TABLE OF CONTENTS

Page

LIST OF TABLES

LIST OF FIGURES

5.0 NUCLEAR CRITICALITY SAFETY

The Nuclear Criticality Safety Program for the Eagle Rock Enrichment Facility (EREF) is in accordance with U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 3.71, Nuclear Criticality Safety Standards for Fuel and Material Facilities (NRC, 2005). Regulatory Guide 3.71 (NRC, 2005) provides guidance on complying with the applicable portions of NRC regulations, including 10 CFR 70 (CFR, 2008c), 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 NEF SAR references Revision 0 of Regulatory Guide 3.71 (NRC, 1998). Revision 1 does not change any of the guidance provided in the initial issuance of Regulatory Guide 3.71; rather, it provides guidance concerning changes that have occurred since the NRC published the original guide in 1998. AREVA Enrichment Services, LLC (AES) is committed to following the guidelines in this regulatory guide for specific ANSI/ANS criticality safety standards. 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) 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 5.4.3.4.4 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)

5.1 THE NUCLEAR CRITICALITY SAFETY (NCS) PROGRAM

The AREVA EREF, located in Bonneville County, Idaho, will be designed, constructed, and operated such that a nuclear criticality event is prevented, and to meet the regulatory requirements of 10 CFR 70 (CFR, 2008c). 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 EREF Integrated Safety Analysis (ISA) 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 EREF meets the double contingency principle. To meet the double contingency principle, the EREF 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.

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 upset events (both to the dump traps and to tails cylinders) which are analyzed assuming that the dumped material has a ^{235}U enrichment of 1.5 $^{w}/_{o}$. Both the 6.0 $^{w}/_{o}$ value and the 1.5 $^{w}/_{o}$ value include appropriate margins to safety. In accordance with 10 CFR 70.61(d) (CFR, 2008a), 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. All NCS controls are preventative in nature; there are no mitigative NCS controls. Interaction controls provide for safe movement and storage of components. Plant and equipment features assure prevention of excessive enrichment. The plant is divided into eight 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 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, 2008b), by observing the double contingency principle throughout the plant, a criticality accident is reduced. 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, 2008c) will be developed, implemented, and maintained.
- 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, 2008b) 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.

Nuclear Criticality Safety 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-8.20-1991, 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.
- Required response to the activation of the Criticality Accident Alarm Signals (CAAS).
- Required response to NCS nonconformance.

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

5.1.2 Control Methods for Prevention of Criticality

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 nuclear criticality safety analyses (NCSAs). NCSAs and Nuclear Criticality Safety Evaluations (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. Where possible, passive engineered controls are used to ensure NCS. The determination of the safe values of the major controlling parameters used to control criticality in the facility is described below.

Moderation control is in accordance with ANSI/ANS-8.22-1997, Nuclear Criticality Safety Based on Limiting and Controlling Moderators (ANSI, 1997). However, for the purposes of the NCSA, 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 MONK8A (SA, 2001) and the JEF2.2 library. Criticality analyses were performed 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 $k_{eff} = 0.95$. Table 5.1-1, Safe Values for Uniform Aqueous Solution of Enriched UO_2F_2, shows both the critical and safe limits for 5.0 $^{w}/_{o}$ and 6.0 $^{w}/_{o}$. The values in Table 5.1-1 are changed from the NEF because the MONK8A (SA, 2001) criticality analyses were performed with a revised correlation for the density of aqueous solutions of UO_2F_2.

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 the EREF will be limited to 5.0 $^{w}/_{o}$ enrichment, as additional conservatism, the values in 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 upset events (both the dump traps and to tails cylinders) which are limited to 1.5 $^{w}/_{o}$ ^{235}U for the dumped material.

The values in Table 5.1-1 are calculated optimum moderation (i.e., various H/U ratios greater than and less than 7 are analyzed) and 30 cm water reflection. The conservative modeling techniques provide for more conservative values than provided in ANSI/ANS-8.1 (ANSI, 1998a). 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, other than the Type 30B and 48Y cylinders and the first stage UF_6 pumps are safe by geometry. 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 Technical Support Building (TSB) criticality safety for uranium loaded liquids is ensured by limiting the mass of uranium in any single tank to less than or equal to 12.2 kg U (26.9 lb U). Individual liquid storage bottles are safe by volume. Interaction in storage arrays is accounted for.

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

<u>Enrichment</u>

Enrichment is controlled to limit the percent ^{235}U within any process, vessel, or container, except the contingency dump system, to a maximum enrichment of 5 $^{w}/_{o}$. The design of the contingency dump system controls enrichment to a limit of 1.5 $^{w}/_{o}$ ^{235}U. Although EREF is limited to a maximum enrichment of 5 $^{w}/_{o}$, as added conservatism nuclear criticality safety is analyzed using an enrichment of 6 $^{w}/_{o}$ ^{235}U.

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.

The safe 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 6 $^{w}/_{o}$ ^{235}U for safety margin.

Moderation

Water and oil are the moderators considered in EREF. At EREF the only system where moderation is used as a control parameter is in the product cylinders. Moderation control is established consistent with the guidelines of ANSI/ANS-8.22-1997 (ANSI, 1997) 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.
- 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 a 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.

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. Analysis or sampling is employed to verify the mass of the material. Conservative administrative limits for each operation are specified in the operating procedures.

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.

Reflection

Reflection is considered when performing NCSAs and NCSEs. The possibility of full water reflection is considered but the layout of the EREF is a very open design and it is highly unlikely that those vessels and plant components requiring criticality control could become flooded from a source of water within the plant. However, some select analyses have been performed using full reflection for conservatism. Partial reflection of 2.5 cm (0.984 in) of water is assumed where limited moderating materials (including humans) may be present. It is recognized that concrete can be a more efficient reflector than water; therefore, it is modeled in analyses where it is present. When moderation control is identified in the ISA Summary, it is established consistent with the guidelines of ANSI/ANS-8.22-1997 (ANSI, 1997).

The NEF SAR stated neither automatic sprinkler nor standpipe and hose systems are provided in the TSB, Separation Buildings, Blending and Liquid Sampling, CRDB, CAB, and Centrifuge Post Mortem areas. Automatic and manual water-based fire protection systems (fire sprinklers and standpipes) are required by the building code in EREF process buildings due to occupancy classification and/or lack of exterior openings. As these include areas containing enriched uranium, the risk of fire versus the risk of criticality has been considered in developing a methodology for evaluating the application or omission of water-based fire suppression. This methodology will be a precursor to final NCSAs and considers the following parameters: presence and quantity of fissile materials and configuration; in-situ combustibles and configuration; probability/presence of transient combustibles; probability/presence of ignition sources; impact of reflection from external water spray; potential to flood fissile components/containers; potential for water to enter non-moderator controlled vacuum systems; potential to displace safe by geometry shapes, vessels, arrays; safeguards/barriers to moderator introduction and impact of natural phenomena hazards and impact on barriers (e.g., seismic, high wind, snow/rain/flood loading, etc.).

The impact of interstitial aerial water density on single components, interacting components, or arrays will be included in criticality safety assessments at detailed design when component locations are known. Literature indicates that maximum aerial density of water from fire sprinklers would not be expected to exceed 2% (0.02 g/cc). Fire sprinkler discharge could also result in sheeting of water on surfaces. This has been shown to not exceed a depth of more than 4 mm (0.16 in) on cylindrical surfaces (DOE, 1997) (DOE, 1994).

To avoid the risk of a criticality due to water ingress, fire sprinkler coverage will not be provided where sprinklers could discharge water on or near sub-atmospheric process systems containing enriched UF_6 above a critical mass and requiring moderator control (i.e., not safe-by-design). Similarly, fire sprinklers will not be provided over areas where discharge could result in accumulation of a critical mass in an unsafe geometry (i.e., non-safe floors, drains, or collection basins). Fire risk in areas where sprinklers are omitted will be controlled through other measures (i.e., limit or exclusion of combustible material, alternate suppression systems, etc.).

Where fire sprinkler systems are installed in areas containing uranium, they will be of the pre-action type to ensure that inadvertent discharge of water does not occur. Pre-action sprinkler systems include closed head sprinklers used in conjunction with a control valve. This requires two independent operations – detection of fire by a separate fire alarm system which opens a sprinkler control valve to allow water into the piping network and actuation of individual sprinkler(s) in response to high temperature – before water will be discharged. Piping integrity is monitored and alarmed during non-fire conditions. Pre-action systems will also be suitably designed to ensure that natural phenomena hazards (NPH) do not result in discharge under non-fire conditions making the probability of inadvertent water discharge both double contingent and non-credible.

The final areas of fire sprinkler system coverage as determined by the decision methodology will be developed jointly between fire safety and criticality safety staff and integrated into NCSA's developed at detailed design. Review and approval of these NCSAs will be in accordance with the Criticality Safety Program and will ensure double contingency and k effective + 3σ < 0.95 is maintained under normal and abnormal conditions.

Because of the size of the facility, fire standpipe systems are also provided in select process areas to facilitate fire response. Standpipes will be routed in a manner and suitably designed against NPH criteria to ensure their failure will not result in flooding of areas containing enriched uranium above a critical mass.

Fire response to all process areas of the facility (whether by the on-site fire brigade or off-site fire department) requires that one member of the response team be assigned as the criticality safety officer. This individual is responsible to ensure that criticality safety is not compromised for any and all firefighting activities including the deployment of any fire hose streams in areas requiring reflection or moderator control.

Chapter 7, Fire Safety, contains additional discussion on fire system locations and application.

Interaction

NCSAs and 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 using the Monte Carlo computer code MONK8A to ensure k_{eff} ($k_{calc} + 3\sigma_{calc}$) < 0.95.

Concentration, Density and Neutron Absorbers

EREF does not use mass concentration, density, or neutron absorbers 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.

Significant accumulations of enriched UF_6 reside only in the Product Low Temperature Take-off Stations, Product Liquid Sampling Autoclaves, Product Blending System, or the UF_6 cold traps. All these, except the UF_6 cold traps, contain the UF_6 in 30B and 48Y cylinders. All these significant accumulations are within enclosures protecting them from water ingress. The facility design has minimized the possibility of accidental moderation by eliminating direct water contact with these cylinders of accumulated UF_6. In addition, the facility's stringent procedural controls for enriching the UF_6 assure that it does not become unacceptably hydrogen moderated while in process. The plant's UF_6 systems operating procedures contain safeguards against loss of moderation control (ANSI, 1997). No neutron poisons are relied upon to assure criticality safety.

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 EREF 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 NCS organization is responsible for implementing the Nuclear Criticality Safety Program. During the design phase, the NCS function is performed within the design engineering organization. The NCS function for operations is described below.

The NCS Manager reports to the Environmental, Health, Safety, and Licensing (EHS&L) Manager as described in Chapter 2, Organization and Administration. The EHS&L Manager is accountable for overall nuclear 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 NCS Manager 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 NCS 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 NCS 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 NCS Program using audits and assessments
- Provide NCS postings that identify administrative controls for operators in applicable work areas.

The minimum qualifications for the NCS Manager and NCS Engineers are described in Section 2.2.4. The EHS&L Manager has the authority and responsibility to assign and direct activities for the NCS Program. The NCS Manager is responsible for implementation of the NCS program. The NCS function will be staffed with suitably trained personnel and provided sufficient resources for operation.

EREF management implements 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 discontinue operation, bring equipment to a safe condition, 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.

5.1.6 Passive IROFS that Contain Safe-by-Design Component Attributes

The passive IROFS that contain safe-by-design component attributes listed in Appendices B and C of the ISAS are designated as IROFS that are safe by their respective geometry attribute (diameter, volume or slab thickness) or as IROFS that are safe by criticality analysis.

The passive, safe-by-design feature of these components do not rely on human interface to perform the criticality safety function. The only potential means to effect a change to the safe-by-design feature that might result in a failure to function, would be to implement a design change. Credible process deviations or events do not adversely impact the performance of the safety function. The evaluation of the potential to adversely impact the safety function of the passive design feature (which includes consideration of potential mechanisms to cause bulging, corrosion, and breach of confinement/leakage and subsequent accumulation of material) is presented in Appendix B and Appendix C of the ISAS, and includes consideration of adequate controls to ensure the double contingency principle is met. For the identified components, Appendix B and Appendix C summarize the rationale for the conclusion that there is no credible means to effect a change to the safe-by-design feature that might result in a failure of the safety function. Appendix B and Appendix C also support the conclusion that significant margin exists.

For components whose passive design feature is the safe-by-design attribute of diameter, volume or slab thickness, significant margin is defined as a margin of at least 10%, during both normal and upset conditions. Margin is calculated between the actual design parameter value of the component and the value of the corresponding 6.0 $^{w}/_{o}$ enrichment critical value of diameter per Table 5.1-1 of the SAR.

For components whose passive design feature is the safe-by-design attribute of physical arrangement, significant margin is defined as a k_{eff} <0.95, where $k_{eff} = k_{calc} + 3\sigma_{calc}$.

The frequency for the criticality initiating event was determined to be “highly unlikely” based on (1) the lack of credible means to effect an adverse change to the safe-by-design feature of these passive design components, (2) the significant margins of safety that exist, (3) the application of the 10 CFR 70.72 Configuration Management System to preserve the safety design features, and (4) the relative low risk of a criticality event at low enriched uranium enrichment facilities. Therefore, an initiating event index of (-5) is appropriate.

QA Level 1 requirements apply to this feature. In addition, the Configuration Management System required by 10 CFR 70.72 (implemented by the Eagle Rock Enrichment Facility Configuration Management Program) ensures the maintenance of the safety function of this feature and assures compliance with the double contingency principle, as well as the defense-in-depth criterion of 10 CFR 70.64(b).

5.2 METHODOLOGIES AND TECHNICAL PRACTICES

This section describes the methodologies and technical practices used to perform the NCS analyses (NCSAs) and NCS evaluations (NCSEs). 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 Monte Carlo tool for NCS 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), NUREG/CR-1071 (NRC, 1980). 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 93 configurations is 1.0017 ± 0.0034 and the results are provided in the MONK8A Validation and Verification report (AREVA, 2008).

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.

In accordance with the guidance in NUREG-1520 (NRC, 2002), code validation for the specific application has been performed (AREVA, 2008). Specifically, the experiments provided in Table 5.2-1, Uranium Experiments Used for Validation, were calculated and documented in the MONK8A Validation and Verification report (AREVA, 2008) for the Eagle Rock Enrichment

Facility. In addition, the MONK8A Validation and Verification report (AREVA, 2008) satisfies the commitment to ANSI/ANS-8.1-1998 (ANSI, 1998a) and includes details of computer codes used, operations, recipes for choosing code options (where applicable), cross sections sets, and any numerical parameters necessary to describe the input.

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 using the Single Sided Lower Tolerance Limit equations from the NUREG/CR-6698, Guide for Validation of Nuclear Criticality Safety Calculational Methodology (NRC, 2001):

$$USL = K_L - \Delta_{SM} - \Delta_{AOA}$$

where

$$K_L = \bar{k}_{eff} - U(S_P)$$

Where

$\bar{k}_{eff}$ is the weighted average K_{eff} of the analyzed benchmark experiments and by analysis is 1.0010. Because of the positive bias, Bias = $\bar{k}_{eff}$ - 1 = 0.0010, $\bar{k}_{eff}$ is conservatively adjusted to unity (1.0),

U is the one-sided lower tolerance factor and is determined from the analysis to be 2.065,

S_P is the square root of the pooled variance and is determined from the analysis to be 0.0044,

Δ_{SM} is the margin of subcriticality and is set to 0.05,

Δ_{AOA} is an additional margin of subcriticality that may be necessary as a result of extensions to the area of applicability (AOA) and is determined from the analysis to be 0.0014. If extensions are not made to the AOA, Δ_{AOA} is zero.

Where the critical experiments are assumed to have a k_{eff} of unity, the bias is the difference of the calculated k_{eff} and the experimental k_{eff} (i.e., Bias = calculated k_{eff} – experimental k_{eff}). From Section 5.2.1.1, Methods Validation, the bias (0.0010) is positive and since a positive bias may be non-conservative, the bias is set to zero and is unity (1.0). The term Δ_{AOA} is an additional subcritical margin to account for extensions in the AOA. Since the experiments in the benchmark are representative of the application, the term Δ_{AOA} is set to zero for systems and components not associated with the Contingency Dump System. For the Contingency Dump System, it was necessary to extrapolate the AOA to include 1.5% enrichment and the Δ_{AOA} term is set to 0.0014 to account for this extrapolation. Thus, the USL becomes:

- USL = 0.9908 - 0.05 - 0.0 = 0.9408 (for systems and components NOT associated with the Contingency Dump System) (AREVA, 2008)
- USL = 0.9908 - 0.05 - 0.0014 = 0.9394 (for the Contingency Dump System) (AREVA, 2008)

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

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

The risk of an accidental criticality resulting from EREF operations is inherently low. The low risk warrants the use of an alternate approach.

At the low enrichment limits established for the EREF, sufficient mass of enriched uranic material cannot be accumulated to achieve criticality without moderation. Uranium in the centrifuge plant is inherently a very dry, unmoderated material. Centrifuge separation operations at EREF do not include solutions of enriched uranium. For most components that form part of the centrifuge plant or are connected to it, sufficient mass of moderated uranium can only accumulate by reaction between UF_6 and moisture in air leaking into plant process systems, leading to the accumulation of uranic breakdown material. Due to the high vacuum requirements for the normal operation of the facility, air in-leakage into the process systems is controlled to very low levels and thus the highly moderated condition assumed represents an abnormal condition. In addition, excessive air in-leakage would result in a loss of vacuum, which in turn would cause the affected centrifuges to crash (self destruct) and the enrichment process in the affected centrifuges to stop. As such, buildup of additional mass of moderated uranic breakdown material, such that component becomes filled with sufficient mass of enriched uranic material for criticality, is precluded. Even when accumulated in large UF_6 cylinders or cold traps, neither UF_6 nor UO_2F_2 can achieve criticality without moderation at the low enrichment limit established for the EREF.

Therefore, due to the low risk of accidental criticality associated with EREF operations and the margin that exists in the design and operation of the EREF with respect to nuclear criticality safety, a margin of sub-criticality for safety of 0.05 (i.e., $k_{eff} = k_{calc} + 3\sigma_{calc} < 0.95$) is adequate to ensure sub-criticality is maintained under normal and abnormal credible conditions. As such, the EREF will be designed using the equation:

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

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 EREF 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. However, some select analyses have full water reflection for conservatism. Otherwise, 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 EREF 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 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 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 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. 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 $UO_2F_2 \bullet 1.5\ H_2O$ and $UO_2F_2\ 2\ H_2O$ 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 $UO_2F_2 \bullet 1.5\ H_2O$ 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:

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

For the NCS 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. Mixtures of UF_6 and PFPE oil would be as conservative a case as the 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 the minimum required 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. Spacing requirements will be determined on a system by system basis, including component insertion or extraction from an array.

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 2,000 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 EREF 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.

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 EREF, the NCS analysis is performed by an NCS engineer and independently reviewed by a second NCS engineer. During the operation of EREF, the NCS analysis is performed by NCS engineer, independently reviewed by a second NCS engineer, and approved by the NCS Manager. Only qualified NCS engineers can perform NCS analyses and associated independent review.

5.2.1.5 Additional Nuclear Criticality Safety Analyses Commitments

The EREF NCS analyses were performed using the methodologies and assumptions in Section 5.2.1.3 and Section 5.2.1.4.

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, 2005) is as follows: EREF 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, except for Item 4 in Section 5.4.3.4.1 and Items 9, 13 and 15 in Section 5.4.3.4.2.
- 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, 2008a), is used.
- ANSI/ANS-8.10-1983 (ANSI, 1983), as modified by Regulatory Guide 3.71 (NRC, 2005), 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.

Engineering judgment of the NCS 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 NCS 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 an NCS engineer. Once the NCSE is completed and the independent review by a NCS engineer is performed and documented, the NCS Manager approves the NCSE. Only NCS 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, 2008d). 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 shall be 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 provided with emergency power and is designed to remain operational during credible events or conditions, including fire, explosion, corrosive atmosphere, or seismic shock (equivalent to the site-specific design-basis earthquake or the equivalent value specified by the uniform building code).

Whenever the CAAS is not functional, compensatory measures, such as limiting access and restricting SNM movement, will be implemented. Should the CAAS coverage be lost and not restored within a specified number of hours or an equivalent level of protection has not been provided (e.g., portable CAAS system), the operations will be rendered safe (by shutdown and quarantine) if necessary. 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.

5.4 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 the 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, 2008e).
- The reporting criteria of 10 CFR 70 Appendix A and the report content requirements of 10 CFR 70.50 (CFR, 2008f) 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, 2008e) Paragraph (a) or (b) apply, the event will be treated as a one-hour reportable event.

5.5 REFERENCES

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TABLES

Table 5.1-1 Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2
(Page 1 of 1)

Parameter	Critical Value k_{eff} = 1	Safe Value k_{eff} = 0.95	Safety Factor
Values for 5.0 $^w/_o$ enrichment			
Volume	30.3 L (8.0 gal)	22.9 L (6.0 gal)	0.76
Cylinder Diameter	26.6 cm (10.5 in)	23.9 cm (9.4 in)	0.90
Slab Thickness	12.8 cm (5.0 in)	11.1 cm (4.4 in)	0.87
Water Mass	18.5 kg (40.8 lb)	14.2 kg (31.3 lb)	0.77
Areal Density	11.8 g/cm^2 (24.2 lb/ft^2)	9.9 g/cm^2 (20.3 lb/ft^2)	0.84
Uranium Mass	36.7 kg U (80.9 lb U)	26.8 kg U (59.0 lb U)	0.73
-no double batching		26.4 kg U (58.2 lb U)	0.72
-double batching		16.5 kg U (36.4 lb U)	0.45
Values for 6.0 $^w/_o$ enrichment			
Volume	25.3 L (6.7 gal)	19.3 L (5.1 gal)	0.76
Cylinder Diameter	24.8 cm (9.8 in0	22.4 cm (8.8 in)	0.90
Slab Thickness	11.6 cm (4.6 in)	10.1 cm (4.0 in)	0.87
Water Mass	15.4 kg H_2O (34.0 lb H_2O)	11.9 kg H2O (26.2 lb H2O)	0.77
Areal Density	9.4 g/cm^2 (19.3 lb/ft^2)	7.9 g/cm^2 (16.2 lb/ft^2)	0.84
Uranium Mass	27 kg (59.5 lb U)	20.1 kg (44.3 lb U)	0.74
-no double batching	Not Applicable	19.4 kg U (43.0 lb U)	0.72
-double batching	Not Applicable	12.2 kg U (26.9 lb U)	0.45

Table 5.1-2 Safety Criteria for Buildings / Systems / Components
(Page 1 of 1)

Buildings/System/Components	Control Mechanism	Safety Criteria
Enrichment	Enrichment	5.0 $^{w}/_{o}$ (6.0 $^{w}/_{o}$ ^{235}U used in NCSAs)
Centrifuges	Diameter	< 22.4 cm (8.8 in)
Product Cylinders (30B)	Moderation	H < 0.92 kg (2.03 lb) – Note 1
Product Cylinders (48Y)	Moderation	H < 1.04 kg (2.29 lb) – Note 1
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 Traps	Enrichment	1.5 $^{w}/_{o}$ ^{235}U
Tanks	Mass	< 12.2 kg U (26.9 lb U) – Note 2
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	< 19.3 L (5.1 gal)
Individual Uranic Liquid Containers, e.g., Fomblin Oil Bottle, Laboratory Flask, Mop Bucket	Volume	< 19.3 L (5.1 gal)
Vacuum Cleaners Oil Containers	Volume	< 19.3 L (5.1 gal)

Notes:

1. Assumes outside storage (e.g., exposed to snow, ice, or rain).
2. Determined for double batch safe mass.

Table 5.2-1 Uranium Experiments Used for Validation
(Page 1 of 1)

MONK 8A Case Set	Case Description	Number of Experiments	Handbook Reference (NEA, 2002)
25	Low-enriched damp U_3O_8 powder in cubic aluminum cans	10	NUREG/CR-1071 (NRC, 1980)
42	MARACAS Program: Polythene reflected critical configurations with low enriched and low moderated uranium dioxide powder $U(5)O_2$	18	LEU-COMP-THERM-049
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
69	Critical arrays of polyethylene-moderated $U(30)F_4$-Polytetrafluoroethylene one-inch cubes	29	IEU-COMP-THERM-001
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-0l0