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Attn: Document Control Desk
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Louisiana Energy Services, LLC
NRC Docket No. 70-3103

Subject: License Amendment Request to Revise Chapter 5 of Safety Analysis Report (LAR-10-07)

Reference:
1. Telecommunication between the NRC and URENCO USA on SAR Chapter 5 Revisions, June 1, 2010
2. LES Configuration Change No: CC-LS-2009-0006, Rev. 0, SAR Enhancements for First Cascade Online
3. LES Configuration Change No: CC-LS-2010-0012, Rev. 1, Changes to Clarify NCS Methodology
4. LES Configuration Change No: CC-EG-2010-0021, Rev. 0, Criticality Assessments
5. LES Configuration Change No: CC-EG-2007-0314, Rev. 0, ETC Calculation ETC4053126, Issue 1, Risk of criticality following a seismic event affecting the NEF Cascade System
6. Letter from Deborah A. Seymour (NRC) to Gregory Smith (LES), NRC Inspection Report No. 70-3103/2010-006 and Notice of Violation, dated May 3, 2010.
7. Letter from LES to NRC, LES-10-00095-NRC, Reply to Notice of Violation 70-3103/2010-006-Part 2, dated May 12, 2010.
8. Letter from LES to NRC, LES -10-00104-NRC, Reply to Notice of Violation 70-3103/2010-006-Part 2- Supplement, dated May 23, 2010.
9. Letter from LES to NRC, LES-10-00108-NRC Reply to Notice of Violation 70-3103/2010-006-Part 2- Second Supplement, dated May 27, 2010

Pursuant to the mutual agreement reached during the Ref. 1 telecommunication with the NRC Staff, URENCO USA (UUSA) herewith submits the subject license amendment request (LAR-10-07) to revise certain sections of Chapter 5 (*Nuclear Criticality Safety*) of the Safety Analysis Report (SAR). The proposed changes to the SAR originated from the Refs. 2-5 Configuration Change Packages (copies of which were forwarded to the NRC on June 1-2, 2010); are described in Enclosure 1; and are illustrated with SAR page mark-ups in Enclosure 2.

The changes proposed herein to Chapter 5 of the SAR were initially made under UUSA's approval authority, in what UUSA perceived to be allowed in accordance with 10 CFR 70.72(c). However, following UUSA's Ref. 7 response to the Ref. 6 Notice of Violation (NOV); the ensuing dialogue with the NRC that led to UUSA's evolving commitments in Refs. 8 and 9 (regarding future SAR changes); and the subsequent NRC feedback during the Ref. 1 telecommunication, it was deemed expedient to request formal NRC approval of the SAR Chapter 5 changes proposed herein. Furthermore, due

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UUSA appreciates the efforts of the NRC staff in supporting the review of this license amendment request; and looks forward to the NRC's timely approval of same. Should the NRC Staff have any questions regarding this submittal, please have them contact Gary Sanford, LES Director of Quality and Regulatory Affairs, at 575.394.5407.

Respectfully,

David E. Sexton
Chief Nuclear Officer and Vice President of Operations

Enclosures: 1) Description of Proposed Changes
2) Mark-ups to Safety Analysis Report (SAR) Pages

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ENCLOSURE 1

License Amendment Request (LAR-10-07) - Description of Proposed Changes, Background, Proposed Changes, Technical Analysis of Proposed Changes, and Safety Significance

1 Introduction

1.1 Purpose

The purpose of this license amendment request (LAR-10-07) is to modify certain sections of Chapter 5 (*Nuclear Criticality Safety*) of the Safety Analysis Report (SAR) to: correct an error in Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2, expand the analyses that were performed at 1.5 $^{w}/_{o}$ to include the entire Dump System, clarify statements concerning the use of neutron absorbers, revise restrictions on vessel movement, and other changes that reduce the margin of subcriticality.

1.2 Background

As noted by the NRC in Violation B of its Notice of Violation (NOV) (Ref.6), 10 CFR 70.72(c) states, in part, that the licensee may make changes to the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel, without prior Commission approval, unless the change as stated in 10 CFR 70.72(c)(4), is otherwise prohibited by this section, license condition, or order; and that:

> 10 CFR 70.61(d) states, in part, that the risk of nuclear criticality accidents must be limited by assuring that, under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety.

The NRC further noted in Violation B of its NOV (Ref. 6) that it had approved the margin of subcriticality for safety, as documented in the licensee's (UUSA's) SAR, Revision 6, with the issuance of SNM-2010; but that as of April 1, 2010, the licensee (UUSA) had made changes to the approved margin of subcriticality for safety without prior NRC approval when implementing the following changes to the SAR:

1. SAR Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2, was changed to increase dimensions of process components after the licensee identified an error when calculating the safe values in the table.

2. SAR Section 5.1.1, Management of the Nuclear Criticality Safety Program, stated in part, that 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 $^{w}/_{o}$. The licensee revised this section to expand the analyses that were performed at 1.5 $^{w}/_{o}$ to include the

entire Dump System. The Dump System includes the Tails Take-Off System and the Contingency Dump System.

3. SAR Section 5.1.2, Control Methods for Prevention of Criticality, stated that NEF does not use neutron absorbers as a criticality control parameter. The licensee made a change to the SAR to take credit for neutron absorbers in standard materials used in construction and processes.

4. SAR Section 5.2.1.3.4, Vessel Movement Assumption, stated in part that any item in movement must be maintained at 60 centimeters (23.6 inch) edge separation from any other enriched uranium and only one item of each type of vessel may be in movement at one time. This section was changed to state that limits were placed on movement of vessels by procedures or work plans that varied by the type of vessel. For some vessels, the separation distance was reduced from 60 centimeters.

UUSA contested the above Violation B (related to nuclear criticality issues) in its Ref. 7 response. In addition, following subsequent discourse with the NRC, UUSA submitted its Ref. 8 response which included a formal commitment not to make further changes to the appropriate portions of the SAR. Further dialogue with the NRC Staff resulted in UUSA's Ref. 9 revision of the formal commitment as follows:

> URENCO USA commits to not make any changes, without prior NRC approval, to the specific sections of the SAR Chapters 3 and 5 that would result in modifying the current values for criticality-based analysis in a less conservative direction until such time that Violations A and B from Inspection Report 70-3103/2010-006 are resolved. Specific Chapter 3 Sections include 3.2.5.2 related to Safe-By Design and Table 3.1-9 Failure Frequency Index Numbers. Chapter 5 Sections include 5.0, 5.1.1 through 5.1.5, 5.2.1.2 through 5.2.1.7, Tables 5.1-1 and 5.1-2. The above sections contain data and discussions related to Safe-By-Design, Nuclear Criticality Safety Analysis, Nuclear Criticality Safety parameters, Commitments, and the margin of safety for subcriticality. Upon disposition of the disputed violation this commitment shall expire, or be revised.

Following the NRC's review and comment (Ref. 1) on the Commitment, UUSA reached the conclusion that it was appropriate to submit the aforementioned SAR Chapter 5 changes via this license amendment request (LAR-10-07) for formal NRC review and approval.

2 Proposed Changes

2.1 Summary of Proposed Change

The changes proposed herein to Chapter 5 of the SAR were initially made under URENCO USA's (UUSA) approval authority, in what UUSA perceived to be allowed in accordance with 10 CFR 70.72(c).

This proposed change provides for revising SAR Chapter 5 as documented in Configuration Change packages CC-LS-2010-0012, Changes to Clarify Nuclear Criticality Safety; CC-EG-2010-0021, Criticality Assessments; CC-EG-2007-0314, Enrichment Technology Corporation (ETC) Calculation ETC4053126, Issue 1, Risk of Criticality following a Seismic Event Affecting the NEF Cascade System; and CC-LS-2009-0006, SAR Enhancements for First Cascade Online.

The SAR Chapter 5 changes provided clarity, distinguished between systems bounded at 6% ^{235}U and 1.5% enrichment levels, and alignment with ETC reports for criticality assessments. Sections updated included 5.1.1, 5.1.2, 5.1.4, 5.2.1, and Tables 5.1-1 and 5.1-2.

2.2 Modification to Safety Analysis Report

2.2.1 CC-EG 2010-0021, SAR Tables 5.1-1 and 5.1-2

SAR Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2, was changed to increase dimensions of process components to correct an error for the calculated safe values in the table. Changes identified in LBDCR-10-0012 as noted in Enclosure 2. (2010-006 NOV B Example 1)

Another change is SAR Table 5.1-2 in which the Safety Criteria Values were revised as necessary to reflect updated criticality assessment data for the associated Buildings, Systems, and Components. Changes identified in LBDCR-10-0012 as noted in Enclosure 2. (Not an example in the NOV)

2.2.2 CC-LS-2010-0012, SAR Sections 5.1.1, 5.1.2, 5.1.4, 5.2.1.2, 5.2.1.3 and Table 5.1-1

SAR Section 5.1.1 was revised to expand the analyses that were performed at 1.5 $^{w}/_{o}$ to include the entire Dump System. The Dump System includes the Tails Take-off System and the Contingency Dump System. Changes identified in LBDCR-10-0037 as noted in Enclosure 2. (2010-006 NOV B Example 2)

SAR Section 5.2.1.3.4 was changed to state that limits placed on movement of individual vessels containing enriched uranium by procedures or work plans. Nuclear Criticality

Safety Evaluations for some vessels have reduced or eliminated the separation distance from 60 cm. (2010-006 NOV B Example 4)

Other changes included: SAR Section 5.1.2 was revised to clarify enrichment control Parameters; Section 5.1.4 to update/clarify safety criteria for components and systems listed in Tables 5.1-1 and 5.1-2, and noting the exception related to the Dump Systems; Section 5.2.1.3.2 change provides a more detailed discussion of enrichment control limits including enrichment analysis assumptions; Section 5.2.1.2 and Table 5.1-2 were revised to add the Tails Take-Off System to clarify that, along with the Contingency Dump System it is also part of the Dump System. Changes identified in LBDCR-10-0037 as noted in Enclosure 2. (Not an example in the NOV)

2.2.3 CC-LS-2009-006, SAR Sections 5.1.2, 5.2.1.3 and Table 5.1-2

SAR Section 5.1.2 change provides a Neutron Absorption discussion related to material at the NEF to take credit for neutron absorbers in standard material used in construction and processes. Changes identified in LBDCR-09-0113 as noted in Enclosure 2. (2010-006 NOV B Example 3).

Other changes include Section 5.2.1.3.3 and Table 5.1-2 which were revised to remove the trade name "Fomblin." Changes identified in LBDCR-09-0113 as noted in Enclosure 2. (Not an example in the NOV)

2.2.4 CC-LS-2007-0314 SAR Table 5.1-1

SAR Table 5.1-1 was revised to update the no double batching parameter Safe Value from 19.5 kg U to 20.1 kg U. Changes identified in LBDCR-08-0106 as noted in Enclosure 2. (Not an example in the NOV)

3 Technical Analysis of Proposed Changes

3.1 Technical Basis for Change

3.1.1 SAR Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2

UUSA identified a technical error in the calculation of the UO_2F_2/water mixture density in 2007 as documented in Condition Report CR 2007-0221. The error was associated with the use of the empirical constants in the Johnson and Krause method, resulting in an over-estimate in the densities. The erroneous densities are higher than they should be when using the same method correctly and need to be corrected.

UUSA performed a comprehensive study to determine a suitable, conservative and authoritative method for calculations of the UO_2F_2/water mixture densities. The study included the following four methods available:

- Johnson & Krause (Journal of the American Chemical Society, V. 75 [p. 4594], 1953)
- Jordan & Turner (ORNL/TM-12292, 1992)
- Garner et al (DEG Report 352, 1961)
- Leclaire & Evo (IRSN Paper, 2007)

The focus was placed mainly on the first two methods, as the first method was the basis used in the original calculation for the license application, and the second method is a newer approach available in an Oak Ridge National Laboratory (ORNL) report that has been used in other uranium enrichment facilities for criticality analyses. The third (from the 1961 British report) and fourth methods (un-reviewed French paper) were evaluated for comparison purpose.

Table 1 provides the results of the study, along with the original densities used for various H/U ratios.

Table 1 Comparison of UO_2F_2/Water Mixture Densities at 6% Enrichment

H/U	$UO_2F_2 \cdot xH_2O$ Density (g/cm^3)				
	Original (erroneous)[a] Johnson and Kraus	**Corrected[b]** Johnson and Kraus	**Jordan and Turner[b]**	**Garner, et al[b]**	**Leclaire and Evo[b]**
1	N/A	5.894915	6.181085	NA	5.912591
2	N/A	5.229961	5.918394	NA	5.529356
3	N/A	4.723909	5.631467	NA	5.209836
4	N/A	4.325888	4.760000	4.760000	4.756468
5	N/A	4.004644	4.342312	4.330457	4.327080
6	N/A	3.739921	4.008109	3.988455	3.985224

H/U	$UO_2F_2 \cdot xH_2O$ Density (g/cm^3)				
	Original (erroneous)[a] Johnson and Kraus	**Corrected**[b] Johnson and Kraus	**Jordan and Turner**[b]	**Garner, et al**[b]	**Leclaire and Evo**[b]
7	3.771289	3.518011	3.734637	3.709707	3.706610
8	3.569753	3.329306	3.506719	3.478150	3.475175
9	3.388665	3.166875	3.313849	3.282736	3.279871
10	3.225062	3.025589	3.148519	3.115614	3.112850
11	3.076528	2.901572	3.005225	2.971056	2.968382
12	2.941074	2.791841	2.879835	2.844781	2.842188
13	2.817045	2.694063	2.769191	2.733526	2.731008
14	2.703053	2.606386	2.670837	2.634762	2.632312
15	2.597928	2.527322	2.582833	2.546498	2.544110
16	2.500674	2.455663	2.503626	2.467143	2.464812
17	N/A	2.390414	2.431961	2.395413	2.393135
18	2.331285	2.330753	2.366809	NA	2.328030
19	2.276507	2.275991	2.307321	NA	NA
20	2.226051	2.225549	2.252788	NA	NA

a. Table 5-1, AREVA 32-9035369-000, *NEF Criticality Assessment under Flooded Conditions*, September 2007.
b. Table 8, LES-M-0002-0, *Density of Uranyl Fluoride Calculation*, December 2007.

The original densities from the Johnson & Krause method were erroneously over-estimated. The corrected densities are lower, but non-conservative compared to the other three methods. URENCO USA selected the ORNL Jordan & Turner method as the preferred method to provide conservatism in the UO_2F_2/water mixture density.

3.1.2 SAR Section 5.1.1, Management of the NCS Program (1.5% vs. 6% enrichment)

The 1.5% enrichment represents an upper mean enrichment in a mixed process stream containing feed, product and tails. This enrichment was actually intended to be applicable to any system with a mixed process stream such as the Contingency Dump System (CDS) and any process associated with a cascade dump (e.g., dump to a tails cylinder). The original statement in SAR Section 5.1.1 is inappropriate, and necessitates a change to this section for clarification.

3.1.3 SAR Section 5.1.2, Control Methods for Prevention of Criticality (neutron absorbers)

UUSA proposes to revise SAR Section 5.1.2, Control Methods for Prevention of Criticality, which states UUSA does not use neutron absorbers as a criticality control parameter. The definition of a neutron absorber, from ANSI/ANS-8.21-1995, *use of fixed neutron absorbers in nuclear facilities outside reactors*, is "A neutron-capture material".

The current nuclear criticality analyses for UUSA credit neutron absorption for the following inherent structural materials to meet the margin of subcriticality or criticality safety criterion of $k_{eff} = k_{calc} + 3_{calc} < 0.95$:

- Chemical trap wall (0.72 cm thick) – SS304

- 30B cylinder wall (1.27 cm thick) – steel
- Roots pump wall (0.4 cm thick) – cast iron alloy

Including credit for neutron absorption by structural materials in criticality analyses reduces k_{eff} relative to the analyses without such credit, unless the material also provides neutron fission, moderation or reflection. Although the structural materials provide some reflection and moderation, the predominant characteristic is neutron absorption which results in a decrease in k_{eff}.

Credit for neutron absorption is necessary due to the combination of multiple normal and upset conditions in the analyses, which result in conservative modeling of the system. Removal of the undue conservatism or unnecessary multiple upset conditions may eliminate the need for such credit for neutron absorption.

Given the definition of neutron absorber in ANSI/ANS-8.21, the statement in SAR Section 5.1.2 needs to be revised. Pending further evaluation of conservatism in system modeling, UUSA proposes to change SAR Section 5.1.2 as follows:

> Neutron Absorbers
> Neutron Absorption is a factor in almost all of the materials at the NEF. The normal absorption of neutrons in standard materials used in the construction and processes at the NEF (uranium, fluorine, water, steel, etc.) is not specifically excluded as a criticality control parameter.
>
> Models incorporate conservative values based on the process function of the neutron absorber. Depending on the function of the material, the bounding value may be validated at receipt, after installation, based on process knowledge during operation or by periodic surveillance.
>
> Additional materials such as cadmium and boron for which the sole purpose would be to absorb neutrons are not incorporated in NEF processes. Solutions of absorbers are not used as a criticality control mechanism.

3.1.4 SAR Section 5.2.1.3.4, Vessel Movement Assumption

The SAR requirements stipulate that 60 cm spacing be maintained between an item in movement and other enriched uranium. These separation requirements are impractical and unnecessary. It is impractical because separation between installed components is less than 60 cm in some cases. Such components could never be moved without a SAR change. The requirement is unnecessary as evaluations and analyses review interaction of between components and may determine the separation is unnecessary to ensure nuclear criticality safety.

The proposed change is to clarify the vessel movement assumption of the SAR section 5.2.1.3.4 to state the following:

> The limits placed on movement of an individual vessel or a specified batch of vessels containing enriched uranium are specified in the facility procedures or work plans, both of which are reviewed by Nuclear

Criticality Safety. Specified limits may not be required based on bounding or process/system-specific NCS evaluations or analysis.

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

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

3.2 *Historical Changes*

As a result of the Notice of Violation and the creation of this License Amendment Request, a historical look at changes made to the Chapter 5 of the Safety Analysis Report has been performed. One change is identified to be included in this License Amendment Request. The change is identified as LBDCR-08-0106 and has been approved in CC-EG-2007-0314. This change along with others has been provided to the NRC in prior annual updates. However, this change is being included as it is similar to the items which are identified in the Notice of Violation. In fact, the reason for the change is identical to the error corrected in CC-EG-2010-021.

In addition, Table 2 is provided reflecting all of the changes made to Chapter 5 of the Safety Analysis Report since License issuance. All of these changes have been reviewed and with exception of those identified by the NRC and the one identified above, none of them resulted in a reduction in any margin of subcriticality.

Table 2
Historical Changes to Chapter 5 of the Safety Analysis Report

SAR Rev.	**LBDCR #**	**Configuration Change Number**	**Description of Change**
13a	07-0022	CC-EG-2007-112, 113, 114, 115, CC-EG-2006-039, 0041	Editorial, update of design code references
15a	07-0050	LBDCR only	Editorial, removed statement that no low enrichment facilities have had an accidental criticality
15a	07-0047	CC-EG-2007-0289	Editorial, to improve the review and approval process
16b	08-0046	LBDCR only	Organizational Changes
16e	08-0017	CC-EG-2008-0044	Noted fire protection sprinkler system not used in SBM and CRDB (Related to Reflection) Page 5.1-4. To conform to the current fire protection design.

SAR Rev.	LBDCR #	Configuration Change Number	Description of Change
16e	08-0050	CC-EG-2007-0124	Noted fire protection sprinkler system not used in SBM and CRDB (Related to Reflection) Page 5.1-5. OAR for CAB as QA-3 rather than QA-1.
16d	007-0002	LBDCR only	Deleted CAB and Post Mortem Areas, for CAB downgrade Page 5.1-4
17a	08-0106	CC-EG-2007-0314	**Included in LAR-10-07**
19a	09-0012	LBDCR only	Organizational Change
20b	09-0068	CC-EG-2008-0246 **LAR-09-07**	Deleted 48Y from text and table. Removal of applying IROFS27c and replacement with IROFS27e to the CRDB superstructure
20b	09-0071	LBDCR only	Deleted TSB and added CRDB as a needed change to reflect changes made in LAR-09 07
22a	09-0113	CC-LS-2009-006	**Identified in NOV, Included in LAR-10-07**
22a	09-0104	CC-OP-2009-004	Sub Hydrogen Fluoride for HF, CRDB not available for initial plant start-up
23a	09-0130	CC-LS-2009-0018	Corrected number of cascade halls –CC removed SWU specific capacity and corrected cascade minihalls number
24b	10-0017	CC-LS-2010-0009	Clarify Responsibilities
24c	10-0012	CC-EG-2010-0021	**Identified in NOV, Included in LAR-10-07**
24d	10-0037	CC-LS-2010-0012	**Identified in NOV, Included in LAR-10-07**
25b	10-0054	LBDCR only	Editorial
25b	10-0048	CC-EG-2010-0114	Editorial

4 Safety Significant Determination

4.1 Safety Significance

4.1.1 SAR Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2

The change to UO_2F_2/water mixture densities results in a change to the calculated critical and safe values in SAR Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2. Similar changes are also made in SAR Table 5.1-2, Safety Criteria for Buildings/Systems/Components.

The updated values in SAR Tables 5.1-1 and 5.1-2 are based on the conservative UO_2F_2/water mixture densities determined by the Jordan & Turner method. Further, the safe values were calculated, using the criticality safety criterion of $k_{eff} = k_{calc} + 3\sigma_{calc} < 0.95$, which includes the approved administrative margin of 0.05. All the safe values in SAR Table 5.1-1 meet the "significant margin" requirement with a margin of at least 10% between the actual design parameter value of the component and the value of the corresponding critical design attribute, as the actual design parameter values for SBD favorable-geometry components are required to be no greater than the safe values.

The changes are considered acceptable for the following reasons:

- The change in the safe values reflected a necessary correction to the technical error identified in CR 2007-0221 through the Corrective Action Program.
- The most conservative method was used to calculate the UO_2F_2/water mixture density.
- No change was made to the criticality safety criterion of $k_{eff} = k_{calc} + 3\sigma_{calc} < 0.95$, which includes the approved administrative margin of 0.05 for deriving the safe values.
- No change was made to the USL determined in the MONK 8A Validation and Verification report where USL was based on the approved administrative margin of 0.05.
- A margin of at least 10% is maintained between the safe and critical values which include $3\sigma_{calc}$ in k_{eff} to meet the "Significant Margin" requirements for the physical parameters.

4.1.2 SAR Section 5.1.1, Management of the NCS Program (1.5% vs. 6% enrichment)

The clarification made to SAR Section 5 for the enrichment parameters and assumptions does not change the criticality safety criterion of $k_{eff} = k_{calc} + 3\sigma_{calc} < 0.95$, which is applicable to both systems and components associated with a cascade dump at 1.5% enrichment and other facility systems at 6% enrichment. Use of 6% enrichment would be considerably more conservative than 1.5%, however, the intent is to use 1.5% enrichment for any system or process associated with a cascade dump. The intended

use of 1.5% enrichment besides the CDS NaF traps is evident in Document UPD 0202631B, *Criticality Safety Evaluation of Evacuating an Assay Unit into a Single Tails Cylinder*, dated December 5, 2002, which is a supporting document for the License Application.

Applying 6% enrichment to the systems or processes associated with a cascade dump would be unnecessarily conservative. If the 6% enrichment is used, then criticality safety criterion of $k_{eff} = k_{calc} + 3\sigma_{calc} < 0.95$ is not met for these systems or processes. However, the subcriticality requirements would remain unchanged, and the margin of subcriticality of 0.05 would still have to be satisfied regardless of the enrichment used.

The methodology employed in the criticality calculations is consistent with the approved technical methodology for determining limits and controls (discussed in SAR 5.2.1.2). A significant margin is defined in the SAR (Section 3.2.5.2), ISA Summary (Section 3.1.1.5.2) and NUREG-1827 (Section 3.3.3.2.2.2), and defined for SBD components as follows:

- SBD Favorable Geometry Components – A margin of at least 10%, during both normal and upset conditions, between the actual design parameter value of the component and the value of the corresponding critical design attribute (k_{eff} = 1.0). [Note - The minimum 10% margin means that the ratio of the actual design parameter value (diameter, slab thickness and volume) of the component to the corresponding critical value is 0.90 or less. In no case does the actual design parameter value exceed the safe value (k_{eff} = 0.95). Both k_{eff} for the critical and safe values include $3\sigma_{calc}$.
- SBD Non-Favorable Geometry Components – $k_{eff} = k_{calc} + 3\sigma_{calc} < 0.95$

No safety significance results from the use of 1.5% enrichment for the system or process associated with a contingency dump, because of the license condition which limits the product enrichment to 5%. Similar operating assay units at a URENCO European facility indicate that in the event of the enrichment control on a centrifuge cascade failing and an enrichment of up to 8% was reached, the maximum mean enrichment would be 1.4%. Therefore, at the license limit of 5%, additional margin is available to the use of 1.5% mean enrichment for the cascade dump.

4.1.3 SAR Section 5.1.2, Control Methods for Prevention of Criticality (neutron absorbers)

The proposed changes to SAR Section 5.1.2 recognize that almost all materials modeled in criticality analyses absorb neutrons. It differentiates between neutron absorbers present for structural or process purposes and absorbers present for the sole purpose of absorbing neutrons. Structural or process materials that absorb neutrons are allowed. Absorbers present for the sole purpose of absorbing neutrons are not allowed.

The basis for differentiating is that neutron absorbers present for structural or process purposes are not considered to be "fixed neutron absorbers". ANSI/ANS-8.21 applies to the use of "fixed neutron absorbers". "Fixed neutron absorbers" are referred to as "poisons" in Section 1 of ANSI/ANS-8.21. To qualify as a "poison", the material must be

a nonfissionable neutron absorber, generally used for criticality control per definition of "neutron poison" in LA-11627-MS, Glossary of Nuclear Criticality Terms.

ANSI/ANS-8.21 recognizes the nuclear safety practices described in ANSI/ANS-8.1 Section 4 apply. ANSI/ANS-8.1 makes a clear distinction between neutron absorbers and the nuclear characteristics of process materials and equipment. ANSI/ANS-8.1, Section 4.2.3, Geometry Control says;

> "Full Advantage may be taken of any nuclear characteristics of the process materials and equipment."

The section invites using nuclear characteristics of process materials and equipment and makes no reference to them being neutron absorbers or to ANSI/ANS-8.21. By contrast, ANSI/ANS-8.21 requirements are cited in ANSI/ANS-8.1, Section 4.2.4 on Neutron Absorbers. Therefore, ANSI/ANS-8.1 makes a clear distinction between neutron absorbers and the nuclear characteristics of process materials and equipment.

Structural materials used at UUSA obviously do not meet the definition of a "fixed neutron absorber", as these materials are needed for structural purposes. These materials lack the specific nuclear characteristics that are unique to "fixed neutron absorbers" such as boron, cadmium and hafnium, including:

- Significantly high neutron absorption cross section
- Depletion of material over time due to neutron exposure
- Effect of neutron flux depressions in the absorber region

Many of the ANSI/ANS-8.21 requirements ensure the fixed neutron absorber remains in place and unmodified by man or environment. The potential for removal or degradation of structural materials is negligible because their continued presence is necessary to maintain plant operations.

NUREG-1520, Section 5.4.3.4.2(15)(b) states that "When using fixed neutron absorbers, the applicant commits to ANSI/ANS-8.21-1995." Since structural materials are not used as fixed neutron absorbers at UUSA, UUSA believes that ANSI/ANS-8.21-1995 does not apply to structural materials.

Nevertheless, UUSA meets the intent of ANSI/ANS-8.21, as described in letter LES-10-00112-NRC, dated 28-May-2010. The structural material at the thickness credited in the criticality analyses has been verified through a QL-1 receipt inspection process as part of the SBD attributes. Rationale is provided for the requirements that are not applicable. For example, ANSI/ANS-8.21 Section 5.2.2.1 requires calculation methods to replicate the effect of neutron flux depressions associated with localized neutron absorbers. The requirement is not applicable because localized neutron absorbers are not used. Further, standard materials such as steel are absent of the effect of neutron flux depressions as observed for boron and cadmium.

Therefore, the safety significance of the proposed change is negligible. It differentiates between structural materials that absorb neutrons and materials used for the sole purpose of absorbing neutrons. Although the potential for removal or degradation of structural materials is negligible, UUSA complies with the intent of ANSI/ANS-8.21.

4.1.4 SAR Section 5.2.1.3.4, Vessel Movement Assumption

UUSA performed a QL-1 nuclear criticality safety evaluation (NCSE) to address criticality safety requirements for movement of components or vessels. The NCSE is conducted in accordance with Procedure EG-3-3200-01, *Nuclear Criticality Safety Evaluations*, and documented in NCS-CSE-021, Rev. 0, *Movement of Components*. As a result, the spacing requirements are changed and SAR Section 5.2.1.3.4 was revised to incorporate the results of NCS-CSE-021. The revision did not change the criticality safety criterion of $k_{eff} = k_{calc} + 3\sigma_{calc} < 0.95$ or the 0.05 margin of subcriticality.

Since the proposed change does not alter the subcriticality requirement, there is no safety significance associated with this change.

5 Environmental Considerations

There are no significant environmental impacts associated with the changes proposed in this License Amendment Request. The proposed changes do not meet the criteria specified in 10 CFR 51.60 (b) (2) since they do not involve a significant expansion of the site, a significant change in the types of effluents, a significant increase in individual or cumulative occupational radiation exposure, or a significant increase in the potential for or consequences from radiological accidents. Consequently, a separate supplement to the Environmental Report is not submitted.

ENCLOSURE 2

Mark-Ups to Safety Analysis Report (SAR) Pages

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SAFETY ANALYSIS REPORT
LAR-10-07

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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, Nuclear Criticality Safety In Operations with Fissionable Materials Outside Reactors. The adopted double contingency principle states "process design should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible." Each process that has accident sequences that could result in an inadvertent nuclear criticality at the 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.

The plant will produce no greater than 5.0 $^{w}/_{o}$ enrichment. However, as additional conservatism, ~~the~~ most nuclear criticality safety analyses for enriched material 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. The exceptions to this are the systems and components associated with a cascade dump which are analyzed assuming 1.5 $^{w}/_{o}$. These include the Contingency Dump System equipment and piping on the 2nd floor of the Process Services Area and the Tails Take-off System. 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 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: LBDCR-10-0037

- An NCS program to meet the regulatory requirements of 10 CFR 70 (CFR, 2003a) 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, 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, Nuclear Criticality Safety Training. 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.
- Information obtained from the analysis of jobs and tasks in accordance with Section 11.3.

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. Nuclear Criticality Safety Evaluations and Analyses 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 nuclear criticality safety~~NCS~~. The determination of the safe values of the major controlling parameters used to control criticality in the facility is described below. LBDCR-09-0113

Moderation control is in accordance with ANSI/ANS-8.22, Nuclear Criticality Safety Based on Limiting and Controlling Moderators. However, for the purposes of the criticality analyses, it is assumed that UF_6 comes in contact with water to produce aqueous solutions of UO_2F_2 as described in Section 5.2.1.3.3, Uranium Accumulation and Moderation Assumption. A uniform aqueous solution of UO_2F_2, and a fixed enrichment are conservatively modeled using 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 $^w/_o$.

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 NEF 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 ~~traps~~ equipment and piping on the 2nd floor of the Process Services Area and the Tails Take-off System which are limited to 1.5 $^w/_o$ ^{235}U. LBDCR-10-0037

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

All Separation Plant components, which handle enriched UF_6, other than the Type 30B cylinders and the first stage UF_6 pumps and contingency dump chemical traps, 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 Cylinder Receipt and Dispatch Building 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 NEF are as follows:

Enrichment

~~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 NEF 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.~~ Enrichment is controlled to limit the percent ^{235}U within any process vessel or container to a maximum of 5% except for the systems and components associated with a cascade dump. For added conservatism the systems controlled to 5% are analyzed at 6%. LBDCR-10-0037

Assuming a product enrichment of 6% limits the upper bound for the average cascade enrichment to less than 1.5%, the systems and components associated with a cascade dump (Tails Take-off System, Contingency Sump System) are conservatively analyzed at 1.5%

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} = kcalc + 3 σcalc < 0.95.

The safe values of geometry/volume in Table 5.1-1 define the characteristic dimension of importance for a single unit such that nuclear criticality safety is not dependent on any other parameter assuming 6 $^{w}/_{o}$ ^{235}U for safety margin. LBDCR-10-0037

Moderation

Water and oil are the moderators considered in NEF. At NEF 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 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 Nuclear Criticality Safety Evaluations and Analyses. The possibility of full water reflection is considered but the layout of the NEF 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. In addition, automatic sprinklers are excluded from Separations Buildings and the CRDB. Fire protection standpipes are located in enclosed stairwells, or are arranged such that flooding from these sources is highly unlikely. Therefore, full water reflection of vessels has therefore been discounted. 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.

Interaction

Nuclear criticality safety evaluations and analyses consider the potential effects of interaction. A non-interacting unit is defined as a unit that is spaced an approved distance from other units such that the multiplication of the subject unit is not increased. Units may be considered non-interacting when they are separated by more than 60 cm (23.6 inches).

If a unit is considered interacting, nuclear criticality safety analyses are performed. Individual unit multiplication and array interaction are evaluated using the Monte Carlo computer code MONK8A to ensure k_{eff} = kcalc + 3 σcalc < 0.95.

Neutron Absorbers

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

Models incorporate conservative values based on the process function of the neutron absorber. Depending on the function of the material, the bounding value may be validated at receipt, after installation, based on process knowledge during operation or by periodic surveillance. LBDCR-09-0113

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

Concentration~~,~~ and Density ~~and Neutron Absorbers~~

NEF does not use ~~mass~~either concentration~~,~~ or density~~, or neutron absrobers as a criticality~~ control parameter.

5.1.3 Safe Margins Against Criticality

Process operations require establishment of criticality safety limits. The facility UF_6 systems involve mostly gaseous operations. These operations are carried out under reduced

atmospheric conditions (vacuum) or at slightly elevated pressures not exceeding three atmospheres. It is highly unlikely that any size changes of process piping, cylinders, cold traps, or chemical traps under these conditions, would lead to a criticality situation because a volume or mass limit may be exceeded.

Within the Separations Building, 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 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/ANS 8.22). 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 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 with the exception of the Tails Take-off and Contingency Dump Systems. These systems are limited to the maximum process system average enrichment, 1.5%. LBDCR-10-0037

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

5.1.5 Organization and Administration

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

The Criticality Safety Officer reports to the Health and Safety Manager as described in Chapter 2, Organization and Administration. The Health and Safety 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 Officer include the following:

- Establish the Nuclear Criticality Safety Program, including design criteria, procedures, and training
- Assess normal and credible abnormal conditions
- Determine criticality safety limits for controlled parameters, with input from the Criticality Safety Engineers

In accordance with the guidance in NUREG-1520, code validation for the specific application has been performed (see AREVA in ISAS table 3.0-1). 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 (see AREVA in ISAS table 3.0-1) for the National Enrichment Facility. In addition, the MONK8A Validation and Verification report (see AREVA in ISAS table 3.0-1) satisfies the commitment to ANSI/ANS-8.1 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 following equation from NUREG/CR-6698, Guide for Validation of Nuclear Criticality Safety Calculational Methodology:

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

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 σBias from the MONK8A Validation and Verification (see AREVA in ISAS table 3.0-1) is 0.0085 and a value of 0.05 is assigned to the subcritical margin, ΔSM. The term ΔAOA 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 ΔAOA 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 area of applicability to include 1.5% enrichment and the term ΔAOA is set to 0.0014 to account for this extrapolation. Thus, the USL becomes:

- USL = 1 + 0 – 0.0085 – 0.05 = 0.9415 (for systems and components NOT associated with the Contingency Dump System)
- USL = 1 + 0 – 0.0085 – 0.05 – 0.0014 = 0.9401 (for the Contingency Dump System and Tails Take-off System) LBDCR-10-0037

NUREG/CR-6698 indicates 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 NEF operations is inherently low. The low risk warrants the use of an alternate approach.

At the low enrichment limits established for the NEF, 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 NEF 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 inleakage 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 NEF.

Therefore, due to the low risk of accidental criticality associated with NEF operations and the margin that exists in the design and operation of the NEF with respect to nuclear criticality safety, a margin of subcriticality for safety of 0.05 (i.e., $k_{eff} = k_{calc} + 3\sigma_{calc} < 0.95$) is adequate to ensure subcriticality is maintained under normal and abnormal credible conditions. As such, the NEF 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 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.~~Enrichment is controlled to limit the percent ^{235}U within any process vessel or container to a maximum of 5% except for the systems and components associated with a cascade dump. For added conservatism the systems controlled to 5% are analyzed at 6%. LBDCR-10-0037

Assuming a product enrichment is 6% limits the upper bound for the average cascade enrichment to less than 1.5% the systems and components associates with a cascade dump (Tails Take-off System. Contingency Dump System) are conservatively analyzed at 1.5%

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 UO2F2·1.5H2O and UO2F2·2H2O 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 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. LBDCR-09-0113

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

Of the subset of individual vessels or groups of vessels that do not have specified controls but are bounded by a the single-parameter SBD limits in Table 5.1-1, separation must be maintained at least 60 cm (23.6 in) from any other enriched uranium. LBDCR-10-0037

Vessels or groups of vessels that do not comply with either of the statements above must not be moved without the written approval of the Criticality Safety Officer. LBDCR-10-0037

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.

5.6 CHAPTER 5 TABLES

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

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~~30.3 L (~~7.6~~8.0 gal)	~~21.6~~22.9 L (~~5.7~~6.1 gal)	~~0.75~~0.76	
Cylinder Diameter	~~26.2~~ 26.6cm(10.5~~3~~ in)	~~23.6~~23.9 cm (9.4~~3~~ in)	0.90	
Slab Thickness	~~12.6~~12.8 cm (5.0 in)	~~10.7~~11.1 cm (~~4.2~~4.4 in)	~~0.85~~0.87	
Water Mass	~~17.3~~18.5 kg H_2O (~~38.1~~40.8 lb H_2O)	~~12.7~~14.2 kg H_2O (~~28.0~~31.1 lb H_2O)	~~0.73~~0.77	
Areal Density	~~11.9~~ 11.8 g/cm^2 (~~24.4~~ 24.2 lb/ft^2)	~~9.8~~ 9.9g/cm^2 (~~20.1~~ 20.3 lb/ft^2)	~~0.82~~0.84	LBDCR-10-0012
Uranium Mass	~~37~~ 36.7 kg U (~~81.6~~80.9 lb U)			
- no double batching		~~26.6~~26.8 kg U (~~58.6~~59.1 lb U)	~~0.72~~0.73	
- double batching		~~16.6~~16.5 kg U (~~36.6~~36.4 lb U)	0.45	
Values for 6.0 $^w/_o$ enrichment				
Volume	~~24~~ 25.3 L (~~6.3~~ 6.7 gal)	~~18~~ 19.3 L (~~4.8~~ 5.1 gal)	~~0.75~~0.76	
Cylinder Diameter	~~24.4~~ 24.8 cm (~~9.6~~ 9.8 in)	~~21.9~~ 22.4 cm (~~8.6~~ 8.8 in)	0.90	
Slab Thickness	~~11.5~~ 11.6 cm (~~4.5~~ 4.6in)	~~9.9~~ 10.1 cm (~~3.9~~ 4.0 in)	~~0.86~~0.87	LBDCR-10-0012
Water Mass	15.4 kg H_2O (34.0 lb H_2O)	~~11.5~~11.9 kg H_2O (~~25.4~~26.2 lb H_2O)	~~0.75~~0.77	
Areal Density	~~9.5~~9.4 g/cm^2 (~~19.5~~19.3 lb/ft^2)	~~7.5~~7.9 g/cm^2 (~~15.4~~16.2 lb/ft^2)	~~0.79~~0.84	
Uranium Mass	27 kg U (59.5 lb U)			
- no double batching		~~19.5 kg U (43.0 lb U)~~20.1 kg U (29.7 kg UF_6)	~~0.72~~0.74	LBDCR-10-0012 LBDCR-08-0106
- 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~~22.4 cm (~~8.6~~8.8 in)	
Product Cylinders (30B)	Moderation	H < ~~0.95~~0.98 kg (~~2.09~~2.16 lb)	
UF_6 Piping	Diameter	< ~~21.9~~22.4 cm (~~8.6~~8.8 in)	
Chemical Traps	Diameter	< ~~21.9~~22.4 cm (~~8.6~~8.8 in)	LBDCR-10-0012
Product Cold Trap	Diameter	< ~~21.9~~22.4 cm (~~8.6~~8.8 in)	
Contingency Dump System ~~Traps~~Tails System	Enrichment	1.5 $^{w}/_{o}$ ^{235}U (used in NCS)	LBDCR-10-0037
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~~19.3 L (~~4.8~~5.1 gal)	
Individual Uranic Liquid Containers, e.g., ~~Fomblin~~PFPE Oil Bottle, Laboratory Flask, Mop Bucket	Volume	< ~~18.0~~19.3 L (~~4.8~~5.1 gal)	LBDCR-10-0012 LBDCR-09-0113
Vacuum Cleaners Oil Containers	Volume	<~~18.0~~19.3 L (~~4.8~~5.1 gal)	