments which are provided in the International Handbook of Evaluated Criticality Safety Benchmark Experiments (NEA, 2002) and Nuclear Science and Engineering (NSE, 1962). The experiments chosen are provided in Table 5.2-1, Uranium Solution Experiments Used for Validation, along with a brief description. The overall mean calculated value from the 80 configurations is $1.0017 \pm 0.0005$ (AREVA, 2004) and the results are shown in Figure 5.2-1,Validation Results for Uranium Solutions, plotted against H/U-fissile ratio. If only the 36 low-enriched solutions are considered, the mean calculated value is $1.0007 \pm 0.0005$.

MONK8A is distributed in ready-to-run executable form. This approach provides the user with a level of quality assurance consistent with the needs of safety analysis. The traceability from source code to executable code is maintained by the code vendor. The MONK8A software package contains a set of validation analyses which can be used to support the specific applications. Since the source code is not available to the user, the executable code is identical to that used for the validation analyses. The criticality analyses were performed with MONK8A utilizing the validation provided by the code vendor.

In accordance with the guidance in NUREG-1520 (NRC, 2002), code validation for the specific application has been performed (AREVA, 2004). Specifically, the experiments provided in Table 5.2-1, Uranium Solution Experiments Used for Validation, were calculated and documented as part of the integrated safety analysis for the National Enrichment Facility. The MONK8A computer code and JEF2.2 library are within the scope of the Quality Assurance Program.

### 5.2.1.2 Limits on Control and Controlled Parameters

The validation process established a bias by comparing calculations to measured critical experiments. With the bias determined, an upper safety limit (USL) can be determined using the following equation from NUREG/CR-6698, Guide for Validation of Nuclear Criticality Safety Calculational Methodology (NRC, 2001):

$$USL = 1.0 + Bias - \sigma_{Bias} - \Delta_{SM} - \Delta_{AOA}$$

Where the critical experiments are assumed to have a $k_{eff}$ of unity, and the bias was determined by comparison of calculation to experiment. From Section 5.2.1.1, Methods Validation, the bias is positive and since a positive bias may be non-conservative, the bias is set to zero. The $\sigma_{Bias}$ from Section 5.2.1.1, Methods Validation is 0.0005 and a value of 0.05 is assigned to the subcritical margin, $\Delta_{SM}$. The term $\Delta_{AOA}$ is an additional subcritical margin to account for extensions in the area of applicability. Since the experiments in the benchmark are representative of the application, the term $\Delta_{AOA}$ is set to zero. Thus, the USL becomes:

$$USL = 1 - 0.0005 - 0.05 = 0.9495$$

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

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

In the NCS analysis, $\sigma_{calc}$ is shown to be greater than $\sigma_{Bias}$; therefore, the NEF will be designed using the more conservative equation:

$$k_{eff} = k_{calc} + 3\,\sigma_{calc} < 0.95$$

Additionally, criticality safety in the NEF is ensured by use of geometry, volume, mass and moderation control. Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched $UO_2F_2$ provides the safe values of geometry, volume and mass at 5.0 $^w/_o$ enrichment $UO_2F_2$ to ensure the USL is met. Moreover, Table 5.1-2, Safety Criteria for Buildings/Systems/Components, provides the additional conservatism used in the design of the NEF. All criticality safety analyses use an enrichment of 6.0 $^w/_o$ $^{235}U$, except for Contingency Dump System traps which are analyzed using an enrichment of 1.5 $^w/_o$ $^{235}U$, while the facility is limited to an enrichment of 5.0 $^w/_o$ $^{235}U$.

### 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

The NEF will operate with a 5.0 $^{w}/_{o}$ $^{235}U$ enrichment limit. However, the nuclear criticality safety calculations used an enrichment of 6.0 $^{w}/_{o}$ $^{235}U$. This assumption provides additional conservatism for plant design.

#### 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 $UO_2F_2 \cdot 2H_2O$ 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 $UO_2F_2 \cdot 1.5H_2O$ is formed and, additionally, that the hydrogen fluoride (HF) produced by the $UF_6$/water vapor reaction is also retained in the uranic breakdown to give an overall reaction represented by:

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

For the MONK8A (SA, 2001) calculations, the composition of the breakdown product was simplified to $UO_2F_2 \cdot 3.5H_2O$ 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, often referred to by the trade name "Fomblin." 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 interaction controls placed on movement of vessels containing enriched uranium are specified in the facility procedures. In general, any item in movement (an item being either an individual vessel or a specified batch of vessels) must be maintained at 60 cm (23.6 in) edge separation from any other enriched uranium, and that only one item of each type, e.g., one trap and one pump, may be in movement at one time. These spacing restrictions are relaxed for vessels being removed from fixed positions. In this situation, one vessel may approach an adjacent fixed plant vessel/component without spacing restrictions.

#### 5.2.1.3.5 Pump Free Volume Assumption

There are two types of pumps used in product and dump systems of the plant:

- The vacuum pumps (product and dump) are rotary vane pumps. In the enrichment plant fixed equipment, these are assumed to have a free volume of 14 L (3.7 gal) and are modeled as a cylinder in MONK8A (SA, 2001). This adequately covers all models likely to be purchased.
- The $UF_6$ pumping units are a combination unit of two pumps, one 500 m$^3$/hr (17,656 ft$^3$/hr) pump with a free volume of 8.52 L (2.25 gal) modeled as a cylinder, and a larger 2000 m$^3$/hr (70,626 ft$^3$/hr) pump which is modeled explicitly according to manufacturer's drawings.

### 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, Safe Values for Uniform Aqueous Solution of Enriched $UO_2F_2$, 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 $k_{eff}$ limit used (0.95).
- 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.
- A discussion on the system(s)/process(es) analyzed and the analysis performed, including a description of the accident or abnormal conditions assumed.
- A discussion of the analysis results, including identification of required limits and controls.

During the design phase of NEF, the NCS analysis is performed by a criticality safety engineer and independently reviewed by a second criticality safety engineer. During the operation of NEF, the NCS analysis is performed by criticality safety engineer, independently reviewed by a second criticality safety engineer and approved by the HS&E Manager. Only qualified criticality safety engineers can perform NCS analyses and associated independent review.

### 5.2.1.5 Additional Nuclear Criticality Safety Analyses Commitments

The NEF NCS analyses were performed using the above methodologies and assumptions. NCS analyses also meet the following:

- NCS analyses are performed using acceptable methodologies.
- Methods are validated and used only within demonstrated acceptable ranges.
- The analyses adhere to ANSI/ANS-8.1-1998 (ANSI, 1998a) as it relates to methodologies.
- The validation report statement in Regulatory Guide 3.71 (NRC, 1998) is as follows: LES has demonstrated (1) the adequacy of the margin of safety for subcriticality by assuring that the margin is large compared to the uncertainty in the calculated value of $k_{eff,}$ (2) that the calculation of $k_{eff}$ is based on a set of variables whose values lie in a range for which the methodology used to determine $k_{eff}$ has been validated, and (3) that trends in the bias support the extension of the methodology to areas outside the area or areas of applicability.
- A specific reference to (including the date and revision number) and summary description of either a manual or a documented, reviewed, and approved validation report for each methodology are included. Any change in the reference manual or validation report will be reported to the NRC by letter.
- The reference manual and documented reviewed validation report will be kept at the facility.

- The reference manual and validation report are incorporated into the configuration management program.
- The NCS analyses are performed in accordance with the methods specified and incorporated in the configuration management program.
- The NCS methodologies and technical practices in NUREG-1520 (NRC, 2002), Section 5.4.3.4, are used to analyze NCS accident sequences in operations and processes.
- The acceptance criteria in NUREG-1520 (NRC, 2002), Section 3.4, as they relate to: identification of NCS accident sequences, consequences of NCS accident sequences, likelihood of NCS accident sequences, and descriptions of IROFS for NCS accident sequences are met.
- NCS controls and controlled parameters to assure that under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety are used.
- As stated in ANSI/ANS-8.1-1998 (ANSI, 1998a), process specifications incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded.
- ANSI/ANS-8.7-1998 (ANSI, 1998b), as it relates to the requirements for subcriticality of operations, the margin of subcriticality for safety, and the selection of controls required by 10 CFR 70.61(d) (CFR, 2003b), is used.
- ANSI/ANS-8.10-1983 (ANSI, 1983b), as modified by Regulatory Guide 3.71 (NRC, 1998), as it relates to the determination of consequences of NCS accident sequences, is used.
- If administrative $k_{eff}$ margins for normal and credible abnormal conditions are used, NRC pre-approval of the administrative margins will be sought.
- Subcritical limits for $k_{eff}$ calculations such that: $k_{eff}$ subcritical = 1.0 - bias - margin, where the margin includes adequate allowance for uncertainty in the methodology, data, and bias to assure subcriticality are used.
- Studies to correlate the change in a value of a controlled parameter and its $k_{eff}$ value are performed. The studies include changing the value of one controlled parameter and determining its effect on another controlled parameter and $k_{eff}$.
- The double contingency principle is met. The double contingency principle is used in determining NCS controls and IROFS.
- The acceptance criteria in NUREG-1520 (NRC, 2002) Section 3.4, as they relate to subcriticality of operations and margin of subcriticality for safety, are met.

### 5.2.1.6 Nuclear Criticality Safety Evaluations (NCSE)

For any change (i.e., new design or operation, or modification to the facility or to activities of personnel, e.g., site structures, systems, components, computer programs, processes, operating procedures, management measures), that involves or could affect uranium, a NCSE shall be prepared and approved. Prior to implementing the change, it shall be determined that the entire process will be subcritical (with approved margin for safety) under both normal and credible abnormal conditions. If this condition cannot be shown with the NCSE, either a new or revised NCS analysis will be generated that meets the criteria, or the change will not be made.

The NCSE shall determine and explicitly identify the controlled parameters and associated limits upon which NCS depends, assuring that no single inadvertent departure from a procedure could cause an inadvertent nuclear criticality and that the safety basis of the facility will be maintained during the lifetime of the facility. The evaluation ensures that all potentially affected uranic processes are evaluated to determine the effect of the change on the safety basis of the process, including the effect on bounding process assumptions, on the reliability and availability of NCS controls, and on the NCS of connected processes.

The NCSE process involves a review of the proposed change, discussions with the subject matter experts to determine the processes which need to be considered, development of the controls necessary to meet the double contingency principle, and identification of the assumptions and equipment (e.g., physical controls and/or management measures) needed to ensure criticality safety.

Engineering judgment of the criticality safety engineer is used to ascertain the criticality impact of the proposed change. The basis for this judgment is documented with sufficient detail in the NCSE to allow the independent review by a second criticality safety engineer to confirm the conclusions of the judgment of results. Each NCSE includes, as a minimum, the following information.

- A discussion of the scope of the evaluation, a description of the system(s)/process(es) being evaluated, and identification of the applicable nuclear criticality safety analysis.
- A discussion to demonstrate the applicable nuclear criticality safety analysis is bounding for the condition evaluated.
- A discussion of the impact on the facility criticality safety basis, including effect on bounding process assumptions, on reliability and availability NCS controls, and on the nuclear criticality safety of connected system(s)/process(es).
- A discussion of the evaluation results, including (1) identification of assumptions and equipment needed to ensure nuclear criticality safety is maintained and (2) identification of limits and controls necessary to ensure the double contingency principle is maintained.

The NCSE is performed and documented by a criticality safety engineer. Once the NCSE is completed and the independent review by a criticality safety engineer is performed and documented, the HS&E Manager approves the NCSE. Only criticality safety engineers who have successfully met the requirements specified in the qualification procedure can perform NCSEs and associated independent review.

The above process for NCSEs is in accordance with ANSI/ANS-8.19-1996 (ANSI, 1996).

#### 5.2.1.7 Additional Nuclear Criticality Safety Evaluations Commitments

NCSEs also meet the following:

- The NCSEs are performed in accordance with the procedures specified and incorporated in the configuration management program.
- The NCS methodologies and technical practices in NUREG-1520 (NRC, 2002), Sections 5.4.3.4.1(10)(a), (b), (d) and (e), are used to evaluate NCS accident sequences in operations and processes.
- The acceptance criteria in NUREG-1520 (NRC, 2002), Section 3.4, as they relate to: identification of NCS accident sequences, consequences of NCS accident sequences, likelihood of NCS accident sequences, and descriptions of IROFS for NCS accident sequences are met.
- NCS controls and controlled parameters to assure that under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety are used.
- The double contingency principle is met. The double contingency principle is used in determining NCS controls and IROFS.
- The acceptance criteria in NUREG-1520 (NRC, 2002) Section 3.4, as they relate to subcriticality of operations and margin of subcriticality for safety, are met.

## 5.3 CRITICALITY ACCIDENT ALARM SYSTEM (CAAS)

The facility is provided with a Criticality Accident Alarm System (CAAS) as required by 10 CFR 70.24, (CFR, 2003d). Areas where Special Nuclear Material (SNM) is handled, used, or stored in amounts at or above the 10 CFR 70.24 (CFR, 2003d) mass limits are provided with CAAS coverage. Emergency management measures are covered in the facility Emergency Plan.

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## 5.4 REPORTING

The following are NCS Program commitments related to event reporting:

- A program for evaluating the criticality significance of NCS events will be provided and an apparatus will be in place for making the required notification to the NRC Operations Center. Qualified individuals will make the determination of significance of NCS events. The determination of loss or degradation of IROFS or double contingency principle compliance will be made against the license and 10 CFR 70 Appendix A (CFR, 2003f).
- The reporting criteria of 10 CFR 70 Appendix A and the report content requirements of 10 CFR 70.50 (CFR, 2003g) will be incorporated into the facility emergency procedures.
- The necessary report based on whether the IROFS credited were lost, irrespective of whether the safety limits of the associated parameters were actually exceeded will be issued.
- If it cannot be ascertained within one hour of whether the criteria of 10 CFR 70 Appendix A (CFR, 2003f) Paragraph (a) or (b) apply, the event will be treated as a one-hour reportable event.

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## 5.5 REFERENCES

ANSI, 1998a. Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors, ANSI/ANS-8.1-1998, American National Standards Institute/American Nuclear Society, 1998.

ANSI, 1998b. Guide for Nuclear Criticality Safety in the Storage of Fissile Materials, ANSI/ANS-8.7-1998, American National Standards Institute/American Nuclear Society, 1998.

ANSI, 1983. Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement, ANSI/ANS-8.10-1983, American National Standards Institute/American Nuclear Society, 1983.

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AREVA, 2004. MONK8A Validation and Verification, AREVA, April 16, 2004.

CFR, 2003a. Title 10, Code of Federal Regulations, Part 70, Domestic Licensing of Special Nuclear Material, 2003.

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CFR, 2003c. Title 10, Code of Federal Regulations, Section 70.64, Requirements for new facilities or new processes at existing facilities, 2003.

CFR, 2003d. Title 10, Code of Federal Regulations, Section 70.24, Criticality accident requirements, 2003.

CFR, 2003e. Title 10, Code of Federal Regulations, Section 70.72, Facility changes and change process, 2003.

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CFR, 2003g. Title 10, Code of Federal Regulations, Section 70.50, Reporting requirements, 2003.

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NEA, 2002. International Handbook of Evaluated Criticality Safety Benchmark Experiments, NEA/NSC/DOC(95)03, Nuclear Energy Agency, September 2002 Edition.

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SA, 2001. Serco Assurance, ANSWERS Software Service, "Users Guide for Version 8 ANSWERS/MONK(98) 6," 1987-2001.

# TABLES

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Table 5.1-1 Safe Values for Uniform Aqueous Solutions of Enriched $UO_2F_2$

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| Parameter | Critical Value $k_{eff} = 1.0$ | Safe Value $k_{eff} = 0.95$ | Safety Factor |
|---|---|---|---|
| **Values for 5.0 $^w/_o$ enrichment** | | | |
| Volume | 28.9 L (7.6 gal) | 21.6 L (5.7 gal) | 0.75 |
| Cylinder Diameter | 26.2 cm(10.3 in) | 23.6 cm (9.3 in) | 0.90 |
| Slab Thickness | 12.6 cm (5.0 in) | 10.7 cm (4.2 in) | 0.85 |
| Water Mass | 17.3 kg $H_2O$ (38.1 lb $H_2O$) | 12.7 kg $H_2O$ (28.0 lb $H_2O$) | 0.73 |
| Areal Density | 11.9 g/cm$^2$ (24.4 lb/ft$^2$) | 9.8 g/cm$^2$ (20.1 lb/ft$^2$) | 0.82 |
| Uranium Mass | 37 kg U (81.6 lb U) | | |
| - no double batching | | 26.6 kg U (58.6 lb U) | 0.72 |
| - double batching | | 16.6 kg U (36.6 lb U) | 0.45 |
| **Values for 6.0 $^w/_o$ enrichment** | | | |
| Volume | 24 L (6.3 gal) | 18 L (4.8 gal) | 0.75 |
| Cylinder Diameter | 24.4 cm (9.6 in) | 21.9 cm (8.6 in) | 0.90 |
| Slab Thickness | 11.5 cm (4.5 in) | 9.9 cm (3.9 in) | 0.86 |
| Water Mass | 15.4 kg $H_2O$ (34.0 lb $H_2O$) | 11.5 kg $H_2O$ (25.4 lb $H_2O$) | 0.75 |
| Areal Density | 9.5 g/cm$^2$ (19.5 lb/ft$^2$) | 7.5 g/cm$^2$ (15.4 lb/ft$^2$) | 0.79 |
| Uranium Mass | 27 kg U (59.5 lb U) | | |
| - no double batching | | 19.5 kg U (43.0 lb U) | 0.72 |
| - double batching | | 12.2 kg U (26.9 lb U) | 0.45 |

Table 5.1-2 Safety Criteria for Buildings/Systems/Components



| Building/System/Component | Control Mechanism | Safety Criteria |
|---|---|---|
| Enrichment | Enrichment | 5.0 $^{w}/_{o}$ (6 $^{w}/_{o}$ $^{235}U$ used in NCS) |
| Centrifuges | Diameter | < 21.9 cm (8.6 in) |
| Product Cylinders (30B) | Moderation | H < 0.95 kg (2.09 lb) |
| Product Cylinders (48Y) | Moderation | H < 1.05 kg (2.31 lb) |
| $UF_6$ Piping | Diameter | < 21.9 cm (8.6 in) |
| Chemical Traps | Diameter | < 21.9 cm (8.6 in) |
| Product Cold Trap | Diameter | < 21.9 cm (8.6 in) |
| Contingency Dump System Traps | Enrichment | 1.5 $^{w}/_{o}$ $^{235}U$ |
| Tanks | Mass | < 12.2 kg U (26.9 lb U) |
| Feed Cylinders | Enrichment | < 0.72 $^{w}/_{o}$ $^{235}U$ |
| Uranium Byproduct Cylinders | Enrichment | < 0.72 $^{w}/_{o}$ $^{235}U$ |
| $UF_6$ Pumps (first stage) | N/A | Safe by explicit calculation |
| $UF_6$ Pumps (second stage) | Volume | < 18.0 L (4.8 gal) |
| Individual Uranic Liquid Containers, e.g., Fomblin Oil Bottle, Laboratory Flask, Mop Bucket | Volume | < 18.0 L (4.8 gal) |
| Vacuum Cleaners Oil Containers | Volume | <18.0 L (4.8 gal) |

Table 5.2-1 Uranium Solution Experiments Used for Validation

Page 1 of 1

| MONK8A Case | Case Description | Number of Experiments | Handbook Reference |
|---|---|---|---|
| 13 | High-enriched uranyl nitrate solutions at various H:U ratios (93.17 $^{w}/_{o}$ $^{235}U$) | 12 | HEU-SOL-THERM-002<br>HEU-SOL-THERM-003 |
| 23 | Uranyl nitrate solution (~ 95 $^{w}/_{o}$ enriched) | 5 | HEU-SOL-THERM-013<br>NS&E |
| 35 | High-enriched uranyl nitrate solutions (U concentration from 20-700 g/L) | 11 | HEU-SOL-THERM-009 - HEU-SOL-THERM-012 |
| 43 | Low-enriched uranyl nitrate solutions | 3 | LEU-SOL-THERM-002 |
| 51 | Low-enriched uranium solutions (new STACY experiments) | 7 | LEU-SOL-THERM-004 |
| 63 | Boron carbide absorber rods in uranyl nitrate (5.6 $^{w}/_{o}$ enriched) | 3 | LEU-SOL-THERM-005 |
| 67 | Highly enriched uranyl nitrate solution with a concentration range between 59.65 and 334.66 g U/L | 10 | HEU-SOL-THERM-001 |
| 68 | Highly enriched uranyl fluoride/heavy water solution with a concentration range between 60 and 679 g U/L and a heavy water reflector | 6 | HEU-SOL-THERM-004 |
| 71 | STACY: 28 cm thick slabs of 10 $^{w}/_{o}$ enriched uranyl nitrate solutions, water reflected | 7 | LEU-SOL-THERM-016 |
| 80 | STACY: Unreflected 10 $^{w}/_{o}$ enriched uranyl nitrate solution in a 60 cm diameter cylindrical tank | 5 | LEU-SOL-THERM-007 |
| 81 | STACY: Concrete reflected 10 $^{w}/_{o}$ enriched uranyl nitrate solution reflected by concrete | 4 | LEU-SOL-THERM-008 |
| 84 | STACY: Borated concrete reflected 10 $^{w}/_{o}$ enriched uranyl nitrate solution in a 60 cm diameter cylindrical tank | 3 | LEU-SOL-THERM-009 |
| 85 | STACY: Polyethylene reflected 10 $^{w}/_{o}$ enriched uranyl nitrate solution in a 60 cm diameter cylindrical tank | 4 | LEU-SOL-THERM-010 |

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# FIGURES

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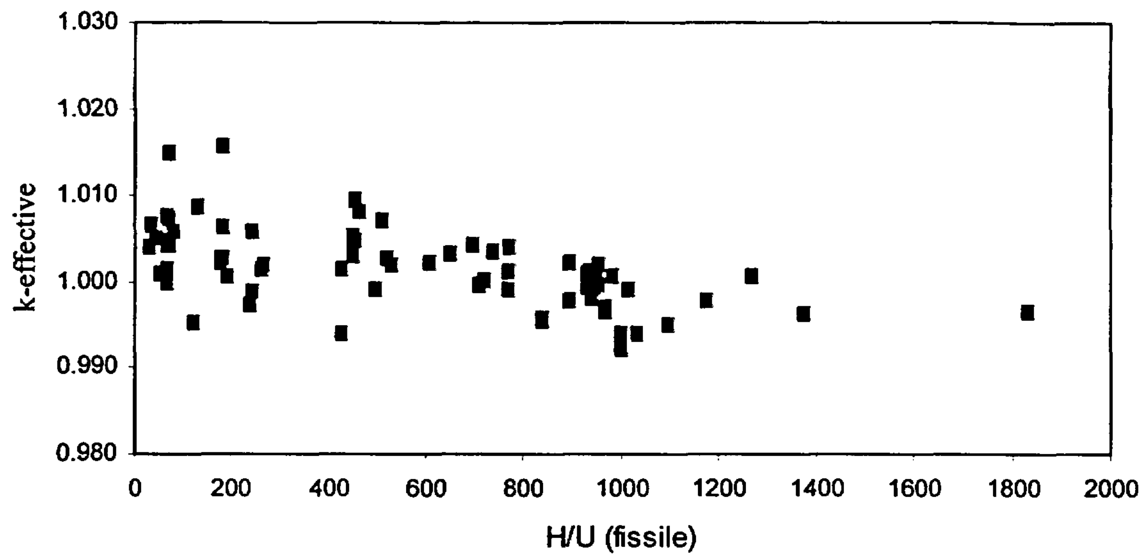


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Figure 5.2-1.doc

**FIGURE 5.2-1**

VALIDATION RESULTS FOR URANIUM SOLUTIONS

REVISION DATE: DECEMBER 2003