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MOX SERVICES, LLC

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Washington, DC 20555

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DCS-NRC-000225

Subject: Docket Number 070-03098
Shaw AREVA MOX Services
Mixed Oxide Fuel Fabrication Facility
Submittal of Responses to Request for Additional Information Regarding the Review of the Criticality Safety Aspects of the Mixed Oxide Fuel Fabrication Facility License Application Request

Reference: (A) Letter, D. Tiktinsky (NRC) to D. W. Gwyn (MOX Services), *Request for Additional Information Regarding the Review of the Criticality Safety Aspects of the Mixed Oxide Fuel Fabrication Facility License Application Request*, April 30, 2008

(B) Letter, K. David Stinson (MOX Services) to Document Control Desk (NRC), Submittal of License Application, DCS-NRC-000202, 4 January 2007

Shaw AREVA MOX Services, LLC (MOX Services) hereby submits to the U.S. Nuclear Regulatory Commission (NRC) responses to the Reference (A) Request for Additional Information (RAI), and the proposed DRAFT page changes required to support a revision to Section 6 of the License Application (LA) for the Mixed Oxide Fuel Fabrication Facility (MFFF). Formal submittal of the updated LA will be included in the next MOX Services 2008 scheduled annual update.

Enclosure (1) provides the detail responses to the Reference (A) RAIs, and indicates corresponding changes to the LA. Enclosure (2) describes other changes to Section 6 which will be included in the 2008 annual update submittal. Enclosure (3) is the proposed revised Section 6 of the LA. This is a complete LA Section 6 revision proposal, and is intended to supersede in its entirety Section 6 of Reference (B) in next submittal of the LA.

If you have any questions, please feel free to contact me or Robert G. Foster, Nuclear Criticality Lead, Nuclear Safety, at (803) 819-2254.

Sincerely,

David Stinson
President and Project Manager

KDS/fhw

NMSS01

Enclosures
1) MOX Services Response to NRC Request for Additional Information (RAI) Regarding the Review of the Criticality Safety Aspects of the MFFF License Application (LA) Dated December 17, 2007
(2) Additional Proposed Page Changes to MFFF License Application, § 6
(3) Proposed Revised § 6, MFFF License Application

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Enclosure 1

Docket Number: 070-03098
MOX Services Response to
NRC Request for Additional Information (RAI)
Regarding the Review of the Criticality Safety Aspects of the
MFFF License Application (LA) Dated December 17, 2007

Section 6.1, "Organization and Administration for Nuclear Criticality Safety (NCS)"

RAI NCS-1

Clarify who has the authority to make commitments to the NRC and who has accountability for the overall safety of MFFF.

10 CFR 70.62(a) states: "Each licensee or applicant shall establish and maintain a safety program that demonstrates compliance with the performance requirements of §70.61." NUREG-1718, Section 6.4.3.1.C, states: "...the staff reports to the safety manager and is independent of operations at the highest practical level, preferably to an official at a sufficiently high level to have the authority to make commitments to the NRC and have accountability for the overall safety of the facility." It is not clear from License Application (LA) Section 6.1 who has this authority and accountability. This information is needed for regulatory clarity.

Response

The requirement for who has the authority to make commitments to the NRC is found in MOX Project procedure PP8-2, Regulatory Commitments, Section 2.0 Scope, which states: "Commitments to regulatory agencies may only be made by the President of MOX Services, the Deputy Project Director, or the Vice President - Engineering." The requirement for who has accountability for overall safety of the MFFF is found in LA Chapter 4.2.1, which states: "The manager of the plant is the MOX Services corporate officer responsible for managing all aspects of the MFFF, including safety and nuclear fuel manufacturing activities at the facility. This individual directs activities of licensed operations and staff functions through designated management personnel. The plant manager provides for the health and safety of the public and workers, and protection of the environment by delegating and assigning responsibility to qualified managers. The plant manager is directly responsible for the following functions: quality assurance, production, regulatory, and support services. These functions are accomplished by delegating and assigning responsibility to qualified personnel."

MOX Services proposes no changes to the LA in response to this RAI.

RAI NCS-2

Clarify LA Section 6.1 regarding the NCS organization. The NCS function is described as part of the MFFF organization in LA Section 6.1, but this is not clear from the description in LA Section 4.2.5. Also, LA Section 6.1 states the following: "The NCS organization, which reports to the manager of the support services function, is responsible for implementing applicable NCS practices for the MFFF," which seems to imply that there is a support services function manager.

10 CFR 70.62(a) states: "Each licensee or applicant shall establish and maintain a safety program that demonstrates compliance with the performance requirements of §70.61." NUREG-1718, Section 6.4.3.1.C, states: "...the staff reports to the safety manager and is independent of operations at the highest practical level, preferably to an official at a sufficiently high level to have the authority to make commitments to the NRC and have accountability for the overall safety of the facility." It is not clear from LA Section 6.1 who the NCS staff reports to (i.e., the NCS function manager or the support services function manager). This information is needed for regulatory clarity.

Response

LA Section 4.2.5 Support Services Function, states: "The managers of this (Support Services) function are also responsible for the HS&E functions of fire safety, criticality safety, and safety analysis." LA Section 6.1 Organization and Administration for NCS, 2nd paragraph, states: "The NCS organization, which reports to the manager of the support services function, is responsible for implementing applicable NCS practices for the MFFF." Both LA Sections 4.2.5 and 6.1 are consistent and clearly state that the NCS organization is under the support services function and report to the support services manager. As discussed above, the NCS "organization" is under the support services function and reports to the support services manager. The NCS organization is responsible for implementing applicable NCS practices for the MFFF. LA Section 6.1 Organization and Administration for NCS, page 6-2, 3rd paragraph, states: "The manager of the NCS function has the authority and responsibility to assign and direct activities for the NCS function. The minimum qualifications for the manager of the NCS function are a Bachelor's degree in science or engineering, or equivalent, with at least three years of nuclear industry experience in criticality safety. The manager of the NCS function has management or technical experience in the application and/or direction of criticality safety programs for nuclear facilities involving SNM." This paragraph clearly establishes that there is a "manager of the NCS function" and outlines this person's minimum qualifications. The NCS staff report to the manager of the NCS function and as stated above, the NCS organization/function, which includes the manager of the NCS function/organization, is under the support services function and reports to the support services manager.

MOX Services proposes no changes to the LA in response to this RAI.

Section 6.2.2, "Audits and Assessments"

RAI NCS-3

Provide a commitment to audit the NCS program at least quarterly such that all NCS aspects of management measures are audited at least every 2 years, or in the alternative, provide a commitment to use a justification on the basis of risk, such as based on the results of the ISA, to determine a frequency for audits.

10 CFR 70.62(a) states: "Each licensee or applicant shall establish and maintain a safety program that demonstrates compliance with the performance requirements of §70.61." NUREG-1718, Section 6.4.3.2.C.iv, states: "The applicant commits to conducting and documenting periodic NCS audits. A less than quarterly frequency may be justified on the basis of risk, such as based on the results of the ISA." It is not clear from LA Section 6.2.2 that there is a commitment to use a risk-informed methodology determination to determine the frequency of audits if less than a quarterly frequency is performed. This information is needed for regulatory clarity.

Response

LA § 6.2.2 was revised to incorporate a commitment to audit the NCS program such that all NCS aspects of management measures are audited at least every two years. Changes made to §6.2.2 are found in Enclosure (3).

RAI NCS-4

Provide a commitment to conduct and document weekly walkthroughs of all operating SNM process areas, or in the alternative, provide a commitment to use a justification on the basis of risk, such as based on the results of the ISA, to determine a frequency for walkthroughs.

10 CFR 70.62(a) states: "Each licensee or applicant shall establish and maintain a safety program that demonstrates compliance with the performance requirements of §70.61." NUREG-1718, Section 6.4.3.2.C.iv, states: "The applicant commits to conduct and document periodic walkthroughs of all operating [SNM] process areas. ... A less than weekly frequency may be justified on the basis of risk, such as based on the results of the ISA." It is not clear from LA Section 6.2.2 that there is a commitment to use a risk-informed methodology determination to determine the frequency of walkthroughs if less than a weekly frequency is performed. This information is needed for regulatory clarity.

Response

A commitment has been added to the LA as seen in § 6.2.3 of Enclosure (3) in response to this Request for Additional Information.

Section 6.2.3, "Procedures"

RAI NCS-5

Provide a commitment to review procedures and their implementation at least annually to ascertain that procedures are being followed and that process conditions have not been altered to adversely affect NCS, or in the alternative, provide a commitment to use a justification on the basis of risk, such as based on the results of the ISA, to determine a frequency for procedure review.

10 CFR 70.62(a) states: "Each licensee or applicant shall establish and maintain a safety program that demonstrates compliance with the performance requirements of §70.61." NUREG-1718, Section 6.4.3.2.C.ii, states: "Operations are reviewed at least annually to ascertain that procedures are being followed and that process conditions have not been altered to adversely affect NCS." It is not clear from LA Section 6.2.3 that there is a commitment to use a risk-informed methodology determination to determine the frequency of procedures reviews if less than an annual frequency is performed. This information is needed for regulatory clarity.

Response

A commitment has been added to LA as seen in § 6.2.4 of Enclosure (3) in response to this Request for Additional Information.

Section 6.3, "Nuclear Incident Monitoring System"

RAI NCS-6

Provide a commitment to have an alarm for the nuclear incident monitoring system that is clearly audible in areas that must be evacuated or that provides alternative notification methods that are documented to be effective in notifying personnel when evacuation is necessary.

10 CFR 70.24(a) states: "Each licensee authorized to possess special nuclear material in a quantity exceeding 700 grams of contained uranium-235, 520 grams of uranium-233, 450 grams of plutonium, 1,500 grams of contained uranium-235 if no uranium enriched to more than 4 percent by weight of uranium-235 is present, 450 grams of any combination thereof, or one-half such quantities if massive moderators or reflectors made of graphite, heavy water or beryllium may be present, shall maintain in each area in which such licensed special nuclear material is handled, used, or stored, a monitoring system meeting the requirements of either paragraph (a)(1) or (a)(2), as appropriate, and using gamma- or neutron-sensitive radiation detectors which will energize clearly audible alarm signals if accidental criticality occurs." It is not clear from LA Section 6.3 that there is a commitment to have a nuclear incident monitoring system that is clearly audible

if an accidental criticality occurs. This information is needed to ensure the alarm for the nuclear incident monitoring system is adequate.

Response

Clear audible alarms, visual light or other notification means shall be installed in the MFFF for all areas affected by the criticality event for prompt employee evacuation. In addition, the alarm signal is sent to an emergency control room for facility notification and response. Changes have been made to § 6.3 as documented in Enclosure (3).

RAI NCS-7

Provide a commitment to immediately render operations safe, by shutdown and quarantine if necessary, in any area where nuclear incident monitoring system coverage has been lost, until compensatory measures approved by the nuclear criticality safety function are in place or the alarm service has been restored. In the alternative, provide a commitment to specify the number of hours on a process-by-process basis before rendering operations safe.

10 CFR 70.24(a) states: "Each licensee authorized to possess special nuclear material in a quantity exceeding 700 grams of contained uranium-235, 520 grams of uranium-233, 450 grams of plutonium, 1,500 grams of contained uranium-235 if no uranium enriched to more than 4 percent by weight of uranium-235 is present, 450 grams of any combination thereof, or one-half such quantities if massive moderators or reflectors made of graphite, heavy water or beryllium may be present, shall maintain in each area in which such licensed special nuclear material is handled, used, or stored, a monitoring system meeting the requirements of either paragraph (a)(1) or (a)(2), as appropriate, and using gamma- or neutron-sensitive radiation detectors which will energize clearly audible alarm signals if accidental criticality occurs." It is not clear from LA Section 6.3 that there is a commitment to specify the number of hours on a process-by-process basis before rendering operations safe. This information is needed for regulatory clarity.

Response

§ 6.3 of the LA already contains a commitment to lost NIMS coverage, "If the NIM system, detection or alarm/notification capability, becomes unavailable, the allowable number of hours during which NIM system coverage is not available is determined on a process-by-process basis. Additionally, this paragraph will be revised to read as follows: The MFFF will maintain safe operations by immediately implementing compensatory measures (e.g., limit personnel access, halt SNM movement or activities, and quarantine affected equipment or areas) as necessary when the NIM system is unavailable or significantly degraded." Additionally, a statement will be added which clarifies that compensatory measures will be approved by the nuclear criticality safety function. There are also plans to use portable criticality monitoring equipment to supplement unusual conditions or failure of permanent equipment. These portable units will have the same specifications as the permanent equipment with the exception of the seismic criteria. There is no change to the LA in response to this RAI.

Section 6.4.1, "Nuclear Criticality Safety Evaluations (NCSE)"

RAI NCS-8

Clarify what MFFF components or systems require development of an NCSE (i.e., the threshold for requiring an analysis). LA Section 6.4.1 states that When an MFFF component or system is designed or modified, an NCSE is developed or updated to determine that the entire process will be subcritical....

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.2, states: "NCSEs should be considered the main source of information regarding the adequacy of criticality controls. NCSEs are the documents used to develop the safety basis of facility operations." The applicant has committed to develop NCSEs to document subcriticality of facility operations, but it is not clear whether there is a threshold for when an NCSE is required. This information is needed for regulatory clarity.

Response

Clarification about the components or systems has been added to LA § 6.4.1 as seen in Enclosure (3). Components potentially containing fissile materials undergoing design or modification that could affect credible criticality sequences predicate the development/update of an NCSE.

RAI NCS-9

Provide a commitment that NCSEs will only be performed by qualified NCS Engineers or qualified Senior NCS Engineers. The duties of a qualified NCS Engineer of Senior NCS Engineer in LA Section 6.1.1 include having the responsibility and authority to conduct activities assigned to the NCS Function, but the license application does not appear to require that only qualified NCS Engineers or Senior NCS Engineers are permitted to perform NCSEs.

10 CFR 70.62(a) states: "Each licensee or applicant shall establish and maintain a safety program that demonstrates compliance with the performance requirements of §70.61." NUREG-1718, Section 6.4.3.1, states: "To provide for NCS, the applicant's organization and administration implementing the safety program in 10 CFR 70.62(a) should be considered acceptable if the applicant has met the following acceptance criteria...(A)...the applicant has described organizational positions, functional responsibilities, experience, and adequate qualifications of persons responsible for NCS." There is no unequivocal statement in the license application that describes who is

qualified to perform criticality analysis. This information is needed to provide assurance of the adequacy of the NCS Program.

Response

LA § 6.4.1 has been updated to include that only qualified NCS Engineers or Senior NCS Engineers are permitted to perform NCSEs as seen in Enclosure (3).

RAI NCS-10

Provide a commitment that NCSEs will be peer reviewed by a qualified Senior NCS Engineer or NCS Manager prior to approval. Describe the NCSE approval process.

10 CFR 70.62(a) states: "Each licensee or applicant shall establish and maintain a safety program that demonstrates compliance with the performance requirements of §70.61." NUREG-1718, Section 6.4.3.1, states: "To provide for NCS, the applicant's organization and administration implementing the safety program in 10 CFR 70.62(a) should be considered acceptable if the applicant has met the following acceptance criteria...(B) The applicant commits to the endorsed requirements related to organization and administration in ANSI/ANS-8.1-1983...Where similar requirements also exist in ANSI/ANS-8.19-1996...the applicant commits to follow the more detailed requirements of ANSI/ANS-8.19-1996". ANSI/ANS-8.19-1996, Section 8.4, states "Before the start of operation, there shall be an independent assessment that confirms the adequacy of the nuclear criticality safety evaluation." Also, NUREG-1718, Section 6.3.3(F), states: "The commitment to prepare and maintain applicable safety basis documentation in enough detail so that criticality controls and double contingency analysis can be reviewed and inspected by NRC and licensee staff." This information is needed to provide assurance of the adequacy of the NCS Program.

Response

In LA § 6.4.1, a commitment has been made for the NCSE approval process. Prior to the approval of NCSEs, a peer review will be performed by a qualified Senior NCS engineer or NCS manager. This is documented in Enclosure (3). The approval of NCSEs is performed in accordance with MFFF project procedures. Briefly, NCSEs are reviewed by a Discipline Reviewer (a Senior NCS Engineer), Interdiscipline Reviewers (assigned by the NCS Manager), a Design Verifier (Senior NCS Engineer – usually the Discipline Reviewer), the Lead Engineer (NCS Manager), and the Responsible Manager (Nuclear Safety Manager).

RAI NCS-11

Revise your statement that The evaluation may include criticality calculations using validated calculational methodologies to demonstrate that both normal and credible abnormal conditions meet the required minimum margin of subcriticality, by replacing the underlined words with are subcritical, including the required minimum margin of subcriticality. The purpose of performing calculations is to demonstrate subcriticality; this is done by comparing calculated results to an Upper Subcritical Limit (USL) that accounts for bias, bias uncertainty, *and* the minimum margin of subcriticality. Thus the minimum margin of subcriticality is only *part of* a subcriticality demonstration.

10 CFR 70.61(d) states: "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." Criticality calculations are performed to demonstrate subcriticality, which includes but is not limited to consideration of an approved margin of subcriticality." This information is needed to ensure processes are adequately subcritical.

Response

LA § 6.4.1 has been updated to address this RAI. Found in Enclosure (3) a change has been made from "meet the required minimum margin of subcriticality" to "are subcritical, including the required minimum margin of subcriticality."

Section 6.4.2, "Analytical Methodology"

RAI NCS-12

State why Section 6.4.2 commits to ANSI/ANS-8.1-1983 (R1988), instead of the newer ANSI/ANS-8.1-1998. Revise your application to provide consistent references to this standard. Some references are given as ANSI/ANS-8.1-1983 (R1988), some as ANSI/ANS-8.1-1983, and some just as ANSI/ANS-8.1. Those references listed as ANSI/ANS-8.1, in particular, should be changed because there is no version number associated with them.

NUREG-1718, Section 6.4.2, states: "Regulatory Guide (RG) 3.71...endorses the American National Standards Institute's (ANSI's) and American Nuclear Society's (ANS's) ANSI/ANS 8 national standards as listed below in part or in full: ANSI/ANS-81-1983 (Reaffirmed in 1988), "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors." It is NRC practice to endorse specific versions of such standards. Use of other than the currently endorsed version should be justified. This information is needed for regulatory consistency.

Response

MOX Services commits to the ANSI/ANS-8.1-1998 standard. This change is implemented through Section 6 of the LA and is documented in Enclosure (3).

RAI NCS-13

Elaborate on your statement: The evaluations of the assumptions are based on realistic processes; conservative assumptions are analytically qualified so as to demonstrate the level of conservatism added. Explain what the underlined words mean, and provide an example of what this evaluation would entail.

The meaning of these words is unclear. This information is needed for regulatory clarity.

Response

Words have been added to § 6.4.2 to add clarification as is documented in Enclosure (3). "Realistic processes" has been reworded to become "realistic process conditions" and "analytically quantified" has been reworded to become "analytically justified." These words more clearly communicate the basis of assumptions in the NCSEs.

RAI NCS-14

Explain the logical relationship between the two halves of your statement: Defense-in-depth practices are incorporated in the MFFF, such as the preferential selection of first passive engineered control, secondly active engineered controls, and then administrative controls, where practical. While defense-in-depth and a preferred control hierarchy are both desirable, it does not appear that the preferred control hierarchy should be listed as an example of defense-in-depth practices.

10 CFR 70.64(b) states: "Facility and system design and facility layout must be based on defense-in-depth practices. The design must incorporate, to the extent practicable: (1) Preference for the selection of engineered controls over administrative controls to increase overall system reliability". NUREG-1718, Section 6.4.3.3.2(C), states: "The applicant commits to the preferred use of passive-engineered controls to ensure NCS. The applicant should commit to the following preference, in general, for controls to ensure NCS: (1) passive-engineered, (2) active-engineered, (3) augmented-administrative, and (4) simple-administrative." However, the relationship between defense-in-depth practices and the preferred control hierarchy is not clear. This information is needed for regulatory clarity.

Response

A rewording of § 6.4.2 occurred in response to this RAI as enclosed in Enclosure (3). A clarification was included that explained that the controls to prevent criticality in compliance with the double contingency principle adheres to a preferential selection. In addition to the preferential selection, defense-in-depth practices are used.

RAI NCS-15

Add the acceptance criterion to the bulleted list in LA Section 6.4.2 that optimum or worst-credible conditions will be assumed for parameters unless they are specifically controlled (or state where such a commitment can be found).

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". With regard to the evaluation of process conditions, NUREG-1718, Section 6.4.3.3.1(F), states The applicant commits to assuming credible optimum conditions (i.e., most reactive conditions physically possible) for each controlled parameter unless specific controls are implemented to limit the controlled parameter to a certain range of values. This information is needed to ensure processes are adequately subcritical.

Response

The acceptance criterion that optimum or worst credible conditions will be assumed for parameters unless they are specifically controlled was included in the bulleted list for acceptance criterion in the LA § 6.4.2 as documented in Enclosure (3).

RAI NCS-16

Clarify our commitment that IROFS associated with maintaining these controlled parameters are noted in the NCSE. State whether all NCS controls needed to meet the double contingency principle or ensure subcriticality will be IROFS. If not, explain when this statement will be applied.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". 10 CFR 70.61(e) states: "Each engineered or administrative control or control system necessary to comply with paragraphs (b), (c), or (d) of this section shall be designated as an item relied on for safety." However, it is not the NRC's position that all controls relied on to meet the double contingency principle, which is required in 10 CFR 70.64(a)(9), be designated as IROFS (see FCSS-ISG-03 for a detailed discussion of this). Therefore, it is necessary to clarify the intent of the statement in the license application. This information is needed for regulatory clarity.

Response

In LA § 6.4.2, in response to RAI NCS-16, it states that all criticality controls are IROFS as noted in the NCSEs. This is documented in Enclosure (3).

RAI NCS-17

Explain your statement that Evaluations based on <u>realistic component parameters</u> are performed to demonstrate that controlled parameters are maintained during both normal and credible abnormal conditions. In particular, explain the meaning of the underlined words and provide an example. Also, explain the statement that controlled parameters will be maintained during credible abnormal conditions. Abnormal conditions normally involve the loss of controlled parameters.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.2(E), states: "The applicant commits to describing controlled parameters for each process used as NCS control." NUREG-1718, Section 6.8, defines "abnormal condition" as "...A condition that can only be reached by exceeding the safety limits of a controlled parameter but that is planned for in CSEs." The phrase "realistic component parameters," and the statement that controlled parameters will be maintained during credible abnormal conditions, is unclear. This information is needed for regulatory clarity.

Response

The sentence referenced has been reworded to provide clarity. Calculations are performed and the evaluations summarize the calculations. In addition, abnormal conditions are further explained for clarity. A practical example of this is as follows: Even though the non-safety control system might fail and attempt to overfill a jar in a powder unit, and even though one of the redundant IROFS safety systems could hypothetically fail (a credible abnormal condition), the other redundant IROFS safety systems would prevent the overfilling of the jar maintaining the safe mass. These proposed changes are included in § 6.4.2 as documented in Enclosure (3).

RAI NCS-18

Explain your statement that Evaluations based on realistic component parameters are performed to demonstrate that controlled parameters are maintained during both normal and credible abnormal conditions. Summaries of these evaluations are provided in the NCSEs. If summaries of the aforementioned evaluations are contained in NCSEs, are the actual evaluations maintained as separate documents? Where?

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.2.0, states: "Tolerances on the controlled parameters should be conservatively taken into account in setting operating limits and controls established to prevent exceeding subcritical values of parameters." Evaluations (and not just summaries of these evaluations) are necessary to determine that controlled parameters will be maintained. This information is needed to ensure processes are adequately subcritical.

Response

In response to RAI NCS-18, as documented in Enclosure (3), LA § 6.4.2 the word "evaluations" was changed to "calculations" for clarity. The calculations support, are documented, and are summarized in the NCSEs.

Section 6.4.3, "Additional Technical Practices"

RAI NCS-19

Explain the underlined words in the following: "...that properly accounts for method bias, including appropriate processes, and uncertainty and administrative margin." It is not clear what these words mean or what they have to do with method bias.

The meaning of these words is unclear. This information is needed for regulatory clarity.

Response

The words "including appropriate processes" has been removed from § 6.4.3 of the LA as documented in Enclosure (3). This change was made to provide regulatory clarity as the words were unnecessary.

Section 6.4.4, "Criticality Control Modes"

RAI NCS-20

Remove the word initially from your statement: "Reliance is initially placed on equipment design using passive engineered controls, rather than administrative controls, where practical." The preference for passive over active and administrative controls should apply to both the initial facility design and future changes.

10 CFR 70.64(b) states: "The design must incorporate, to the extent practicable: (1) Preference for the selection of engineered over administrative controls to increase overall system reliability". NUREG-1718, Section 6.4.3.3.2(C), states: "The applicant commits to the preferred use of passive-engineered controls to ensure NCS. The applicant should commit to the following preference, in general, for controls to ensure NCS: (1) passive-engineered, (2) active-engineered, (3) augmented-administrative, and (4) simple-administrative." This information is needed for assurance that IROFS will be adequately reliable.

Response

The word initially has been removed from LA § 6.4.4 is response to RAI NCS-20. This is documented in Enclosure (3).

RAI NCS-21

Provide a commitment to demonstrate your adherence to the hierarchical preference of control (i.e., passive over active over administrative control; passive geometry control) as part of facility design. One way to demonstrate this is through a commitment to justify deviations from the preferred design hierarchy in the ISA, preferably in facility NCSEs.

10 CFR 70.64(b) states: "The design must incorporate, to the extent practicable: (1) Preference for the selection of engineered over administrative controls to increase overall system reliability". NUREG-1718, Section 6.4.3.3.2(C), states: The applicant should demonstrate how it is meeting this commitment to the preferred design approach [preference for passive over active or administrative controls], such as by providing justification when using other than passive-engineered control. This demonstration should also be documented in the ISA. NUREG-1718, Section 6.4.3.3.2.0, states: "The applicant should demonstrate how it is meeting this commitment [preference for geometry control], such as by providing justification when using other than passive geometry for criticality control." This information is needed for assurance that IROFS will be adequately reliable.

Response

MOX Services commits to use hierarchical preference to control to the extent practical. The explanation of "to the extent practical" has been added to § 6.4.4 and documented in Enclosure (3). "To the extent practical" means that the hierarchy is followed wherever practicable as determined by the process. For example, in liquid systems, the geometry of the vessels (Passive Engineered Controls) is used to ensure subcriticality. Similarly, in powder units, the geometry of the jars and the geometry of filling hoppers (Passive Engineered Controls) are used to ensure subcriticality. On the other hand, abnormal potential spills (due, for example, to equipment failure) can not be prevented by a passive control, since that is due to the failure of the control itself. Instead, redundant IROFS powder weighing systems (Active Engineered Controls) are employed to detect such conditions and automatically stop the process. Additionally, in those manual processes such as the laboratory operation, waste operations, infrequent maintenance crane operation, Administrative Controls are used to ensure safety.

RAI NCS-22

Explain your statement "Controlled parameters and techniques for controlling associated modes...are established and justified." Explain where this justification is documented.

10 CFR 70.64(b) states: "The design must incorporate, to the extent practicable: (1) Preference for the selection of engineered over administrative controls to increase overall system reliability". NUREG-1718, Section 6.4.3.3.2(C), states: The applicant should demonstrate how it is meeting this commitment to the preferred design approach [preference for passive over active or administrative controls], such as by providing justification when using other than passive-engineered control. This demonstration

should also be documented in the ISA. NUREG-1718, Section 6.4.3.3.2.0, states: The applicant should demonstrate how it is meeting this commitment [preference for geometry control], such as by providing justification when using other than passive geometry for criticality control. It is unclear where this justification is to be documented. This information is needed for regulatory clarity.

Response

Justification is provided in the NCSEs as is documented in LA § 6.4.4 of Enclosure (3).

Section 6.4.4.1, "Geometry Control"

Note that parallel language occurs in multiple sections corresponding to different controlled parameters. Where a question pertains to more than one parameter, it is listed under the first parameter to which the question applies (but other affected parameters are noted).

N.B. The list of affected parameters may not be comprehensive. The attempt to identify all parameters to which the question may apply notwithstanding, the applicant should review sections for all other parameters to ensure that any changes that may be necessary in other sections are also made.

RAI NCS-23

Revise your statement that Geometry parameters are established... to read Geometry limits are established.... In standard criticality terminology, geometry is considered a parameter and the specific values associated with that parameter are termed limits.

Also, explain how you will determine an adequate margin of subcriticality including margins to protect against uncertainties in process variables and against limits being accidentally exceeded, and state whether these will be based on k_{eff} sensitivity studies and the ability of controls to maintain operating limits.

This question also applies to the other parameters.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.4(D), states: The applicant commits to determining operation limits for controlled parameters, such that there is an adequate margin of safety to ensure the subcritical limit will not be exceeded. The applicant should commit to performing studies of the sensitivity of k_{eff} to variations in the parameters. The margin of safety should be based on these sensitivity studies and the ability of the control to maintain the operating limits. While the aforementioned statement in Section 6.4.4.1 satisfies the first sentence of the acceptance criterion, it does

not address the use of sensitivity studies to ensure those limits are appropriate. This information is needed to ensure that processes are adequately subcritical.

Response

In section 6.4.4 of the LA, a commitment to perform sensitivity studies was added. Also, consistency in the wording of the different controls was added. These proposed changes have been documented in Enclosure (3).

RAI NCS-24

Explain the reason for differences in your commitments to establish margins sufficient to account for uncertainties and variability, for the various parameters. In particular, staff notes the following differences in terminology:

Geometry, Isotopics, Moderation: including margins to protect against uncertainties in process variables and against limits being accidentally exceeded

Mass, Density: including margins to protect against uncertainties in process variables and against limits being inadvertently exceeded

Reflection, Interaction, Neutron Absorber, Heterogeneity, Physicochemical Control: no corresponding commitment

Concentration, Volume: including margins to protect against uncertainties in process variables

Verify that there is no intended difference between accidentally and inadvertently. Explain why five parameters (not including process variable control, which indirectly places limits on other parameters) do not have a corresponding commitment. Explain why the phrase and against limits being accidentally/inadvertently exceeded has been dropped from the sections on concentration and volume control. Furthermore, explain how adequate margins will be determined (e.g., through sensitivity studies).

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.4(D), states: The applicant commits to determining operation limits for controlled parameters, such that there is an adequate margin of safety to ensure the subcritical limit will not be exceeded." This information is needed to ensure that processes are adequately subcritical.

Response

The differences in terminology for the commitment to establish margins sufficient to account for uncertainties and variability, for various parameters have been eliminated in sections 6.4.4.1 though 6.4.4.13 of the LA to become consistent with each other. The change is documented in Enclosure (3). The discussion of adequate margins is made in the response to the RAI NCS-23.

RAI NCS-25

State how margin in geometry limits will be determined when standards or handbooks are used to determine subcriticality.

This question also applies to density control.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.2.2(C), states When using large single units, conservative margins of safety (such as 90% of the minimum critical cylinder diameter, 85% of the minimum critical slab thickness, and 75% of the minimum critical sphere volume) are used. Justification should be provided for proposed alternatives to these limits, taking system sensitivities into account....Reliance on engineering judgement does not substitute for this justification. These commitments are not necessary when relying on calculational methods for which bias and uncertainty have been determined by validation and an acceptable subcritical margin in k_{eff} applied. However, when subcriticality is determined by other means, the method of determining subcritical margin should be described. This information is needed to ensure that processes are adequately subcritical.

Response

The use of standards and handbooks to determine margins in limits on geometry and density are not implemented in the criticality safety strategy of MOX Services and as such, the words have been removed from LA sections 6.4.4.1, 6.4.4.3, 6.4.4.4, 6.4.4.3, 6.4.4.7, and 6.4.4.10. Additionally, the use of standards and handbooks to determine margins in limits to other parameters, except for mass, are not implemented. This is documented in Enclosure (3).

Section 6.4.4.2, "Mass Control"

RAI NCS-26

Revise your commitment that "When process variables can affect the bounding weight percent of SNM in the mixture, the SSCs or procedures that affect the process variables are <u>evaluated</u> to state ...are <u>controlled as IROFS in the NCSEs and ISA Summary</u>.

This question also applies to heterogeneity control.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are

subcritical". NUREG-1718, Section 6.4.3.3.2.1(A), states: "When process variables can affect the bounding weight percent of SNM in the mixture, controls to maintain the process variables are identified as IROFS in the NCSEs and ISA Summary." This information is needed to ensure that processes are adequately subcritical.

Response

The change predicated by RAI NCS-26 has been implemented in the sections on density (6.4.4.3), concentration (6.4.4.7), moderation (6.4.4.6), mass (6.4.4.2), heterogeneity (6.4.4.11), and physicochemical control (6.4.4.12). This is documented in Enclosure (3).

RAI NCS-27

Clarify the underlined words in your statement: "Theoretical densities for fissile mixtures are used, unless lower densities are ensured, or data are available." State that any data used for this purpose must be justified to be applicable and reliable in NCSEs.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.2.1(B), states: "Theoretical densities for fissile mixtures are used unless lower densities are ensured by the establishment of NCS controls." It does not contain this provision for relying on historical data for less than theoretical density. This information is needed to ensure that processes are adequately subcritical.

Response

For clarification, the words "or data are available" have been removed as it not applicable to the criticality safety strategy of MOX Services. This is documented in Enclosure (3).

Section 6.4.4.4, "Isotopic Control"

RAI NCS-28

Explain your statement: "In addition, the determination of isotopic content is based on compliance with the double contingency principle." Does this mean that when less than the bounding isotopic abundance is assumed (96wt% ^{239}Pu and 93.2wt% ^{235}U), it will be based on two independent isotopic measurements?

This question also applies to moderation control.

The meaning of these words is unclear. This information is needed for regulatory clarity.

Response

The meaning of the words dictates that when less than the bounding isotopic abundance is assumed, it will be based on two independent isotopic measurements. MOX Services do not use less than the bounding isotopic abundance for Pu (Powder dilutions are performed based upon dual independent mass measurements). MOX Services only reference less than 93.2% U when the material is diluted. Then confirmation is made with dual independent sampling.

RAI NCS-29

Clarify your statement: "Consideration is given to sample analysis and verification activities associated with MFFF and vendor (DOE)-supplied measurements. DOE (PDCF) and vendor data are qualified in accordance with an approved QA plan and are audited by the MFFF QA function. The use of qualified nondestructive assay (NDA) measurement systems is also acceptable in establishing compliance with the double contingency principle. Add a commitment that vendor data alone will not be used to determine isotopic content, but must be confirmed by MFFF measurements. NUREG-1718, Section 6.4.3.3.2.4(A), states: "...determinations of isotopic content shall be based on dual independent sampling and analysis of each lot of fissile material."

This question also applies to moderation control. (For moderation, if reliance is to be placed solely on vendor data, provide justification that moderation levels will not change during transportation and storage.)

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.2.4(B), states: "When physical measurement of the isotopics is needed, the measurement is obtained by using instrumentation subject to facility quality assurance measures as specified in 10 CFR 70.22(f)." How control will be exercised over measurements of isotopic and moisture content without confirmatory measurement at the MFFF is not apparent. This information is needed to ensure that processes are adequately subcritical.

Response

1.0 Background

Principle criticality safety feed material requirements for feed material to the MFFF are as follows:

1.1. Isotopics

The Pu isotopic is a maximum of 95% Pu-239 and minimum of 5% Pu-240. Additionally, the maximum U-235 enrichment is 93.2%

1.2. Moisture

The maximum moisture in the feed material is 0.5% (per the 3013 standard).

1.3. Density

For PDCF material, the maximum density feed material requirement is 7 g/cc. For AFS material, the maximum density feed material requirement is maximum theoretical density (11.249 g/cc).

1.4. Impurities (AFS only)
For AFS material, the maximum impurities is 20% of the mass if the can net mass exceeds 2.5 kg and 28% of the mass if the can net mass is less than 2.5 kg.

1.5. U content (AFS only)
For AFS material, the maximum U/Pu value is 0.3.

2.0 <u>Approach</u>

2.1 MFFF Criticality Safety Methodology

a. Isotopics
Analysis of criticality safety is by very conservative, high assumed fissile material isotopic contents. In particular, MFFF criticality analyses are performed on an isotopic distribution of 96% Pu-239 and 93.2% U-235. While these are very high, near 100% values, maximum isotopic fissile material concentrations need to be assured by the supplier.

b. Moisture
Since the received containers are sealed, the internal moisture will not credibly change after packaging and thus the adherence to the 3013 standard of 0.5% must be assured by the supplier (DOE). However, criticality analyses for unopened cans in the MFFF have been performed on the basis of 1.0 percent (two times the 3013 standard requirement). Furthermore, after cans have been opened, bounding moisture of 3% for PDCF material (100% relative humidity value) and approximately 300% for AFS material (bounding value due to any included impurities, assumed to be salt pickup under IROFS environmental controls) and thus credible moisture content from the supplier is irrelevant after the cans are opened. Of course, after initial processing after can opening, the powder is dissolved in an aqueous solution in the electrolyzer and then the moisture content is not relevant since optimum moderation conditions are analyzed.

c. Density
Prior to can opening, no assumptions about density are made in the criticality safety analysis. That is, maximum theoretical densities are assumed. However, before the powder is sent to the electrolyzer to be dissolved, the density must be shown to be less than 7 g/cc

d. Impurities (AFS only)
Impurities, conservatively assumed to be salts, are required to be controlled to ensure maximum credible moisture values in opened AFS cans. The maximum impurities in AFS cans is based on DOE NDA measurements of fissile material contents as well as fissile material measurements performed by MFFF NDA with impurities being assumed to be the balance of the balance of the weight of the container.

e. U content (AFS only)

For most, low moisture applications (i.e., unopened cans, singly or in storage, even PuO_2 in the AP process-Electrolyzer, tanks, filters, etc), the presence of U content (rather than Pu content) has been shown to reduce unit reactivity and thus its value is of no consequence. However, in dilute applications or in AP as nitrate prior to dilution by depleted U nitrate (after which its initial U content is no longer relevant), the presence of uranium is potentially slightly (around a maximum of 1%) more reactive than the presence of plutonium, depending upon the dilution. Thus the U content must be controlled. The maximum U content will be based on DOE NDA measurements of Pu contents as well as Pu contents measurements performed by MFFF NDA with impurities being assumed to be the balance of the balance of the weight of the container.

2.2. Controls by Supplier (DOE)

a. Isotopics
Isotopic determinations by the supplier (DOE) are combinations of at least two independent measurements by TIMS and NDA, and/or other information such as documentation of maximum isotopics content.

b. Moisture
Moisture measurements to ensure that the 3013 standard requirement of maximum moisture 0.5% in the container are expected to be based on actual measurements limits on the time and temperature of the calcination furnace, and (in the case of AFS material) historical data .

c. Density
For PDCF material, the maximum density of 7 g/cc for the feed material is to be ensured by tap density measurements. For AFS material, the density of the feed material does not need to be controlled since the MFFF analysis assumes maximum theoretical density of the feed material.

d. Impurities (AFS only)
The maximum impurities is expected to be based on DOE NDA measurements of fissile material contents (Pu) with impurities being assumed to be the balance of the balance of the weight of the container.

e. U content (AFS only)
The maximum U content is expected to be based on DOE NDA measurements of Pu contents with impurities being assumed to be the balance of the balance of the weight of the container.

2.3. MFFF QA of Supplier (DOE) of MFFF Feed Powder
The minimum requirements for qualifying a supplier are a qualification audit prior to the commencement of quality activities to verify the QA program and implementation meet the requirements of 10CFR50 Appendix B using the implementation requirements of NQA-1, 1994 including 1995 addendum, Part I basic and supplementary requirements, Part II and Appendix 2A-1 of Part III. The verification will be performed to the extent that those requirements apply to the activities being performed. The activities include blending, sampling, analysis of the

powder (impurities, isotopic and physical characteristics including NDA) and control of the material after being prepared for analysis. Subsequent to initial qualification, annual evaluations of performance are required and every three years another audit is required. The annual evaluations receive input based on receipt inspections of PuO_2, technical reviews of submitted documentation and performance of material during processing at the MOX facility.

For the powder supplier (PDCF and K Area Material Storage-supplier of AFS powder), MFFF QA will perform the initial audit as stated above. If satisfactory, the respective powder supplier will be added to the approved supplier's list as a supplier of feed material (PuO2 or AFS powder). Initially there will be a restriction requiring that MOX Services perform a surveillance of the first production run of the feed material. This will afford the opportunity to observe activities and practices as applied to the feed material to be provided to MOX Services. If the surveillance is acceptable, then the restriction will be removed from the approved suppliers list. A second surveillance will be performed about six months later to verify sustained performance. If results are satisfactory, then remaining oversight will be in accordance with the minimum requirements as specified above. If unacceptable performance is observed it will be corrected and enhanced oversight will be performed commensurate with the significance of the observed weaknesses until the sustained performance can be verified.

2.4. Measurements by MFFF

Currently, MFFF relies on and credits DOE to provide doubly contingent controls on supplied material for isotopics and moisture and singly contingent controls for density (PDCF only), impurities, and U content. As noted previously, the details will be governed by an Interface Control Document (ICD) and controlled by QA requirements. However, these details are not firmly established yet. However, regardless of the DOE controls, MFFF will perform the following measurements:

a. Isotopics

The MFFF NDA equipment includes gamma spectroscopy. Every can will be measured which will be used to determine the distribution of Pu fissile material isotopic contents. Additionally, MFFF criticality safety calculations are based on 96% Pu-239 and 4% Pu-240 which is conservative to the supplier (DOE) requirements of 95% and 5%, respectively.

b. Moisture

As noted previously, since the received containers are sealed, the internal moisture will not credibly change after packaging and thus the adherence to the 3013 standard of 0.5% must be assured by the supplier (DOE). However, criticality analyses for unopened cans in the MFFF have been performed on the basis of 1.0 percent (100% increase in the 3013 standard). Furthermore, after cans have been opened, bounding moisture of 3% for PDCF material (100% relative humidity value) and approximately 300% for AFS material (bounding value due to any included impurities, assumed to be salt pickup under IROFS environmental controls.) Of course, after initial processing after can opening, the powder is dissolved in an aqueous solution in the electrolyzer and then the moisture content is not relevant since optimum moderation conditions are analyzed.

c. Density
While it has been shown that the receiving area in the MFFF, including the 3013 storage unit, is safe for any density up to maximum theoretical density (11.46 g/cc for PuO_2 or 11.249 g/cc for AFS material), the design of the dissolution unit is based on 7 g/cc. Therefore, the material must be controlled in the dissolution unit and downstream to a maximum density of 7 g/cc. For PDCF material, density of the unopened cans is measured by an X-ray device to determine the volume which along with the mass provides the density to confirm that it is less than 7 g/cc. For AFS material, redundant samples analyzed in the MFFF laboratory are used to ensure that the density is less than 7 g/cc.

d. Impurities (AFS only)
The maximum impurities will be based on MFFF NDA measurements of fissile material contents with impurities being assumed to be the balance of the balance of the weight of the container.

e. U content (AFS only)
The maximum U content is will be based on MFFF NDA measurements and/or Pu contents with impurities being assumed to be the balance of the balance of the weight of the container. Note, while the requirements of the feed material are for U/Pu of 0.3, MFFF analyses have been performed at a conservative value of U/Pu of 0.5.

3.0 Summary

The double contingency principle is applied to MFFF feed material characteristics. Feed material characteristics are based, depending upon the parameter, upon a combination of supplier (DOE) determinations as well as MFFF measurements.

With regards to the question concerning moderation control, welded, triply contained, sealed containers are used. This is documented in § 6.4.4.6 of the LA in Enclosure (3).

Section 6.4.4.5, "Reflection Control"

RAI NCS-30

Explain what conditions constitute loss of reflection control. While the discussion in this section appear to involve the presence of mainly hydrogenous materials around the boundaries of fissile material units, ISA Summary Section 5.3.7.2.37, Purification (KPA) Unit Criticality Controlled Parameter, discusses loss of reflection control as follows:

> Loss of reflection control could occur due to leak of a process material achieving an unsafe volume and greater reflector proximity to equipment in any glovebox drip tray or process cell drip tray.

Also, in ISA Summary Table 5.3.7-61, accident sequence KPA-10 is described (under the heading Volume and Reflection Events as:

> There is a leak of solution that could provide additional reflection of fissile bearing process equipment.

The leakage of fissile solutions is typically considered a loss of geometry control, and a decrease in the distance between fissile units a loss of interaction control. Reflection is normally considered an effect of non-fissile bearing materials. It is therefore necessary to explain what is meant by reflection control, versus geometry, interaction, or volume.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718 Sections 6.4.3.3.2.1 through 6.4.3.3.2.12 contain acceptance criteria for use of each of the twelve different controlled parameters. This information is needed to ensure that the criteria that must be met for each of the controlled parameters is appropriate.

Response

Reflection control is generally ensured by passive controls. For example, reflection controlled units in gloveboxes have their geometry fixed by virtue of the location of the equipment in the glovebox. In these cases, the glovebox walls prevent excessive reflection, due for example, to personnel. Additionally, in liquid units (such as in AP), the units are designed with passive IROFS piping which prevent excessive reflection, due for example, to pipe leaks of materials which could affect reflection. In those cases where leaks between piping components are credible (although unlikely to occur), engineered IROFS (drip tray level alarms, glovebox leak detectors, drains to geometrically safe tanks) are used to prevent excessive reflection. MOX Services proposes no change to the LA in response to RAI NCS-30. The ISA Summary will be changed in a future submittal to provide clarification in accordance with this response.

RAI NCS-31

Explain what is meant by sufficient water reflection in your statement: Sufficient water reflection is conservatively used in evaluations to simulate potential personnel and/or other transient reflectors. Explain why no minimum reflector conditions are specified.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". Construction Authorization Request (CAR) Section 6.3.3.2.5, stated: At a minimum, reflection conditions equivalent to 1-in (2.5-cm) tight-fitting water jacket are assumed to account for personnel and other transient incidental reflectors not evaluated in the unreflected models. NUREG-1718, Section 6.4.3.3.2.5, contains the same language. This information is needed to ensure that processes are adequately subcritical.

Response

The minimum reflector conditions as stated in the CAR have been re-included in the LA § 6.4.4.5 as documented in Enclosure (3).

RAI NCS-32

Add the phrase 12-in (30-cm) tight-fitting water jacket to your statement: In cases where reflection control is not indicated, water reflection of process stations or fissile units is represented by a tight-fitting water jacket....

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". CAR Section 6.3.3.2.5 contained the 12-in thickness, as does NUREG-1718, Section 6.4.3.3.2.5(D). This information is needed to ensure that processes are adequately subcritical.

Response

The phrase "12-in (30-cm) tight-fitting water jacket" was added to LA § 6.4.4.5. This is documented in Enclosure (3).

Section 6.4.4.6, "Moderation Control"

RAI NCS-33

Explain your statement: Moderation control is used in MFFF design applications where the process function is not compatible with a worst-case SNM moderator content (i.e., optimum moderation) or process/storage area flooding assumption. In particular:

a. Staff assumes that this means that the system will be evaluated to determine if it is subcritical under an optimum moderation or full flooding assumption; if it cannot be shown to be subcritical, then moderation controls will be established. Is this what is meant by not compatible? If this understanding is correct:

b. Explain the sense in which the word <u>or</u> is meant. Does this mean that an evaluation of both optimum and full flooding conditions must be done, or is an evaluation of either condition sufficient? Optimum moderation may not occur at full flooding conditions.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.1(F) states: "The applicant commits to assuming credible optimum conditions (i.e., most reactive conditions physically possible) for each controlled parameter unless specified controls are implemented to limit the controlled parameter to a certain range of values." This information is needed to ensure that processes are adequately subcritical.

Response

The assumption that the system will be evaluated to determine if it is subcritical under optimum moderation or full flooding assumption and if it cannot be shown to be subcritical, then moderation controls will be established is correct that this is what is meant by not compatible. The following words "or process/storage area flooding assumption" have been removed as they are misleading and not consistent with the MFFF Criticality Safety Strategy. This is documented in LA § 6.4.4.6 in Enclosure (3).

RAI NCS-34

Add the phrase ...in NCSEs and the ISA Summary to your statement When process variables can affect moderation, the SSCs or procedures that affect those process variables are identified as IROFS.

This question also applies to concentration and heterogeneity control.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.2.6(B), states: "When process variables can affect the moderation, controls to maintain the process variables are identified as IROFS in the CSEs and ISA Summary." While the license application states that these items will be identified as IROFS, it does not state where this will be documented. This information is needed for regulatory clarity.

Response

The phrase "...in the NCSEs and the ISA Summary" has been added to all the apposite sentences in the LA § 6 as documented in Enclosure (3).

RAI NCS-35

Explain your statements "the sampling program is based on compliance with the double contingency principle" and "The sampling process incorporates independent verification as part of the sampling and analysis program." Does placing this under this bullet imply that sampling will always be done to determine moderator content (when the amount of moderator is controlled)? What about the statement that consideration will be given to sampling analysis and verification...? Will dual independent sampling and analysis be used, or only independent verification? If not, of what does independent verification consist?

This question also applies to concentration control (independent verification sampling methods).

10 CFR 70.64(a)(9) states: "The design must provide for criticality control including adherence to the double contingency principle." NUREG-1718, Section 6.4.3.3.2.6(E) states: "When sampling of the moderator is needed, the sampling program uses dual independent sampling and analysis methods." The information is needed to ensure compliance with the double contingency principle.

Response

For clarification to the sampling program at the MFFF, the sentence concerning the NDA measurement systems and previous lines have been removed for regulatory clarity. The proposed change to § 6.4.4.6 of the LA is documented in Enclosure (3).

RAI NCS-36

Explain how competing fire and criticality risks will be managed in determining what fire protection systems may be used. You state "The effects of credible fire events and the consequences associated with the potential use of moderating material in mitigating such fires are evaluated, as applicable." However, it is not clear how the results of this evaluation are used.

NUREG-1718, Section 6.4.3.3.2.6(G), states The ISA may weigh the competing risks from criticality accidents and fires and determine that the overall risk to the worker and the public is minimized by allowing the use of water. CAR Section 6.3.3.2.6 stated in addition that in the MFFF moderation-controlled areas, hydrogenous fire-fighting materials are not allowed. This information is needed to ensure that both fire and criticality risks are adequately addressed.

Response

The results of the evaluation are incorporated in section 7.3.3.1 of the LA. It states "Due to nuclear criticality safety concerns, hydrogenous material (e.g. water) is not used as a suppression agent in process rooms and in areas that contain nuclear material." The reference is included in § 6.4.4.6 of the LA and documented in Enclosure (3).

Section 6.4.4.7, "Concentration Control"

RAI NCS-37

Clarify that concentration-based single-parameter limits are based on conservative (full) reflection in addition to conservative (spherical) geometry. Single-parameter limits are generally determined assuming all other parameters are at their worst-case values.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.1(F), states: "The applicant commits to assuming credible optimum conditions (i.e., most reactive conditions physically possible) for each controlled parameter unless specified controls are implemented to limit the controlled parameter to a certain range of values." For single-parameter limits, this means that all other parameters are evaluated at their worst-case credible conditions. This information is needed to ensure that processes are adequately subcritical.

Response

Single-parameter limits for concentration are conservatively bound, taking into consideration conservative geometry and conservative reflection assumptions. Clarification was made in LA § 6.4.4.7 concerning RAI NCS-37 and documented in Enclosure (3).

RAI NCS-38

Add the phrase so that a single operator cannot defeat the control mechanism to your statement When using a tank containing concentration-controlled solution, access to the tank is controlled.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.2.7(C), states: "When using a tank containing concentration-controlled solution, the tank is normally closed and locked. Access should be controlled to ensure that a single operator cannot defeat the control mechanism." This information is needed to ensure that processes are adequately subcritical.

Response

The phrase "so that a single operator cannot defeat the control mechanism" was added to LA § 6.4.4.7 in response to RAI NCS-38. This is documented in Enclosure (3).

RAI NCS-39

State whether sampling alone may be relied on to meet the double contingency principle when concentration control is the only means of ensuring subcriticality in unfavorable geometry equipment. If so, justify reliance on sampling alone and clarify requirements for sampling (e.g., dual independent sampling and analysis, independent verification).

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.2.7(E) states: In such cases, due to the difficulties involved with dual sampling, another means (such as an in-line monitor) should be used in conjunction with dual sampling to provide reasonable assurance of safety. This information is needed to ensure that processes are adequately subcritical.

Response

When sampling alone is relied on the meet the double contingency principle, reliance is based on dual independent sampling using independent verification. This clarification is made in LA § 6.4.4.7 and documented in Enclosure (3).

RAI NCS-40

Describe what surveillance will be performed to verify that over-concentration has not occurred (in concentration-controlled processes).

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.2.7(G), discusses identification and control of mechanisms leading to over-concentration (as in your application), but also states Surveillance is provided to ensure the effectiveness of these controls. This information is needed to ensure that processes are adequately subcritical.

Response

Documented in the NCSEs for applicable units, long term accumulation monitoring detectors will be incorporated into the MFFF. These detectors provide surveillance to verify that over-concentration does not occur. This clarification provided in LA § 6.4.4.7 is documented in Enclosure (3).

Section 6.4.4.8, "Interaction Control"

RAI NCS-41

Describe whether passive engineered devices for interaction control will be periodically inspected for deformation.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.2.8(A), states, in addition to the commitments in your application, that if the engineered devices are part of the structure

of the unit (such as moveable birdcage drums) or subjected to significant mechanical stresses, they should be periodically inspected for deformation. This information is needed to ensure that processes are adequately subcritical.

Response

Other than passive structural elements, passive engineered devices are not used for interaction control. Passive structural elements are verified during startup and if deformation is credible, periodic inspections will be incorporated into the NCS program. The commitment to NCS assessment, surveillance, and walk-downs is discussed in LA § 6.2.3 and documented in Enclosure (3). Clarification has been added to LA § 6.4.4.8 to the use of passive engineered features for interaction control.

RAI NCS-42

Describe whether visual indicators and/or postings are used when interaction is procedurally controlled.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.2.8(B), states, "Unit spacing is controlled by rigorous procedures (if the spacing is identified in workstation procedures with visual indicators and postings). This should include visible guides (such as painted lines and postings). This information is needed to ensure that processes are adequately subcritical.

Response

Interaction control is only used in waste units (VDR, VDQ, and VDT) and for hand carry. Interaction control will be implemented by procedures. In these situations; visual indicators and/or postings will be used. This clarification is provided in LA § 6.4.4.8 and documented in Enclosure (3).

Section 6.4.4.10, "Volume Control"

RAI NCS-43

Specify either to what percentage of the minimum critical volume volume will be limited, or how this percentage will be determined.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.2.10(C), includes this acceptance criterion, but a verbatim restatement of the criterion is not enough to demonstrate adequacy. When it states that "some percentage should be used, either that percentage or the method of determining it should be described. This information is needed to ensure that processes are adequately subcritical.

Response

MOX Services rarely uses volume control. If it is used it is based on a calculation performed at the USL for spherical geometry. For that reason the ambiguity of the statement that "volume is limited to a percentage of the minimum critical volume" was removed. This change is reflected in LA § 6.4.4.10 and documented in Enclosure (3).

Section 6.4.4.11, "Heterogeneity Control"

RAI NCS-44

Commit that assumptions about the physical scale of heterogeneity are based on the observed physical characteristics of the material, and appropriately controlled, and that modeled conditions are at least as reactive as suggested by the physical data.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.2.11(C), states: "Assumptions about the physical scale of heterogeneity (as used in computer calculations) are based on the observed physical characteristics of the material, and appropriately controlled." This information is needed to ensure that processes are adequately subcritical.

Response

A commitment on assumptions about the physical scale of heterogeneity as requested in the RAI has been implemented in the LA § 6.4.4.11. This is documented in Enclosure (3).

Section 6.4.4.12, "Physicochemical Control"

RAI NCS-45

Commit that when process variables can affect the physicochemical form, controls to maintain it are identified as IROFS in the NCSEs and ISA Summary.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". While there is no corresponding acceptance criterion in NUREG-1520, there is a similar criterion for other parameters and the same principle should hold. This information is needed to ensure that processes are adequately subcritical.

Response

A commitment that when process variables can affect the physicochemical form, controls to maintain it are identified as IROFS in the NCSEs and ISA Summary has been added to the LA § 6.4.4.12. This is documented in Enclosure (3).

Section 6.4.4.13, "Process Variable Control"

RAI NCS-46

Provide examples of the type(s) of conditions that may be considered process variable control. The license application does not describe what process conditions may be credited or what bounding normal operational tolerances on process parameters and upset conditions means.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1520, Section 6.4.3.3.2.12, describes examples such as furnace temperature credited in excluding moderation, mechanical forces credited in limiting density, and the effect of background radiation on mass measurement. Indicate if this is consistent with your understanding of what process variable control is. This information is needed for regulatory clarity.

Response

MOX Services does not currently use process variable control for criticality safety. Subsequently, process variable controls are not identified as IROFS. For clarification, a change was made in the LA § 6.4.4.13 to reflect the MFFF criticality safety strategy. This is documented in Enclosure (3).

Section 6.4.5.2, "Regulatory Requirements, Guidance, and Industry Standards"

RAI NCS-47

Clarify your commitment to only use validated computer methods, and the document the results in a validation report. The license application has the unusual verbiage Industry standards note that a validation report is developed... without explicitly stating that only validated methods will be used and the results documented in a report

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.1(C) states: "the applicant has, at the facility, a documented, reviewed, and approved validation report (by NCS and management) for each methodology that will be used to make an NCS determination". If a commitment to this effect is intended, it is not clear. This information is needed for regulatory clarity and to ensure that processes are adequately subcritical.

Response

Clarification has been provided in § 6.4.5.1 of the LA as requested that only validated methods with a validation report are used for each methodology used to make an NCS determination. This change is documented in Enclosure (3).

RAI NCS-48

Clarify for what types of methods validation is required. The license application does not specify this information.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.1(C), states that validation will be done for Aeach methodology that will be used to make an NCS determination (e.g., experimental data, reference books, hand calculations, deterministic computer codes, probabilistic computer codes). This information is needed to ensure that processes are adequately subcritical.

Response

In § 6.4.5.1 of the LA, clarification was added to describe for what types of methods validation is required. These validation reports are required for computer codes, according to project procedures. (MOX Services only used validated computer codes in the determination of criticality safety. Upon review of "hand carry" evaluations, these also are based upon validated computer codes.) This change is documented in Enclosure (3).

Section 6.4.5.3, "Criticality Code Validation Methodology"

RAI NCS-49

State what information must be described in the documented, reviewed, and approved validation report.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.1(c), states that the validation report should contain the information in paragraphs (i) through (vii). Specifically, state that the documented, reviewed, and approved validation report should contain the information in paragraphs (i), (iii), (iv), (v), and (vi) of this section. (Note that the area(s) of applicability and the software/hardware to be used are described in the license application, so that items (ii) and (vii) do not need to be addressed.) This information is needed to ensure that processes are adequately subcritical.

Response

In § 6.4.5.3 of the LA as documented in Enclosure (3), the criteria for the validation report was added. These commitments are consistent with the requirements listed in NUREG 1718.

Section 6.4.5.5, "Summary of USL for Each Area(s) of Applicability (AOA)"

RAI NCS-50

With regard to Table 6.4-1 of the license application, revise the descriptions of AOA(1), AOA(4), and AOA(5) to what was reviewed and approved at the CAR stage (NUREG-1821, Tables 6.1-1, 6.1-5, and 6.1-6, or justify why the AOAs described in Table 6.4-1 of the license application are broader than what was reviewed and approved. Specifically, Table 6.4-1 does not describe limitations on the thickness and composition of cadmium and borated concrete in AOA(1) and AOA(5), and does not describe restrictions in H/(U+Pu) and EALF for AOA(4) due to outstanding questions about the applicability of certain benchmark experiments.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". NUREG-1718, Section 6.4.3.3.1(C), states: "The validation report should contain the following...(ii) A description of the AOA that identifies the range of values for which valid results have been obtained for the parameters used in the methodology." The validation was reviewed in detail during the CAR review. If any changes are needed for the current licensing review, they should be justified. This information is needed to ensure that processes are adequately subcritical.

Response

The changes to the (AOA)s of the upper safety limits as documented in Table 6.4-1 of the LA in Enclosure (3) are consistent with the AOA key parameters and definitions that were approved at the stage of the CAR. Details have been provided to mitigate any confusion about differences in the (AOA)s reviewed during the CAR submittal and clarify that the (AOA)s used in the criticality safety strategy of the MFFF are consistent with what was approved at the time of the CAR submittal.

Section 6.4.6, "Implementation of NCS in the ISA"

RAI NCS-51

Explain the following statement: "Where practical, nuclear criticality is precluded by demonstrating that the design is subcritical without the need to implement controls."

Provide an example of this, and explain why those design features that maintain subcritical conditions need not be identified as controls.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". 10 CFR 70.61(e) states: "Each engineered or administrative control or control system necessary to comply with paragraphs (b), (c), or (d) of this section shall be designated as an item relied on for safety." This statement would appear to contradict the requirements of 10 CFR 70.61(e). This information is needed to ensure that controls relied on to meet the performance requirements are identified as IROFS.

Response

For clarification, controls mentioned here are "active engineered or administrative" controls. This is consistent with the commitment to a preferential hierarchy of controls to prevent criticality. This change is documented in § 6.4.6 of the LA in Enclosure (3).

RAI NCS-52

Explain what is meant by the following two statements: (1) "In those cases in which it is not possible to demonstrate that a criticality is not credible, criticality control parameters are selected and limits on these parameters are established." (2) "Passive engineered, active engineered, and administrative criticality safety controls relied on to meet double contingency ensure that a criticality cannot occur under credible conditions." The first statement seems to suggest that criticality controls are only established when criticality cannot be shown to be "not credible," while the second seems to suggest that it is the criticality controls that make criticality "not credible." These statements appear to be inconsistent.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". 10 CFR 70.61(e) states: "Each engineered or administrative control or control system necessary to comply with paragraphs (b), (c), or (d) of this section shall be designated as an item relied on for safety." The two statements appear to be inconsistent, and the second appears to contradict 10 CFR 70.61(e). This information is needed to ensure that controls relied on to meet the performance requirements are identified as IROFS.

Response

For clarification, the phrase "ensure that a criticality cannot occur under credible conditions" has been changed to "and to demonstrate that a criticality is highly unlikely." This change has been made to make the two sentences consistent with each other and the implemented criticality safety strategy of the MFFF. This change is documented in § 6.4.6 of the LA in Enclosure (3).

RAI NCS-53

Clarify the following statement: "MFFF design and safety features are NCS calculations and NCSEs that are documented, controlled, and maintained by implementing the management measures described in Chapter 15."

Design and safety features are normally understood to encompass structures, systems, and components, relied on for safety, and not calculations or analyses. The usage of this term in this sentence appears to be inappropriate. This information is needed for regulatory clarity.

Response

For clarification, the words "evaluated in" were added before "NCS." This is documented in LA § 6.4.6 in Enclosure (3).

Section 6.5, "Regulatory Guidance Applicability"

RAI NCS-54

Clarify whether MFFF intends to adhere to the exception from Regulatory Guide 3.71 for Section 4.3.6 of ANSI/ANS-8.1-1998, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors." LA Section 6.5 does not mention this exception in its clarification for ANSI/ANS-8.1. Also, clarify whether MFFF intends to adhere to the 1998 version of ANSI/ANS-8.1 as specified in Regulatory Guide 3.71.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". Regulatory Guide 3.71 states the following exception for ANSI/ANS-8.1-1998: an "applicant should provide the details of validation (as stated in Section 4.3.6 of the standard) to (1) demonstrate the adequacy of the margins of subcriticality relative to the bias and criticality parameters, (2) demonstrate that the calculations embrace the range of variables to which the method will be applied, and (3) demonstrate the trends in the bias upon which the licensee or applicant will base the extension of the area of applicability. In addition, the details of validation should state computer codes used, operations, recipes for choosing code options (where applicable), cross-section sets, and any numerical parameters necessary to describe the input." If a commitment to this effect is intended, it is not clear. This information is needed for regulatory clarity and to ensure that processes are adequately subcritical.

Response

RG 3.71 as referred to in RAI NCS-54, noted in the License Application, is adhered to including the exceptions. MFFF intends to adhere to the 1998 version of ANSI/ANS-8.1 as specified in RG 3.71. The applicable portion of the MFFF License Application Section 6.5 will be revised as noted in Enclosure (3).

RAI NCS-55

Clarify whether MFFF intends to adhere to the exceptions from Regulatory Guide 3.71 for ANSI/ANS-8.3-1997, "Criticality Accident Alarm System." LA Section 6.5 does not explicitly mention this exception in its clarification for ANSI/ANS-8.3 (i.e., LA Section 6.5 states that MFFF operations comply with the "corresponding guidance in Regulatory Guide 3.71."

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". Regulatory Guide 3.71 states the following with regard to ANSI/ANS-8.3-1997: "The guidance on criticality accident alarm systems, as specified in ANSI/ANS-8.3-1997 (reaffirmed in 2003), is generally acceptable to the NRC staff. An exception is that 10 CFR 70.24, "Criticality Accident Requirements," requires criticality alarm systems in each area in which special nuclear material is handled, used, or stored, whereas Section 4.2.1 of the standard merely requires an evaluation for such areas. Another exception is that 10 CFR 70.24 and 10 CFR 76.89, "Criticality Accident Requirements," require that each area must be covered by two detectors, whereas Section 4.4.1 of the standard permits coverage by a single reliable detector. Finally, 10 CFR 70.24 and 10 CFR 76.89 require a monitoring system capable of detecting a nuclear criticality that produces an absorbed dose in soft tissue of 20 rads of combined neutron and gamma radiation at an unshielded distance of 2 meters from the reacting material within 1 minute." If a commitment to this effect is intended for these exceptions, it is not clear. This information is needed for regulatory clarity and to ensure that processes are adequately subcritical.

Response

Concerning the exception in R.G. 3.71 about Criticality Accident Alarm Systems has been addressed as follows. The design places detectors to monitor a criticality event in areas where 450 grams of Pu-239 is handled, used or stored in compliance with 10CFR70.24 and R.G. 3.71. Although, several places within the MFFF are inaccessible, i.e., storage vault, BAP process cells, etc, that make it impractical to locate the detector in those rooms. Detectors are placed outside of those rooms with the capability of detecting the radiation consequences of a criticality event. Alarms are positioned in personnel access areas associated with each criticality event for prompt evacuation. A statement to address this has been added to the LA as can be observed in Enclosure (3).

The design covers areas potentially affected with 3 detectors with a 2 out of 3 logic for alarm. This approach to the exception in §4.4.1 of R.G 3.71 has been addressed in the LA §6.3 as seen in Enclosure (3).

Concerning the exception to 10 CFR 70.24 and 10 CFR 76.89 are as follows. Computer runs are performed to define the flux spectrum for all fissile material configurations within the MFFF to determine the most conservative case. Both neutron and photon fluxes are included in the total response at the detector. A source normalization factor is applied to adjust the total source to an absorbed dose of 20 rads in 1 minute. A gamma sensitive detector will be employed and a threshold established below the photon dose rate as determined in the analysis. A statement has been added to §6.3 of the LA for clarity as can be observed in Enclosure (3).

RAI NCS-56

Clarify whether MFFF intends to adhere to the exception from Regulatory Guide 3.71 for ANSI/ANS-8.17-1984, "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors," on burnup credit. LA Section 6.5 does not mention this exception in its clarification for ANSI/ANS-8.17.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". Regulatory Guide 3.71 states the following exception for ANSI/ANS-8.17-2004: "licensees and applicants may take credit for fuel burnup only when the amount of burnup is confirmed by physical measurements that are appropriate for each type of fuel assembly in the environment in which it is to be stored." If a commitment to this effect is intended, it is not clear. This information is needed for regulatory clarity and to ensure that processes are adequately subcritical.

Response

MOX Services intends to adhere to ANSI/ANS-8.17-1984 and the exception noted in Regulatory Guide 3.71 which states that licensees and applicants may take credit for fuel burnup only when the amount of burnup is confirmed by physical measurements that are appropriate for each type of fuel assembly in the environment in which it is to be stored. The applicable portion of the MFFF License Application Section 6.5 will be revised as noted in Enclosure (3).

RAI NCS-57

Explain the reason for the additional clarification for ANSI/ANS-8.17, which is stated in LA Section 6.5 as the following: "This commitment is considered applicable to process, material handling, or storage area designs where a criticality event has been determined to be credible."

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". This information is needed for regulatory clarity and to ensure that processes are adequately subcritical.

Response

These words have been removed from LA § 6.5 as they are unnecessary. This is documented in Enclosure (3).

RAI NCS-58

Clarify whether MFFF intends to adhere to the version of the standards endorsed in Regulatory Guide 3.71. For example, in LA Section 6.5, MFFF commits to ANSI/ANS-8.7-1975, "Nuclear Criticality Safety in the Storage of Fissile Materials." However, ANSI/ANS-8.7-1998 is the version of the standard endorsed in Regulatory Guide 3.71. Also, MFFF commits to ANSI/ANS-8.19-1996, "Administrative Practices for Nuclear Criticality Safety." However, ANSI/ANS-8.19-2005 is the version of the standard endorsed in Regulatory Guide 3.71. As an additional example, MFFF commits to ANSI/ANS-8.17-1984, "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors." However, ANSI/ANS-8.17-2004 is the version of the standard endorsed in Regulatory Guide 3.71. Finally, as stated previously, ANSI/ANS-8.1-1998 is the version of the standard endorsed in Regulatory Guide 3.71, but MFFF commits to the 1983 version of the standard.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". ANSI/ANS-8.1, ANSI/ANS-8.7, ANSI/ANS-8.17, and ANSI/ANS-8.19 have all been revised with additional requirements that have been endorsed by Regulatory Guide 3.71. However, MFFF commits to earlier versions of these standards that do not have these additional requirements. This information is needed for regulatory clarity and to ensure that processes are adequately subcritical.

Response

The applicable portions of the MFFF License Application will be revised to state the following revision levels of each referenced ANSI/ANS standard to be consistent with R.G. 3.71, Revision 1, October 2005:
8.1-1998
8.3-1997 (R2003)
8.7-1998
8.9-1987 (R1995)
8.10-1983 (R2005)
8.12-1987 (R2002)
8.15-1981 (R1995)
8.17-2004
8.19-2005
8.20-1991 (R1999)
8.21-1995 (R2001)
8.22-1997
8.23-1997

This change is implemented in the LA and documented in Enclosure (3). How MOX Services adheres to each standard is stated in the LA, section 6.5.

RAI NCS-59

Clarify the statement: "may be used if the need arises," which was used in the clarifications for ANSI/ANS-8.7 and ANSI/ANS-8.12 in LA Section 6.5, since it implies that should operations that involve these standards be implemented at a later date, there is no commitment to follow these standards.

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". It is unclear whether MFFF has committed to ANSI/ANS-8.7 and ANSI/ANS-8.12. This information is needed for regulatory clarity and to ensure that processes are adequately subcritical.

Response

These words have been removed from LA § 6.5 as they are unnecessary. This is documented in Enclosure (3).

RAI NCS-60

Explain the reason for the exception to Section 4.1.7 of ANSI/ANS-8.22, which is stated in LA Section 6.5 as the following: "The design of MFFF fissile material storage areas has been reviewed, and administrative controls limiting the introduction of combustible materials during operation applied to ensure that an acceptable combustible loading is maintained. Fire protection provisions (i.e., fire suppression) in areas where fissile material is processed, handled, or stored are documented and justified."

10 CFR 70.61(d) states: "the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical". This information is needed for regulatory clarity and to ensure that processes are adequately subcritical.

Response

These words have been removed from LA § 6.5 as they are unnecessary. This is documented in Enclosure (3).

Enclosure 2
Additional Proposed Changes to MFFF License Application, § 6

In addition to the changes mentioned in response to the NRC's Request for Additional Information, the following proposed changes are included in Section 6 of the License Application.

Section 6.2.3 – The title of the section has been renamed to "NCS Surveillance and Walk-downs." This change was made to more accurately reflect the content of the section.

Section 6.4.4 – The Section has been renamed to remove the word 'mode' and the word 'mode' has been removed form the rest of the section. This is to accurately reflect the controls and mitigate any potential confusion over the definition of 'mode.'

Enclosure 3

MFFF License Application, § 6

Proposed Page Changes

6. NUCLEAR CRITICALITY SAFETY

As described in this chapter, nuclear criticality safety (NCS) practices for the Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF) are in accordance with U.S. Nuclear Regulatory Commission (NRC) regulations. The regulations for NCS are found in Title 10 of the Code of Federal Regulations (CFR) Part 70. In addition, MFFF practices for NCS draw, as needed, from guidance contained in Regulatory Guide 3.71, *Nuclear Criticality Safety Standards for Fuels and Materials Facilities,* Revision 1, October 2005 including the exceptions noted to American National Standards Institute (ANSI) and American Nuclear Society (ANS) ANSI/ANS 8 national standards.

6.1 ORGANIZATION AND ADMINISTRATION FOR NCS

The MFFF NCS program fosters ownership of nuclear criticality safety by the MFFF organization. The NCS program requires personnel to report defective NCS conditions to the manager of the regulatory function, directly or through a designated supervisor, and requires that the MFFF staff or management take no further action not specified by approved written procedure, until the NCS function has analyzed the situation.

The NCS organization, which reports to the manager of the support services function, is responsible for implementing applicable NCS practices for the MFFF. The NCS organization is independent of operations to the extent practical.

The NCS organization is responsible for implementing NCS practices of ANSI/ANS-8.1-1998, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*. The MFFF also implements the administrative practices for nuclear critical safety, as described in ANSI/ANS-8.19-2005, *Administrative Practices for Nuclear Criticality Safety*. The manager of the regulatory function and other key management functions are described in Chapter 4.

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Deleted: 1996

The NCS organization is administratively independent of production responsibilities, and has the authority and responsibility to shut down potentially unsafe MFFF operations. Specific responsibilities of the NCS organization are to:

- Establish the NCS program, including design criteria, procedures, and training
- Provide NCS support for integrated safety analyses and configuration control
- Assess normal and credible abnormal conditions
- Determine criticality safety limits for controlled parameters
- Develop and validate methods to support nuclear criticality safety evaluations (NCSEs)
- Perform criticality safety calculations and prepare NCSEs
- Review and approve proposed changes in process conditions or equipment involving fissionable material as part of the MFFF configuration management and design change process to determine whether the facility changes require prior NRC approval in accordance with the criteria of 10 CFR §70.72, *Facility Change Process*
- Specify NCS control requirements and functionality

- Review and approve MFFF operations and operating procedures that involve fissionable material
- Support emergency response planning and events
- Assess the effectiveness of the NCS program through the audit/assessment program
- Identify NCS posting requirements that provide administrative controls for operators in applicable work areas
- Maintain NCS programs for the MFFF in accordance with applicable regulatory guides and industry standards
- Be the single point of contact for nuclear criticality issues with internal and external groups or agencies, coordinating with and taking direction from the manager of the regulatory function.

The NCS organization is also responsible for the NCS function for analysis and corrective action. The nuclear criticality process requires that upon identification of a defective NCS condition, the MFFF organization take no further action not specified by approved written procedures, until the NCS function has analyzed the situation. The NCS organization shall be staffed by qualified engineers or technical staff with experience at nuclear facilities involving special nuclear material (SNM).

The manager of the NCS function has the authority and responsibility to assign and direct activities for the NCS function. The minimum qualifications for the manager of the NCS function are a Bachelor's degree in science or engineering, or equivalent, with at least three years of nuclear industry experience in criticality safety. The manager of the NCS function has management or technical experience in the application and/or direction of criticality safety programs for nuclear facilities involving SNM.

A senior NCS engineer has the authority and responsibility to conduct activities assigned to the criticality safety function, as directed by the manager of the NCS function. The minimum qualifications for a senior NCS engineer are a Bachelor's degree in science or engineering, or equivalent, with at least three years of nuclear industry experience in criticality safety.

An NCS engineer has the authority and responsibility to conduct activities assigned to the criticality safety function, with the exception of independent verification of NCSEs. The minimum qualifications for an NCS engineer are a Bachelor's degree in science or engineering, or equivalent, with at least one year of nuclear industry experience in criticality safety.

See Chapter 4 for discussion of equivalent relevant work experience that may be substituted for educational Bachelor's degree requirements.

6.2 MANAGEMENT MEASURES FOR NCS

The management practices for MFFF NCS are based on ANSI/ANS-8.1-1998, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, which provides guidance on administration, technical practices, validation of calculational methods, and on various acceptable limits for fissile nuclides. MFFF NCS management practices are

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implemented in Shaw AREVA MOX Services, LLC (MOX Services) procedures, and provide reasonable assurance that NCS-related items relied on for safety (IROFS) are available and reliable to perform their designated safety functions when needed. Chapter 15 describes the MFFF management measures implemented to supplement IROFS, including training, audits and assessments, and procedures.

6.2.1 Nuclear Criticality Safety Training

The NCS practices and associated procedures comply with regulatory requirements and subscribe to ANSI/ANS industry standards. MOX Services endorses the training requirements of ANSI/ANS-8.19-2005, *Administrative Practices for Nuclear Criticality Safety,* and ANSI/ANS-8.20-1991 (R1999), *Nuclear Criticality Safety Training*. The training is appropriately tailored to the staff's function within the MFFF.

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In addition, the MFFF NCS staff develops:

1. NCS training that includes facility, materials, operations, methodologies, design solutions, work stations, and storage locations that provide operators with knowledge and rules to ensure MFFF maintains the nuclear safety margin
2. Instructions regarding the use of process variables for NCS control, when controls on such parameters are credited for nuclear criticality safety (e.g., IROFS)
3. Training that includes the policy to identify NCS posting requirements for administrative controls that provide operators with reference for ensuring conformance and safe operation
4. Training associated with the operation of plutonium containing systems to prevent criticality events.

NCS training is based on ANSI/ANS-8.20-1991 (R1999), *Nuclear Criticality Safety Training* and is appropriately tailored to the staff's function with the MFFF. NCS training is developed by the NCS organization and implemented in conjunction with the MFFF training function. The instructors of NCS-related material are selected by the manager of the NCS function, in cooperation and coordination with the MFFF training function. Training is on nuclear criticality topics and is performed by the criticality functional organization. The manager of the NCS function ensures that the NCS training is current and adequate and contains the required skills and knowledge, by periodically reviewing training content. Records of currently trained MFFF employees are retained in accordance with the records management program. Visitors are trained commensurate with the scope of their visit and/or are escorted by MOX Services employees who are fully trained for the scope of the visit, including the criticality safety requirements for the area(s) to be accessed.

6.2.2 Audits and Assessments

MOX Services utilizes distinct levels of activities to evaluate the effectiveness of the NCS program and other management measures to ensure that operations conform to criticality safety requirements and controls in accordance with ANSI/ANS-8.19-2005, *Administrative Practices for Nuclear Criticality Safety*. Internal or external audits, which are independently planned and

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documented evaluations, are performed by the quality assurance (QA) organization. Assessments are management directed evaluations, within their area of responsibility, to assess the adequacy, programmatic compliance, and implementation effectiveness of the NCS program and other management measures. The manager of the NCS function, or designee, is lead for NCS assessments, surveillances, and walk-downs. QA audits are consistent with MOX Project QA Plan (MPQAP) requirements. Representatives of the NCS function conduct scheduled assessments, surveillances, and/or walk-downs of applicable MFFF manufacturing and support areas in accordance with approved written procedures.

Deleted: Additionally, periodic surveillances and/or walk-downs of areas or activities involving fissile material operations are conducted.

Quality-affecting activities of the NCS program are evaluated annually by either periodic audits or assessments. As a minimum, regularly scheduled internal audits of the NCS functional area quality-affecting activities shall be performed at least once every two years. The frequency for audits of operational phase IROFS related activities will be based on the risk-informed methodology determination which will consider the safety significance of the activity, results of the Integrated Safety Analysis (ISA) and/or performance history so that each area is evaluated annually (Assessment or Audit) and audited at least once every two years. Personnel performing audits shall be independent of the direct responsibility for performing the work being audited. Written notification of a planned audit shall be provided to the functional organization at a reasonable time before the audit is to be performed.

Audit results are communicated in writing to the cognizant management of the audited function/organization. Internal management assessment results identifying findings and recommendations are communicated in writing to the cognizant management having responsibility for the area/activity evaluated and to the manager of the NCS function. Responsible management of the audited function/organization shall complete corrective action(s) including remedial action(s) and action(s) to prevent recurrence and document completion of the action(s) in a timely manner. An extent of condition will also be evaluated where appropriate for findings affecting the NCS function.

6.2.3 NCS Surveillance and Walk-downs

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Periodic walkthroughs of all areas or activities involving fissile material operations are conducted and documented weekly. The frequency for walkthroughs, if less than weekly, will be based on the risk-informed methodology determination which will consider the safety significance of the activity, results of the Integrated Safety Analysis (ISA), and/or performance history. The manager of the NCS function may utilize a risk-informed methodology determination based upon the compliance results of these evaluations, to increase or decrease the scheduled frequency of these reviews or the scope of the evaluations. The evaluations are documented (e.g., by a checklist). Identified weaknesses are incorporated into the MFFF Corrective Action Program, and are promptly and effectively resolved.

Deleted: Procedures and their implementation are reviewed periodically to ensure their continued accuracy and usefulness, and to ensure that procedures are being followed and that process conditions have not changed so as to adversely affect NCS requirements and/or controls. The reviews are conducted, in consultation with operating personnel, by MFFF staff that are knowledgeable in nuclear criticality safety. NCS assessments, surveillance, and walk-downs of the operating MFFF SNM process areas are conducted periodically.

6.2.4 NCS Procedures

Procedures are established and implemented for nuclear criticality safety in accordance with ANSI/ANS-8.19-2005, *Administrative Practices for Nuclear Criticality Safety*. NCS posting requirements at the MFFF are established that identify administrative controls applicable and

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appropriate to the activity or area. NCS procedures and postings are controlled to ensure that they are maintained current. Procedures and their implementation are reviewed periodically, but at least once every two years, to ascertain that procedures are being followed and that process conditions have not been altered to adversely affect NCS requirements and/or controls. The frequency for procedure reviews, if less than annually, will be based on the risk-informed methodology determination which will consider the safety significance of the activity, results of the Integrated Safety Analysis (ISA), and/or performance history. The reviews are conducted, in consultation with operating personnel, by MFFF staff that are knowledgeable in the nuclear criticality safety.

6.2.5 Change Management

The NCS functional organization shall review proposed changes to structures, systems and components (SSCs), hardware, software, processes and procedures to ensure that proposed facility changes are managed to maintain the integrity of the facility's safety basis and to ensure that proposed changes receive the appropriate level of NCS review. The NCS review assures that the ability of the NCS credited SSCs and/or IROFS to perform their function when needed is maintained. The NCS functional organization reviews and approves proposed changes in process conditions or equipment involving fissionable material as part of the MFFF configuration management and design change process to determine whether the facility changes require prior NRC approval in accordance with the criteria of 10 CFR §70.72, *Facility Change Process*.

6.3 NUCLEAR INCIDENT MONITORING SYSTEM

The purpose of the nuclear incident monitoring (NIM) system is to reduce risk to personnel by providing prompt warning and notification should a nuclear criticality event occur. The design and operation of the NIM system also takes into consideration the avoidance of false alarms. Alarm actuation setpoint(s) are specified with consideration of normal operating background radiation levels such that spurious actuations from sources other than criticality do not occur. The NIM system monitors MFFF areas in which 450 grams of Pu-239 is handled, used, or stored.

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In the highly unlikely event of a nuclear criticality, the NIM system is intended to:

- Monitor for excessive radiation
- Monitor appropriate areas
- Warn personnel as quickly as possible.

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The NIM system, which utilizes both fixed and portable monitoring units, is designed in accordance with generally accepted practices in R.G. 3.71, Rev 1 October 2005 and those required by 10 CFR §70.24. ANSI/ANS-8.3-1997 (R2003), *Criticality Accident Alarm System*, is the guidance document that defines the design criteria and functional operation requirements of the NIM system (or criticality accident alarm system). These features assure detection capability and prompt notification by clear audible alarm, visual light, or other notification means to warn personnel of a criticality condition. Criticality monitoring is performed by groups of detectors called "monitoring units." Each NIM system monitoring unit contains multiple gamma detectors that provide a redundant detector actuation logic thus minimizing false alarms. The design covers areas potentially affected with 3 detectors with a 2 out of 3 logic for alarm. The data from the NIM system monitoring units is sent real time to the emergency control consoles. Clearly audible alarms, visual lights, or other notification means are provided for areas that require evacuation.

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If the NIM system, detection or alarm/notification capability, becomes unavailable, the allowable number of hours during which NIM system coverage is not available is determined on a process-by-process basis. The MFFF will maintain safe operations by implementing compensatory measures (e.g., limit personnel access, halt SNM movement or activities) as necessary when the NIM system is unavailable or significantly degraded.

The evaluation of the effectiveness of NIM system detectors (detection criteria and location/spacing) takes into account the effect of existing shielding. NIM system detector coverage is determined through the use of three dimensional radiation transport codes.

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6.3.1 NIM System Principles of Operation

The NIM system is designed to detect radiation in the highly unlikely occurrence of a criticality event. The nuclear criticality audible alarm, visual light, or other notification means are provided clearly in accessible locations of the facility. Indication that a NIM system alarm condition has occurred is also sent to an emergency control console in the control room and/or a remote facility. The criticality alarm is designed to accommodate the working environment within the MFFF.

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6.3.2 NIM System Design

NIM system design features:

- Prevent spurious alarms through the use of redundant detectors and alarm actuation setpoint determination
- Produce event records through the use of the Emergency Control Consoles.

The design criteria for the NIM system are:

- **Reliability** – NIM system components do not require frequent servicing. The system is designed to reduce the effects of non-use, deterioration, power surges, and other adverse conditions. The design ensures reliable actuation of an alarm, while avoiding false alarms.
- **Seismic tolerance** – The NIM system is designed to remain operational in the event of a seismic shock equivalent to the MFFF design basis earthquake.
- **System vulnerability** – NIM system components are protected in order to reduce the potential for damage in case of fire, explosion, corrosive atmosphere, or other probable extreme conditions. The system is designed to reduce the potential of failure, including false alarms.
- **Failure warning** – The NIM system provides a visual or audible warning signal to indicate system malfunction or the loss of primary power.
- **Response time** – The NIM system produces a criticality alarm signal within one-half second of detector recognition of a criticality event.
- **Detection** – The NIM system is designed to detect the minimum event of concern. In areas where fissionable material is handled, used, or stored, the minimum event of concern is analytically determined based on the process, materials, geometry, and process equipment present in each covered area. The minimum event of concern delivers the equivalent of an absorbed dose in soft tissue of 20 rads of combined neutron and gamma radiation at an unshielded distance of 6.6 feet (2 meters), within one minute. Only the gamma component of the total is considered for detector placement.

- **Coverage** – NIM system detector coverage is designed to detect the smallest criticality event as defined above. The location and spacing of detectors are chosen to account for the effect of shielding walls.
- **Electrical power** – The NIMS components will obtain normal facility power at 117 ± 15 volts AC or 102 to 132 volts AC, and a frequency range of 57 to 61 Hz
- **Alarm Response** – The alarm system covers all areas that may result in an absorbed dose of 12 rads or greater associated with the largest criticality event consequences.
- **Staff emergency response** – The nuclear criticality accident onsite emergency planning and response for the MFFF staff follows the guidance in ANSI/ANS-8.23-1997, *Nuclear Criticality Accident Emergency Planning and Response*. (As described in Chapter 14, an emergency plan is not required to be submitted.)
- **Emergency procedure** – The MFFF staff maintains an emergency procedure, which covers the entire facility including locations where licensed SNM is handled, used, or stored, to ensure that personnel can be withdrawn to a safe area upon the actuation of the NIM system alarm notification.

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6.4 NCS TECHNICAL PRACTICES

6.4.1 Nuclear Criticality Safety Evaluations

When an MFFF component or system containing fissile materials is designed or modified that could potentially affect credible criticality sequences, an NCSE is developed or updated to determine that the entire process will be subcritical under both normal and credible abnormal conditions.

NCSEs are documented with sufficient detail and clarity to allow independent review and approval of results, and to explicitly identify the controlled nuclear and process parameters, and the associated limits on which nuclear criticality safety depends. NCSEs are only performed by qualified NCS Engineers or qualified Senior NCS Engineers. Prior to approval, NCSEs will be peer reviewed by a qualified Senior NCS Engineer or NCS Manager. The approval of NCSEs is performed in accordance with MFFF project procedures.

Deleted: , specifically PP9-8 (Technical Documents) and its primary reference, PP9-3 (Design Control). Briefly, NCSEs are reviewed by a Discipline Reviewer (who as stated is a Senior NCS Engineer), Interdiscipline Reviewers (assigned by the Lead Engineer - NCS Manager), a Design Verifier (who as stated is a Senior NCS Engineer – usually the Discipline Reviewer), the Lead Engineer - NCS Manager, and the Responsible Manager – Nuclear Safety Manager.

An evaluation is performed to determine credible event sequences and identify controls such that double contingency protection is provided. The evaluation may include criticality calculations using validated calculational methodologies to demonstrate that both normal and credible abnormal conditions are subcritical, including the required minimum margin of subcriticality. IROFS are identified in the NCSE. Features that ensure that the criticality controls identified in the NCSE are sufficiently available and reliable are provided through implementation of management measures such as: procedures, training, maintenance procedures, and surveillance. The NCSE provides documentation that demonstrates that potential credible events are highly unlikely to cause a criticality.

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6.4.2 Analytical Methodology

The double contingency principle specified in 10 CFR §70.64(a)(9) and ANSI/ANS-8.1-1998, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors* requires

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that the process incorporates sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality event can occur. NCSEs of the design of the MFFF demonstrate compliance with the double contingency principle and the adequacy of criticality controls. The NCSEs, which are part of the integrated safety analysis (ISA), identify the assumptions used in the criticality evaluations. The evaluations of the assumptions are based on realistic process conditions; conservative assumptions are analytically justified so as to demonstrate the level of conservatism added. The ISA also documents a comprehensive systematic review of MFFF hazards in Process Hazards Analysis (PrHAs), including criticality, and provides additional confirmation of the acceptability of the selected means of criticality control.

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Compliance with the double contingency principle is demonstrated by identifying two or more controls on which reliance is placed to ensure criticality safety. Controls to prevent criticality are identified according to a preferential selection. Preferential selection manifests itself as first passive engineered controls, secondly active engineered controls, and then administrative controls, where practical. Common mode failures and the potential interaction between units containing fissionable material are appropriately taken into account. In addition to providing a basis for identifying IROFS, the hazard identification and review processes documented in the ISA are used to promote defense-in-depth practices in MFFF design and layout. Defense-in-depth practices are incorporated in the MFFF.

Deleted: Examples of defense-in-depth practices are as follows: 1) Use of non-safety credited automated systems to control processes, 2) Operator supervision of automated processes, 3) Control of non-credited humidity systems in MP operations, 4) Excess nitric acid in AP, which is a neutron absorber is not credited, and 5) Optimum moderation and concentration in some moderation controlled storage areas is shown to be subcritical.

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Acceptance criteria applied in performing double contingency and criticality hazard assessments are summarized as follows:

- When applying a single control to maintain limits on two or more controlled parameters, credit is taken for a single component only, for double contingency compliance.
- No single credible event or failure will result in a criticality.
- Geometry control constitutes the preferred controlled parameter, with fixed neutron absorbers employed as necessary.
- Where practical, reliance is placed on equipment design that uses passive engineered controls, rather than on administrative controls.
- Controlled parameters are identified in the NCSE evaluations. IROFS associated with maintaining these controlled parameters are noted in the NCSE. All controls identified to prevent criticality are designated as IROFS. The criticality safety controlled parameters are transferred into appropriate operating and maintenance procedures.
- Calculations are performed to demonstrate that controlled parameters are maintained during both normal and credible abnormal conditions. For example, using IROFS, the controlled parameters are maintained in spite of abnormal conditions that may occur as a result of (non-safety system) control failures. Summaries of these calculations are provided in the NCSEs. Demonstrated in the NCSEs, it is highly unlikely that controlled parameters exceed the safety limit. In cases where controlled parameters are controlled by measurement, reliable methods that ensure representative sampling and analysis are used.
- Optimum or worst-credible conditions are assumed for parameters unless they are specifically controlled.

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6.4.3 Additional Technical Practices

A design application (system) for an MFFF unit is considered subcritical when the calculated multiplication factor for the design application (system) (ANSI/ANS-8.17 Section 5 [2004], *Criticality Safety Criteria for the Handling, Storage, and Transportation of Light Water Reactor (LWR) Fuel Outside Reactors*) is shown to be less than or equal to an established maximum allowed value that properly accounts for method bias, uncertainty, and administrative margin. An administrative margin of 0.05 is used for MFFF design applications. See Section 6.4.5 for discussion of the upper safety limit (USL) for each MFFF area of applicability (AOA).

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6.4.4 Criticality Controls

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Criticality controls are the methods of criticality safety control selected for various MFFF process stations and areas. Reliance is placed on equipment design using passive engineered controls, rather than administrative controls, where practical. Techniques for criticality control, listed in order of preference, are:

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- **Passive Engineered Controls** – Controls that employ permanent and static design features or devices to preclude inadvertent criticality. No human intervention is required, except for maintenance and inspection.
- **Active Engineered Controls** – Controls that use active hardware to sense conditions and automatically place a system in a safe state or mode. Actuation and operation of these controls do not require human intervention.
- **Enhanced Administrative Controls** – Controls that rely on human judgment, training, and actions for implementation, and employ active warning devices (audible or visual) that prompt specific human actions to occur before the process can exceed established limits.
- **Simple Administrative Controls** – Controls that rely solely on human judgment, training, and actions for implementation.

The MFFF uses controls of hierarchical preference, to the extent practical, to provide correspondingly higher reliability when assessing criticality risks and demonstrating compliance with the double contingency principle. "To the extent practical" means that the hierarchy is followed wherever practicable as determined by the process. To ensure criticality control in activities involving significant quantities of fissionable materials, one or several of the following available controls are used:

Deleted: For example, in liquid systems, the geometry of the vessels (Passive Engineered Controls) are used to ensure subcriticality. Similarly, in powder units, the geometry of the jars and the geometry of filling hoppers (Passive Engineered Controls) are used to ensure subcriticality. On the other hand, abnormal potential spills (due, for example, to equipment failure) can not be prevented by a passive control, since that is due to the failure of the control itself. Instead, redundant IROFS powder weighing systems (Active Engineered Controls) are employed to detect such conditions and automatically stop the process. Additionally, in those manual processes such as the laboratory operation, waste operations, infrequent maintenance crane operation, Administrative Controls are used to ensure safety. ¶

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- Geometry Control
- Mass Control
- Density control
- Isotopic control
- Reflection control
- Moderation control
- Concentration control
- Interaction control

- Neutron absorber control
- Volume control
- Heterogeneity control
- Physicochemical control
- Process variable control.

Geometry control constitutes the preferred control, with fixed neutron absorbers employed as necessary. Although geometry control is preferred, several methods of criticality control are employed in the aqueous polishing (AP) and MOX processing (MP) designs.

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Controlled parameters and techniques for associated criticality controls that minimize the risk of inadvertent criticality are established and justified in the NCSEs. Tolerances on controlled parameters are conservatively taken into account in establishing operating limits and controls. The potential for neutron interaction between units is evaluated to ensure that the process remains subcritical under normal and credible accident conditions. Additional controls on spacing are identified as IROFS as necessary. Sensitivity studies are performed in calculations to demonstrate that the reactivity of units employing criticality controls are subcritical under all credible conditions. MFFF management measures described in Chapter 15 are generally required to ensure double contingency compliance.

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6.4.4.1 Geometry Control

Geometry control involves the use of passive engineered devices to control worst-case geometry within ensured tolerances. Geometry limits are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods. Geometry control is used in MFFF design wherever possible, including the following design applications:

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- For storage systems containing large quantities of fissile material (for which mass or mass and moderation control is not applicable)
- For process equipment whenever the imposed geometry is compatible with the applicable process function.

When the possibility of neutron interaction with other fissile units exists, interaction control or neutron absorber control may also be indicated, in conjunction with geometry control.

Geometry control parameter limits are established and implemented as follows:

- Dimensions and nuclear properties of MFFF features relying on geometry control are subject to QA measures during design and fabrication, and are verified prior to beginning operations. The MFFF configuration management program (see Chapter 15) is used to maintain these dimensions and nuclear properties.
- Credible means of transferring fissile materials to an unfavorable geometry are identified and evaluated, and controls (i.e., IROFS) are established to ensure that such transfers are precluded. In particular, leaks from favorable-geometry process vessels are collected in favorable-geometry drip trays.
- Tolerances on nominal design dimensions are treated conservatively.
- Possible mechanisms for changes to fixed geometry are evaluated, and controls are established as necessary. Credible mechanisms that could result in component deformation or changes in geometry are identified and evaluated. Where such credible

mechanisms exist, applicable design allowances and/or the surveillance program are specified.

6.4.4.2 Mass Control

Deleted: <#>Sensitivity studies are performed in calculations to demonstrate that the reactivity of units employing geometry controls are subcritical under all credible conditions.¶

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Mass control involves the use of mass-based, single-parameter limits established on conservative geometry (i.e., spherical) and SNM form (e.g., metal, oxide, aqueous solution), unless these parameters are controlled by IROFS (i.e., implementation of another criticality control mode(s) in addition to mass control). Single-parameter limits are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods, standards, or handbooks. Mass control is used in MFFF design applications where the process function is not compatible with geometry control. Mass control is generally used in combination with moderation control (i.e., allowable mass with moderation control is higher than without moderation control). The mass is generally controlled through a process variable control (i.e., required process controls include weighing and material mass balance functions). When the possibility of neutron interaction with other fissile units exists, interaction control or neutron absorber control may also be indicated, in conjunction with mass control.

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Mass control is available as a control mode where the limitation of mass is compatible with the process function and where mass can be reliably controlled during process operations (e.g., by direct weighing and/or mass balances).

Mass control parameter limits are established and implemented as follows:

- Mass limits are derived for a material that is assumed to have a given weight percent of SNM, based on conservative assumptions. Determinations of mass are based on either (1) weighing the material and assuming the entire mass is SNM, or (2) taking physical measurements to establish the actual weight percent of SNM in the material. When process variables can affect the bounding weight percent of SNM in the mixture, the SSCs or procedures that affect the process variables are controlled as IROFS in the NCSEs and ISA Summary.

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- Theoretical densities for fissile mixtures are used, unless lower densities are ensured.

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- Reasonable batch sizes are considered:
  - When overbatching of SNM is possible, the mass of SNM in a single batch is limited so that the mass of the largest overbatch resulting from a single failure is safely subcritical, taking system uncertainties into account. Overbatching beyond double batching is considered when the unit allows additional material to be accepted, to establish the margin of safety.
  - When overbatching of SNM is not possible, the mass of SNM in a batch is limited to be safely subcritical, taking system uncertainties into account.
- Mass limits are established taking tolerances into account. The determination of minimum critical mass is based on spherical geometry, unless actual fixed geometry is controlled.

- Instrumentation used to physically measure mass is subject to QA controls.

Establishing a mass limit involves consideration of potential moderation, reflection, geometry, spacing, and material concentration. The evaluation considers normal operations and expected process upsets for determination of the actual mass limit for the system and for the definition of subsequent controls.

6.4.4.3 Density Control

Density control involves taking credit for controls on SNM density in which non-optimal SNM density characteristics are used in the performance of criticality safety design calculations. SNM density limits are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods. Density control is used in the MFFF design, where the process function is not compatible with a worst-case SNM density assumption (i.e., maximum theoretical density), and is generally used in combination with mass, geometry, and/or moderation control.

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Density control parameter limits are established and implemented as follows:

- Conservative assumptions are made about the density of the fissile material.
- Instrumentation used to physically measure density is subject to QA controls.
- When process variables can affect the density, controls to maintain the process variables are identified as IROFS in the related NCSE and ISA Summary.

6.4.4.4 Isotopic Control

Isotopic abundance control involves taking credit for established realistic or conservative assumptions regarding SNM isotopic abundance in the performance of criticality safety design calculations. Isotopic control includes both the ^{235}U/U concentration (enrichment) and the concentration of fissile and nonfissile plutonium isotopes (e.g., ^{239}Pu, ^{240}Pu, ^{241}Pu), as well as the relative abundance of plutonium to uranium. The presence of ^{240}Pu (5% to 9%) and ^{242}Pu (<0.02%) offsets the contribution from ^{241}Pu (<1%), such that their presence can be neglected for ^{239}Pu in the range from 90% to 95%, as is expected to be the case for the MFFF. This will be demonstrated in the criticality calculation to be referenced in the NCSEs. Justification will be provided in the NCSEs. SNM fissile and neutron absorption isotope abundance limits are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods.

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Isotopic control parameter limits are established and implemented as follows:

- When taking credit for isotopic mixtures (where different isotopic mixtures could coexist), controls are established to segregate clearly labeled SNM of different isotopic mixtures. This is provided by sample analysis and verification activities associated with MFFF and vendor (DOE)-supplied measurements. DOE (PDCF) and vendor data are qualified in accordance with an approved QA plan and

Deleted: In addition, the determination of isotopic content is based on compliance with the double contingency principle. Consideration is given to are audited by the MFFF QA function. MFFF will comply with the double contingency principle for isotopic content of feed material. This will be based on the isotopic information supplied by the vendor (DOE). DOE will use sample destructive analysis such as thermal ionization mass spectrometry (TIMS), nondestructive assay (NDA), and/or other information to ensure that isotopic content is consistent with the isotopic characterizations specified in the safety documentation.

- Instrumentation used to physically measure isotopics is subject to QA controls.

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6.4.4.5 Reflection Control

Reflection control involves the control of fissile unit geometry and the presence of neutron-reflecting materials in process areas to increase neutron leakage from a subcritical fissile system and thereby reduce the calculated subcritical multiplication factor for the system. Although reflection control is generally applied as a passive engineered feature (i.e., configuration of concrete walls or the construction of fixed personnel barriers), reflection control generally also requires surveillance procedures to ensure that neutron-reflecting materials are excluded from the process area, or to confirm continued efficacy of personnel barriers. Single-parameter limits for reflection are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods.

Reflection control parameter limits are established and implemented as follows:

Deleted: Reflection control is generally ensured by passive controls. For example, reflection controlled units in gloveboxes have their geometry fixed by virtue of the location of the equipment in the glovebox. In these cases, the glovebox walls prevent excessive reflection, due for example, to personnel. Additionally, in liquid units (such as in AP), the units are designed with passive IROFS piping which prevent excessive reflection, due for example, to pipe leaks. In those cases where leaks between piping components are credible (although unlikely to occur), engineered IROFS (drip tray level alarms, glovebox leak detectors, drains to geometrically safe tanks) are used to prevent excessive reflection.¶

- When determining subcritical limits for an individual unit, the wall thickness of the unit and reflecting adjacent materials of the unit are conservatively bounded by the assumed reflection conditions, leaving allowances for transient reflectors as discussed below.
- Sufficient water reflection is conservatively used in evaluations to simulate potential personnel and/or other transient reflectors. At a minimum, reflection conditions equivalent to 1-in (2.5 cm) tight-fitting water jacket are assumed to account for personnel and other transient incidental reflectors not evaluated in the unreflected models.
- In cases where loss of reflection control can lead to criticality, by itself or in conjunction with another single failure, rigid and testable barriers are established and maintained by MFFF management measures (i.e., configuration management and maintenance programs) described in Chapter 15.
- In cases where reflection control is not indicated, water reflection of process stations or fissile units is represented by a minimum of 12-in (30 cm) tight-fitting water jacket, unless consideration of other materials present in the design (e.g., concrete, carbon, or polyethylene) may be a more effective, more conservative assumption, than water.
- Conservative reflection conditions are established when evaluating the criticality safety of arrays. For example, conservative minimum distances from arrays to reflecting materials are established (e.g., concrete or water).

6.4.4.6 Moderation Control

Moderation control involves taking credit for non-optimal SNM moderator content or presence within process equipment or areas, in the performance of criticality safety design calculations. SNM moderator content limits or exclusion controls for areas are established in a manner that ensures a conservative margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods. Moderation control is used in MFFF design applications where the process function is not compatible with a worst-case SNM moderator content (i.e., optimum moderation). Moderation

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Deleted: or process/storage area flooding assumption control is generally used in combination with mass or geometry control. Moderation control sometimes requires process variable control or other surveillance activities.

Moderation control is particularly useful in situations where process capacity requirements are not satisfied using mass control alone, and where the level of moderation is easily bounded or controlled (e.g., equipment in the powder handling stations confined within gloveboxes).

Potential sources of moderation that are considered include:

- Residual humidity present in powders
- Organic additives (e.g., lubricant, poreformer) used as part of a process
- Moderating fluids (e.g., water or certain oils), which could potentially enter process stations or storage areas under normal or abnormal conditions
- Presence of polyethylene, particularly in waste handling units.

Certain moderators (e.g., humidity and organic additives) exist during normal operations. Criticality safety calculations employ assumptions or process information to account for moderators normally anticipated being present in processes (see below). Moderation control parameter limits are established and implemented as follows:

- Moderation control is implemented consistent with guidance provided in ANSI/ANS-8.22-1997, *Nuclear Criticality Safety Based on Limiting and Controlling Moderators*.
- When process variables can affect moderation, the SSCs or procedures that affect those process variables are defined as IROFS in the NCSEs and the ISA Summary.
- Physical structures credited with performing moderator exclusion functions are designed to preclude ingress of moderator.
- When sampling of moderation properties is required, the sampling program is based on compliance with the double contingency principle (i.e. dual independent sampling).
- The sampling process incorporates independent verification as part of the sampling and analysis program.
- Fire protection system design, and fire-fighting procedures and training programs are developed with appropriate restrictions placed on the use of moderating materials as stated in section 7.3.3.1. The effects of credible fire events and the consequences associated with the potential use of moderating material in mitigating such fires are evaluated, as applicable.
- Credible sources of moderation are identified and evaluated for potential intrusion into moderator-controlled process stations or areas, and the ingress of moderator is precluded or controlled.

Deleted: Consideration is given to sample analysis and verification activities associated with MFFF and vendor-supplied measurements. Vendor data are qualified in accordance with an approved QA plan and are audited by the MFFF QA function.

Deleted: The use of qualified NDA measurement systems is also acceptable in establishing compliance.

- The effects of varying levels of credible interstitial moderation are evaluated when considering neutron interaction between physically separated fissile units.
- Instrumentation used to physically measure moderators is subject to QA controls.
- Drains are provided to prevent water accumulation, if that accumulation could lead to unfavorable configurations of fissile material.
- Moderation control is implemented and maintained during transportation and storage by means of welded, triply contained, sealed containers.

6.4.4.7 Concentration Control

Concentration control involves the use of concentration-based single-parameter limits established based on conservative case geometry (i.e., spherical) and SNM fissile composition, unless these parameters are controlled by IROFS (i.e., implementation of another criticality control mode(s) in addition to concentration control). Concentration control is generally applied to process equipment handling solutions with low fissile material concentration. Single-parameter limits for concentration are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded), using documented and approved methods. These limits are based on conservative (full) reflection in addition to conservative (spherical) geometry. Concentration control typically includes process variable control to ensure that concentration limits are not exceeded.

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Concentration control parameter limits are established and implemented as follows:

- When process variables can affect the concentration, those process variables are defined and controlled in the NCSEs and ISA Summary.
- Concentrations of SNM in excess of controlled parameter limits are precluded.
- When using a tank containing concentration-controlled solution, access to the tank is controlled so that a single operator cannot defeat the control mechanism.
- When sampling of the concentration is specified, a program based on duel independent sampling and analysis using independent verification sampling methods using two people is implemented.

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- Concentration-controlled processes are designed and operated in a manner that ensures that possible precipitating agents are not inadvertently introduced to the process, or that the effects of precipitation are taken into account.
- Instrumentation used to physically measure concentration is subject to QA controls.
- Concentration-controlled processes are designed and operated in a manner that prevents overconcentration in excess of controlled parameter limits. Monitoring controls are implemented to detect and prevent long-term fissile material accumulation.

6.4.4.8 Interaction Control

Interaction control involves the use of spacing to limit neutron interaction between fissile units. Single-parameter limits for interaction are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods. When interaction control is employed using passive engineered features (e.g., fuel assembly storage racks), interaction control is considered equivalent to geometry control in terms of hierarchical preference.

When neutron absorbers are used to limit interaction between fissile units, neutron absorber control is indicated in lieu of interaction control.

Interaction control parameter limits are established and implemented as follows:

- When maintaining physical separation between units, passive engineered features (i.e., spacers or other passive geometrical means) are used to the extent practical. The structural integrity of such engineered features is sufficient for normal and design basis conditions. Passive engineered features used as criticality safety controls are passive structural elements designed to withstand deformation. If needed, passive interaction controls are periodically inspected for deformation.
- When unit spacing is controlled by procedure, it is demonstrated that multiple procedural violations do not by themselves lead to criticality. Visual indicators and/or posting are used where interaction is procedurally controlled.
- When evaluating the criticality safety of units in an array or pairs of arrays, the spacing limits in ANSI/ANS-8.7-1998, *Guide for Nuclear Criticality Safety in the Storage of Fissile Materials* are used, or spacing is based on validated calculational methods.

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6.4.4.9 Neutron Absorber Control

Neutron absorber control involves the use of supplemental neutron absorber features to limit subcritical multiplication of a single fissile unit (e.g., cadmium coatings and borated concrete), or to limit neutron interaction between multiple (spaced) fissile units. Single-parameter limits for neutron absorber features are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods. When using fixed neutron absorbers, MFFF design and procedural controls are implemented consistent with guidance provided in ANSI/ANS-8.21-1995 (R2001), *Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors*.

6.4.4.10 Volume Control

Volume control involves the use of volume-based single-parameter limits established based upon worst-case geometry (i.e., spherical) and SNM form (e.g., metal, oxide, aqueous solution), unless these parameters are controlled by IROFS (i.e., implementation of another criticality control mode(s) in addition to volume). Single-parameter limits are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods. When volume control is employed using passive engineered features (e.g., use of approved fixed-geometry containers), volume control is considered equivalent to geometry control in terms of hierarchical preference. When the possibility of neutron interaction with other fissile units exists, interaction control or neutron absorber control may be indicated in conjunction with volume control.

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Volume control parameter limits are established and implemented as follows:

- When using volume control, geometric devices typically are used to restrict the volume of SNM, which limits the accumulation of SNM.
- Instrumentation used to determine volume is subject to QA controls.

6.4.4.11 Heterogeneity Control

Heterogeneity control involves taking credit for the distribution of fissile material. Heterogeneity control is applied in conjunction with another control mode (e.g., mass control,

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geometry control). Single-parameter limits for heterogeneity are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods. Heterogeneity control is typically implemented through process variable control as well. Additionally, it may be important to control the lattice pitch (i.e., spacing) in a heterogeneous configuration, such as a fuel rod or for pellet fabrication.

Heterogeneity control parameter limits are established and implemented as follows:

- When process variables can affect heterogeneity, the SSCs or procedures that affect process variables and potential mechanisms affecting homogeneity or nonhomogeneity are controlled as IROFS in the NCSEs and ISA Summary.
- Computer calculations that take heterogeneity into account are appropriately validated.
- Assumptions about the physical scale of heterogeneity are based on the observed physical characteristics of the material and appropriately controlled (size of pellets, rod assemblies, etc.) and are conservatively bound. The reactivity in modeled conditions is conservatively bound as suggested by the physical data.

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6.4.4.12 Physicochemical Control

Control of physicochemical characteristics is applied to several MFFF process units where non-optimal solution chemistry or specific values for some parameters (e.g., pellet diameter) are used in the definition of the fissile media and are assumed in criticality design calculations. The physicochemical form of the fissile material is defined by:

- Its chemical composition
- The pellet diameter (if applicable)
- The rod characteristics (if applicable)
- The assembly characteristics (if applicable).

For the AP process, a conservative or realistic (based on process information) assumption concerning the chemical form of the fissile matter is made for each step of the process, taking into account not only the nominal conditions, but also possible process upsets (e.g., failure of a PuO_2 filter or unwanted soda introduction that may cause precipitates) defined based on the double contingency principle. Single-parameter limits for physicochemical characteristic control are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded) using documented and approved methods. The different chemical forms used in the criticality analyses are:

- PuO_2
- $Pu(NO_3)_4$
- $Pu(NO_3)_3$
- Plutonium oxalate.

In the MP process, no chemical transformations take place. As a consequence, the oxide form of the fissile medium (PuO_2 and/or UO_2) is assumed.

When process variables can affect the physicochemical form, controls to maintain it are identified as IROFS in the NCSEs and ISA Summary.

6.4.4.13 Process Variable Control

MOX Services does not currently use process variable control for criticality safety except for those situations when process variables are monitored and can affect one of the other twelve parameters.

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parameter safety limit is not exceeded. The use of management measures discussed in Chapter 15 is required to ensure double contingency compliance.

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6.4.5 Margin of Subcriticality and Double Contingency Principle

To develop the USL for each of the AOAs, accepted industry codes such as SCALE code packages using an accepted cross-section library (e.g., CSAS26 (KENOVI) sequence and the 238 energy group cross-section library 238GROUPNDF5) are used. (Other computation code systems may be used if they are qualified in accordance with the MPQAP.)

6.4.5.1 Regulatory Requirements, Guidance, and Industry Standards

Title 10 CFR §70.61(d) requires that "under normal and credible abnormal conditions, nuclear processes are subcritical, including use of an approved margin of subcriticality for safety." To comply with this requirement, an industry-accepted standard practice is used (i.e., ANSI/ANS-8.1-1998). According to industry standards, a validation report for computer codes is developed that describes the development of the USL, including (1) demonstrating the adequacy of the margin of subcriticality for safety by assuring that the margin is relatively large compared to the uncertainty in the calculated value of k_{eff}, and (2) determining the AOAs and use of the code within the AOA, including justification for extending the AOA by using trends in the bias. Only these validated methods with the corresponding validation reports are used for each methodology used to make an NCS determination.

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6.4.5.2 Calculational Method

The SCALE code package is the computational system used for MFFF criticality analyses. (Other computation code systems may be used if they meet the requirements of the MPQAP.) This code package is available from the Radiation Safety Information Computational Center.

SCALE is a collection of modules designed to perform nuclear criticality, shielding, and thermal calculations. Each SCALE functional module may be run individually, or a sequence of functional modules may be executed using a special module referred to as a control module. For criticality analyses, various criticality safety analysis sequence (CSAS) control modules are available. The CSAS control modules differ in the specific functional modules executed and in the processing of cross sections used as input. As a practice, MFFF criticality analyses are performed using approved and industry-accepted control module and cross-section libraries. The calculation of k_{eff} is performed using the KENO VI Monte Carlo transport code.

6.4.5.3 Criticality Code Validation Methodology

To establish that a system or process is subcritical under normal and credible abnormal conditions, it is necessary to establish acceptable subcritical limits for the operation, and then show that the proposed operation will not exceed such subcritical limit. Software, meeting the requirements of the MPQAP, is used to determine the USL for each of the AOAs. Each documented, reviewed, and approved methodology validation report is incorporated into the configuration management program. Each report includes the following:

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- A description of the theory of the methodology including the validity of assumptions and independent duplication of results.

- A description of the use of pertinent computer codes, assumptions, and techniques in the methodology.

- A description of the verification of the proper functioning of the mathematical operations in the methodology.

- A description of the benchmark experiments and data derived therefrom that were used for validating the methodology.

- A description of the bias, uncertainty in the bias, uncertainty in the methodology, uncertainty in the data, uncertainty in the benchmark experiments, and Margin of Subcriticality for Safety, as well as the basis for these items.

The criticality code validation methodology is divided into four steps:

- Identify general MFFF design applications. The MFFF design applications and key parameters are associated with normal and design abnormal conditions.
- Select applicable benchmark experiments and group them into AOAs.
- Model the criticality experiments and calculate k_{eff} values of selected critical benchmark experiments.
- Perform statistical analysis of results to determine computational bias and the USL.

There are several substeps associated with selecting and grouping benchmark experiments. First, based on the key parameters, the AOA and expected range of the key parameter are identified. ANSI/ANS-8.1-1998 defines the AOA as "The range of material composition and geometric arrangements within which the bias of a calculational method is established." AOAs covering plutonium (Pu) and MOX applications are as follows: (1) Pu-nitrate solutions; (2) MOX pellets, fuel rods, and fuel assemblies; (3) PuO_2 powders; (4) MOX powders; and (5) aqueous solutions of Pu compounds. After identifying the AOAs, a set of critical benchmark experiments is selected. Benchmark experiments for the AOAs are selected from industry-accepted data.

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6.4.5.4 Determination of Bias

ANSI/ANS-8.1-1998, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors* requires a determination of the calculational bias by "correlating the results of critical and exponential experiments with results obtained for these same systems by the calculational method being validated." The correlation must be sufficient to determine if major changes in the bias can occur over the range of variables in the operation being analyzed. The standard permits the use of trends in the bias to justify extension of the area of applicability of the method outside the range of experimental conditions.

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The recommended approach for establishing subcriticality based on numerical calculations of the neutron multiplication factor is prescribed in Section 5.1 of ANSI/ANS-8.17-2004, *Criticality Safety Criteria for the Handling, Storage, and Transportation of Light Water Reactor (LWR) Fuel Outside Reactors*. The criteria to establish subcriticality requires that for a design application (system) to be considered subcritical, the calculated multiplication factor for the system, k_s, is noted to be less than or equal to an established maximum allowed multiplication factor, based on benchmark calculations and uncertainty terms. That is:

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$$k_s \le k_c - \Delta k_s - \Delta k_c - \Delta k_m \qquad \text{(Eq. 6.4.5.4-1)}$$

where:

k_s	=	the calculated allowable maximum multiplication factor, (k_{eff}) of the design application (system)
k_c	=	the mean k_{eff} value resulting from the calculation of benchmark critical experiments using a specific calculation method and data
Δk_s	=	the uncertainty in the value of k_s
Δk_c	=	the uncertainty in the value of k_c
Δk_m	=	the administrative margin.

Sources of uncertainty that determine Δk_s include:

- Statistical and/or convergence uncertainties
- Material and fabrication tolerances
- Limitations in the geometric and/or material representations used.

Sources of uncertainty that determine Δk_c include:

- Uncertainties in critical experiments
- Statistical and/or convergence uncertainties in the computation
- Extrapolation outside the range of experimental data
- Limitations in the geometric and/or material representations used.

Subcriticality requires the determination of an acceptable margin, based on known biases and uncertainties. The USL is defined as the upper bound for an acceptable calculation, as follows:

$$k_s + \Delta k_s \leq USL \qquad \text{(Eq. 6.4.5.4-2)}$$

The USL takes into account bias, uncertainties, and administrative and/or statistical margins, such that the calculated configuration is subcritical with a high degree of confidence.

6.4.5.5 Summary of USL for Each AOA

The development of the USLs takes into account bias and uncertainties, as well as an administrative margin. See Section 6.4.3 for a discussion of the administrative margin used for MFFF design applications within the AOAs. The USLs are applied as the basis for each nuclear criticality evaluation performed for MFFF. Table 6.4-1 identifies the USL, the key parameters and a definition of the MFFF AOAs.

6.4.6 Implementation of NCS in the ISA

Nuclear criticality calculations are performed for potentially fissile-bearing systems. In the design process, criticality safety calculations are performed to specify requirements for the design concept. The NCSEs assess both normal operating and process upset conditions. Where practical, nuclear criticality is precluded by demonstrating that the design is subcritical without the need to implement active engineered or administrative controls. In those cases in which it is not possible to demonstrate that a criticality is not credible, criticality control parameters are selected and limits on these parameters are established. Using the results of validated calculational methodologies, NCSEs demonstrate that both normal and process upset conditions meet the required minimum margin of subcriticality, and IROFS are identified to provide double contingency protection.

The NCSE evaluates normal and credible abnormal conditions developed in the component/ system Process Hazards Analysis (PrHA). The NCSEs demonstrate compliance with the double contingency principle. Passive engineered, active engineered, and administrative criticality safety controls are relied on to meet double contingency and to demonstrate that a criticality is

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Criticality safety during design and operation is ensured for the MFFF. MFFF design and safety features are evaluated in NCS calculations and NCSEs that are documented, controlled, and maintained by implementing the management measures described in Chapter 15.

6.5 REGULATORY GUIDANCE APPLICABILITY

Regulatory Guide 3.71, *Nuclear Criticality Safety Standards for Fuels and Materials Facilities, Revision 1, October 2005* endorses specific NCS standards drafted by Subcommittee ANS-8 (Fissionable Materials Outside Reactors) of the ANS Standards Committee for these purposes. The MFFF criticality design basis includes use of ANSI/ANS standards endorsed by Regulatory Guide 3.71, *Nuclear Criticality Safety Standards for Fuels and Materials Facilities* including the exceptions noted to the ANSI/ANS Standards as described in this chapter. MFFF operations comply with the guidance ("shall" statements) and implement the appropriate recommendations ("should" statements) of the applicable ANSI/ANS standards referenced below.

ANSI/ANS-8.1-1998, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors,* is part of the design basis of MFFF processes, and fissile material handling and storage areas. The standard provides general guidance addressing administrative and technical practices, as well as single-parameter and multiparameter control limits for systems containing ^{233}U, ^{235}U, and ^{239}Pu. Of particular significance to the MFFF design, ANSI/ANS-8.1-1998 provides guidance for performing NCS analysis methodology validation. ANSI/ANS-8.1-1998 NCS practices are referenced in the NCSEs to support MFFF design and operational approach. MFFF processes and storage areas that contain plutonium, uranium, or plutonium-uranium mixtures are explicitly evaluated using validated NCS analysis methodology, in accordance with the technical practice guidance of ANSI/ANS-8.1-1998. However, criticality safety may be demonstrated by reference to ANSI/ANS-8.1-1998 single-parameter and multiparameter control limits, in lieu of analysis.

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Clarifications are noted as follows:

- Section 4.2.2: MFFF process, material handling, or storage area designs incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality event is possible. For the purposes of demonstrating compliance with this requirement, "unlikely" is defined as events or event sequences that are not expected to occur during the facility lifetime, but are considered credible.
- Section 4.2.3: MFFF process design relies on engineered features where practical, rather than administrative controls.
- Section 4.3.2: In cases where an extension in the area(s) of applicability of a NCS analysis methodology is required, the method is supplemented by other calculational methods to provide estimate of bias in the extended area(s). As an alternative, the

- extension in the area(s) of applicability may also be addressed through an increased margin of subcriticality.

ANSI/ANS-8.3-1997 (R2003), *Criticality Accident Alarm System*, is part of the design basis of MFFF process and fissile material handling and storage areas. The standard provides general guidance for the design, testing, and maintenance of criticality accident alarm systems at facilities where a criticality event may lead to excessive exposure to radiation. The scope of guidance provided in ANSI/ANS-8.3-1997 (R2003) is applicable to MFFF design and operation.

MFFF operations comply with the guidance ("shall" statements) and implement the recommendations ("should" statements) of ANSI/ANS-8.3-1997 (R2003) (and the corresponding guidance in Regulatory Guide 3.71, Revision 1, October 2005 *Nuclear Criticality Safety Standards for Fuels and Materials Facilities*).

ANSI/ANS-8.5-1996, *Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material,* is not part of the design basis, nor are Raschig Rings used at the MFFF.

ANSI/ANS-8.6-1989, *Safety in Conducting Subcriticality Neutron-Multiplication Measurements In Situ*, is not part of the design basis of the MFFF.

ANSI/ANS-8.7-1998, *Guide for Nuclear Criticality Safety in the Storage of Fissile Materials*, is not part of the design basis of the MFFF at this time.

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ANSI/ANS-8.9-1987 (R1995), *Nuclear Criticality Safety Criteria for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Materials,* has been officially withdrawn by the ANS-8 working group, but continues to be available for reference. This standard is not referenced as a basis for design of the MFFF. Intersections of process components and piping containing aqueous solutions of fissile materials are evaluated using validated NCS analysis methodology, in accordance with ANSI/ANS-8.1-1998.

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ANSI/ANS-8.10-1983 (R2005), *Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement*. MFFF NCSEs performed for each process unit or area demonstrate compliance with the double contingency principle, consistent with guidance provided in Section 4.2.2 of ANSI/ANS-8.1-1998. Therefore, this standard is not part of the design basis of the MFFF.

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ANSI/ANS-8.12-1987 (R2002), *Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors,* may be reaffirmed or withdrawn in future action by the ANS-8 working group (reference ANS-8 meeting minutes, Albuquerque, New Mexico, March 30, 2000). This standard is not part of the design basis of the MFFF at this time.

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ANSI/ANS-8.15-1981 (R1995), *Nuclear Criticality Control of Special Actinide Elements*, is not part of the MFFF criticality design basis, as it is applicable to operations with isolated units containing special actinide nuclides other than ^{233}U, ^{235}U, and ^{239}Pu. Nuclear criticality control of special actinide nuclides is evaluated using validated NCS analysis methodology, in accordance with ANSI/ANS-8.1-1998.

ANSI/ANS-8.17-2004, *Criticality Safety Criteria for the Handling, Storage, and Transportation of Light Water Reactor (LWR) Fuel Outside Reactors,* is part of the design basis of MFFF fissile material handling and storage areas. The standard provides guidance addressing general safety criteria and criteria for establishing subcriticality for handling, storage, and transportation of LWR fuel rods outside reactor cores. Of particular significance to the MFFF design, ANSI/ANS-8.17-2004 provides general guidance for combining the various biases, uncertainty, and administrative safety margin terms that are considered when performing criticality calculations to establish a final k_{eff} acceptance criterion.

MFFF operations will comply with the guidance (shall statements) and implement the recommendations (should statements) of ANSI/ANS-8.17-2004. Clarifications are noted as follows:

- Section 4.11: Fuel units and rods are handled, stored, and transported in a manner that provides a sufficient factor of safety to require at least two unlikely, independent, and concurrent changes in conditions before a criticality event is possible.
- Section 5.1: The criticality experiments used as benchmarks in computing k_c have physical compositions, configurations, and nuclear characteristics (including reflectors) similar to those of the system being evaluated.

- MOX Services intends to adhere to the exception noted in Regulatory Guide 3.71 which states that licensees and applicants may take credit for fuel burnup only when the amount of burnup is confirmed by physical measurements that are appropriate for each type of fuel assembly in the environment in which it is to be stored.

ANSI/ANS-8.19-2005, *Administrative Practices for Nuclear Criticality Safety,* is part of the design basis of MFFF processes, and fissile material handling and storage areas. This standard provides criteria for the administration of a NCS program for operations outside reactors, for which there exists a potential for criticality events. An exception is noted as follows:

- Section 10: Guidance for planned response to nuclear criticality events are addressed by ANSI/ANS-8.23-1997. Therefore, no commitment is made to satisfy the guidance or recommendations of this section.

ANSI/ANS-8.20-1991 (R1999), *Nuclear Criticality Safety Training*, is part of the design basis for MFFF operational practices. The standard provides detailed guidance for NCS training for personnel associated with (non-reactor) operations where a potential exists for criticality events.

MFFF operations will comply with the guidance ("shall" statements) and implement the recommendations ("should" statements) of ANSI/ANS-8.20-1991 (R1999). No exceptions or clarifications are noted.

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ANSI/ANS-8.21-1995 (R2001), *Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors,* is part of the MFFF design basis. The standard provides detailed guidance for use of fixed neutron absorbers in criticality control.

The MFFF will comply with the guidance of this standard ("shall" statements) and recommendations ("should" statements) to assure fixed neutron absorber material integrity and reliability to perform NCS functions. The guidance includes no recommendations that require further clarification and no exceptions are taken.

ANSI/ANS-8.22-1997, *Nuclear Criticality Safety Based on Limiting and Controlling Moderators,* is part of the MFFF design basis. The standard provides detailed guidance for limiting and controlling moderators to achieve criticality control.

- MFFF operations comply with the guidance ("shall" statements) and implement the recommendations ("should" statements) of ANSI/ANS-8.22-1997. This standard will be used as a guide and sections of it will be implemented as needed.

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ANSI/ANS-8.23-1997, *Nuclear Criticality Accident Emergency Planning and Response*, is part of the MFFF design basis (although not part of the criticality safety basis). The standard provides guidance for onsite emergency planning and response to nuclear criticality accidents.

The MFFF will comply with the guidance of this standard ("shall" statements) and recommendations ("should" statements) for guidance for onsite emergency planning and response to nuclear criticality accidents. The guidance includes no recommendations that require further clarification, and as discussed in Chapter 14, an Emergency Plan is not required to be submitted.

Tables

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Table 6.4-1. MOX MFFF Area of Applicability (AOA) and Upper Safety Limit (USL)

Area of Applicability (AOA)	AOA Key Parameters and Definition	Upper Safety Limit (USL) or Maximum K_{eff}
AOA (1)	Plutonium Nitrate Solutions • Geometry – Cylinder, slab, annular cylinders & arrays of cylinders • Reflectors – Full water, cadmium/water, & borated concrete (1) Cd limited to 0.05-cm- (0.02-in.-) thick sheet surrounding 4.5-9.5-cm (1.8-3.7-in.-) thick slab tanks (2) Borated Concrete limited to 15 cm (5.9 in.) inside and outside 7-7.5-cm- (2.76-2.95-in.-) thick annulus and separated from tank by 1.8-2-cm (0.71-0.79-in.-) gap, with the composition below: $^{10}B = 1.59 \times 10^{-3}$ atoms/b·cm (2.61×10^{22} atoms/in.^{3}) $^{11}B = 7.04 \times 10^{-3}$ atoms/b·cm (1.15×10^{23} atoms/in.^{3}) Ca = 4.65×10^{-3} atoms/b·cm (7.62×10^{22} atoms/in.^{3}) Fe = 5.01×10^{-4} atoms/b·cm (2.61×10^{21} atoms/in.^{3}) Si = 1.66×10^{-4} atoms/b·cm (2.61×10^{21} atoms/in.^{3}) H = 2.17×10^{-2} atoms/b·cm (2.61×10^{23} atoms/in.^{3}) Al = 1.96×10^{-3} atoms/b·cm (2.61×10^{22} atoms/in.^{3}) O = 3.25×10^{-2} atoms/b·cm (2.61×10^{23} atoms/in.^{3}) • Chemical Form – Plutonium nitrate solution • Pu/(U + Pu) – 100 wt% • ^{240}Pu – 4 wt% • H/Pu – 100 – 200 • gPu/l – 125 – 237 • EALF – 0.14 – 0.25 eV	0.9370
AOA (2)	MOX Pellets, Fuel Rods and Fuel Assemblies • Geometry – Heterogeneous, rectangular lattices • Reflectors – Water • Chemical Form – MOX fuel • Pu/(U + Pu) – 6.3 wt% • ^{240}Pu – 4 wt% • v^f/v^m – 1.9 - 10 • EALF – 0.1 – 0.66 eV	0.9321
AOA (3)	Plutonium Oxide Powder • Geometry – Parallelepipeds, arrays of cylinders, spheres • Reflectors – Water • Chemical Form – PuO_2 powder • Pu/(U + Pu) – 100 wt% • ^{240}Pu – 4 wt% • H/Pu – 0 - 15 • EALF – 5.0 eV – 266 keV	0.9345

Table 6.4-1. MOX MFFF Area of Applicability (AOA) and Upper Safety Limit (USL) (continued)

Area of Applicability (AOA)	AOA Key Parameters and Definition	Upper Safety Limit (USL) or Maximum K_{eff}
AOA (4)	Mixed Oxide Powder • Geometry – Parallelepipeds, spheres • Reflectors – Water, depleted uranium up to a reflector of 60 cm thickness • Chemical Form – MOX powder • Pu/(U + Pu) – 8 – 22 wt% • ^{240}Pu – 4 wt% • H/(U + Pu) – 2.77 – 2.79 • EALF – 1.49 – 43.5 eV	0.9249 * * 0.9349 + an additional nonparametric margin (NPM) of 0.01
AOA (5)	Solutions of Plutonium Compounds • Geometry – Parallelepipeds, arrays of cylinders, spheres • Reflectors – Water, cadmium, & borated concrete (1) Cd limited to 0.05-cm- (0.02-in.-) thick sheet surrounding 4.5-9.5-cm (1.8-3.7-in.-) thick slab tanks (2) Borated Concrete limited to 15 cm (5.9 in.) inside and outside 7-7.5-cm- (2.76-2.95-in.-) thick annulus and separated from tank by 1.8-2-cm (0.71-0.79-in.-) gap, with the composition below: $^{10}B = 1.59 \times 10^{-3}$ atoms/b·cm (2.61×10^{22} atoms/in.^{3}) $^{11}B = 7.04 \times 10^{-3}$ atoms/b·cm (1.15×10^{23} atoms/in.^{3}) Ca = 4.65×10^{-3} atoms/b·cm (7.62×10^{22} atoms/in.^{3}) Fe = 5.01×10^{-4} atoms/b·cm (2.61×10^{21} atoms/in.^{3}) Si = 1.66×10^{-4} atoms/b·cm (2.61×10^{21} atoms/in.^{3}) H = 2.17×10^{-2} atoms/b·cm (2.61×10^{23} atoms/in.^{3}) Al = 1.96×10^{-3} atoms/b·cm (2.61×10^{22} atoms/in.^{3}) O = 3.25×10^{-2} atoms/b·cm (2.61×10^{23} atoms/in.^{3}) • Chemical Form – PuO_2F_2 solution • Pu/(U + Pu) – 100 wt% • ^{240}Pu – 4 wt% • H/Pu – 30 – 50, 85 - 210 • EALF – 0.685 – 4900 eV, 0.135 - 0.551 eV	0.9328

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