6.0 NUCLEAR CRITICALITY SAFETY

6.1 CONDUCT OF REVIEW

This chapter of the draft Safety Evaluation Report (DSER) contains the staff's review of the nuclear criticality safety (NCS) analysis performed by the applicant in Chapter 6.0 of the Construction Authorization Request (CAR). The objective of this review is to: 1) ensure that special nuclear material (SNM) storage and processing remains subcritical under normal and credible abnormal conditions during all operations, transfers, and storage at the Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF) and 2) determine whether the principal structures, systems, and components (PSSCs) and their design bases identified by the applicant provide reasonable protection against natural phenomena and the consequences of potential accidents. The staff evaluated the information provided by the applicant for NCS by reviewing Chapter 6 of the CAR, other sections of the CAR, and supplementary information provided by the applicant. The review of NCS design bases and strategies was closely coordinated with the review of chemical process safety, the Safety Assessment of the Design Bases (see Chapter 5.0 of this DSER), and the review of other plant systems.

The staff reviewed how the NCS information in the CAR addresses or relates to the following regulations, which are the top-level criticality safety requirements to be factored into the technical practices used in designing the MFFF:

- Section 70.23(b) of 10 CFR states, as a prerequisite to construction approval, that the design bases of the PSSCs and the quality assurance program be found to provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents.

- Section 70.24 of 10 CFR requires licensees authorized to possess specified quantities of SNM to have a criticality accident alarm system (CAAS).

- Section 70.64 of 10 CFR requires that baseline design criteria (BDC) and defense-in-depth practices be incorporated into the design of new facilities. With respect to NCS, the design of new facilities must provide for criticality control, including adherence to the double contingency principle (DCP) pursuant to 70.64(a)(9).

The review for this construction approval focused on the design basis of the aqueous polishing (AP) and mixed oxide process (MP) systems, their components, and other related information. The review also encompassed proposed design basis considerations such as redundancy and independence. The staff used Chapter 6.0 in NUREG-1718 as guidance in performing the review. NUREG-1718, Sections 6.3.1 to 6.3.3, concern the NCS Organization and Administration, Management Measures, and Technical Practices.

6.1.1 Organization and Administration

NUREG-1718, Section 6.3.1, states that staff should review the organization and administration to determine whether the applicant has identified the responsibilities and authorities for organizations and individuals implementing the NCS Program. For construction authorization, the design basis includes a description of the roles and responsibilities of the NCS Function during the design phase, and NCS staff education and experience levels. Certain aspects of the NCS Program are only relied on during the operations phase–such as audits and assessments, event investigations, and emergency response; other aspects are relied on during design. However, because specific items relied on for safety (IROFS) have not been identified at the construction authorization stage, significant reliance is placed on the NCS Program in providing reasonable assurance that adequate controls will be established to protect against an accidental criticality, in the event that the MFFF is licensed to operate.

CAR Section 6.1.1 contains information relating to the organization and administration of the NCS Program during the operations phase of the facility. Section 6.1 states that "during the design phase, the criticality safety function is performed within the design engineering organization" but the NCS function during the design phase is not described. NUREG-1718, Section 6.4.3.1, "Organization and Administration," states that the NCS Function should be independent of operations to the extend practical. However, during the construction phase, the staff considers it appropriate that the NCS function be part of the design team to ensure that criticality safety features are designed into the facility.

In DSER Reference 6.3.2, RAI 68, the applicant stated that the qualifications for NCS staff during the design phase would be analogous to those during the operational phase. The specific duties during the design phase differ from those during the operational phase and include establishing NCS design criteria, determining NCS limits and controls, performing criticality calculations, and documenting these results in NCS Evaluations (NCSEs). Staff reviewed the list of duties during the design phase and determined that they were appropriate and likely to ensure that the design would incorporate an adequate level of criticality prevention. This determination was based on standard industry practice and staff's experience licensing other fuel facilities' NCS Programs. Commitments to specific NCS American National Standards Institute/American Nuclear Society (ANSI/ANS) -8 standards (ANSI/ANS-8.1-1983 (R1988) and -8.19-1996) are discussed in the DSER Section 6.1.4, "Design Bases."

The staff evaluated the minimum qualifications (education and experience levels) for the NCS Function Manager, Senior NCS Engineer, and NCS Engineer. The applicant proposed the following education and experience levels for these positions:

- <u>NCS Function Manager:</u> Bachelor of Science (BS) or Bachelor of Arts (BA) degree in science or engineering; at least 2 years nuclear industry experience in criticality safety.

- <u>Senior NCS Engineer</u>: BS or BA in science or engineering; at least 2 years nuclear industry experience in criticality safety.

- <u>NCS Engineer:</u> BS or BA in science or engineering; at least 1 year nuclear industry experience in criticality safety.

The staff concluded that the education levels for these positions were appropriate based on a knowledge of comparable education requirements at other fuel facilities. The staff conducted a review of experience requirements for other nuclear fuel facilities, as documented in NRC's November 9, 2001, letter. Given the lack of experience relevant to Pu processing, the staff concluded that the experience levels cited were not sufficient without additional justification. Moreover, comparable industry experience is likely to consist of experience at low- and high-enriched uranium fuel fabrication and enrichment plants. Much of criticality safety is practiced by "skill-of-the-craft" and requires an intuitive understanding of the neutron physics and margins of safety for the materials and systems being evaluated. Therefore, the staff recommends that plutonium (Pu) or MOX-specific experience be included for at least senior NCS positions. In addition, the applicant stated that the NCS Function Manager should have some experience directing an NCS Function; given the relative risk and difficulty in controlling Pu solution systems, the amount of such experience should be specified. The U.S. Nuclear Regulatory Commission (NRC) provided guidance to the applicant summarizing the regulatory requirements for other NRC-regulated facilities in DSER Reference 6.3.8 In DSER Reference 6.3.6 the applicant provided additional commitments, adding the requirement that the NCS Function Manager's degree be in nuclear science or engineering, and increasing the experience levels for the NCS Function Manager and Senior Engineer from 2 to 3 years which addresses many of the staff's concerns. However, the need for experience in Pu or MOX processing has not been fully addressed. Therefore, the experience levels for NCS staff has not been adequately resolved and is considered an open issue.

6.1.2 Management Measures

NUREG-1718, Section 6.3.2, states that staff should review the applicant's management measures in support of the applicant's ability to implement and maintain the NCS Program, and to ensure the continued availability and reliability of IROFSs. During the construction authorization review, the specific IROFS have not been identified, and therefore, the specific management measures that are applied to them cannot be specified. However, the staff reviewed the quality level definitions and the associated statements pertaining to quality levels of individual IROFSs, in Section 2.2, "Graded Quality Assurance," of the MOX Process Quality Assurance Plan (MPQAP). The staff considered a description of the management measures and quality levels part of the design basis of the facility. These quality level definitions were approved in DSER Reference 6.3.7; questions remained, however, concerning the examples of how these quality levels are applied in practice (i.e., application to different classes of IROFSs), as discussed below.

The controlled parameters have been identified as part of the design basis of the facility for NCS, but the specific controls to be relied on for NCS have not, as they will be later specified as IROFS by the applicant, and identified in its ISA summary to be submitted as part of its application for a 10 CFR Part 70 operating license. Since the specific control features for prevention of criticality will be developed during the design stage, it is appropriate that the means of providing for criticality control (as required by 10 CFR 70.64(a)(9)) will be described at the parameter level for construction review. Therefore, there is no information regarding the selection of specific controls for criticality safety or the management measures to be applied to them. The only specific structures, systems, and components (SSC) addressed in the CAR is the criticality alarm system required by 10 CFR 70.24 (discussed in DSER Section 6.1.3.2). DSER Reference 6.3.7 approved the description of the quality levels as discussed in Section 2.2 of the MPQAP and the September 4, 2001, letter. However, the categorization of specific SSCs was not approved in DSER Reference 6.3.7. Therefore, the staff makes no conclusion concerning the adequacy of the application of quality levels to specific criticality-related SSCs at this time. Conclusions regarding the quality levels are not required in deciding whether to approve the CAR.

Specific administrative management measures applicable to NCS (e.g., training, procedures, and audits and assessments) are not applicable during the construction phase. Commitments to specific NCS ANSI/ANS-8 standards (ANSI/ANS-8.1-1983 (R1988) and -8.19-1996) are discussed in this DSER, Section 6.1.4, "Design Bases."

6.1.3 Technical Practices

NUREG-1718, Section 6.3.3, states that staff should review the applicant's implementation of NCS technical practices to ensure safe operation of the facility. For construction authorization, only those technical practices related to the design of criticality safety PSSCs are applicable. Design basis information includes: (1) commitments describing the design philosophy for meeting the performance requirements of 10 CFR 70.61 and baseline design criteria of 10 CFR 70.64; (2) the technical practices related to determination of criticality safety limits, including calculational methods and criticality code validation; and (3) the technical practices related to determination of controls, including the preferred hierarchy of controls and measures to ensure control reliability and availability. Many of these technical practices are contained in ANSI/ANS-8 Standards, and therefore the technical practices CAR Section should contain a description of which codes and standards the applicant is committing to. Commitments to specific ANSI standards (ANSI/ANS-8.1-1983 (R1988) and -8.19-1996) are discussed in this DSER, Section 6.1.4, "Design Bases."

6.1.3.1 Commitment to Baseline Design Criteria

The staff reviewed the MFFF CAR and the associated RAI responses for a commitment to the BDC and found that the applicant committed to the double contingency principal as required by 10 CFR 70.64(a)(9). The staff considers the commitments implementing the requirement to ensure that processes remain subcritical under normal and credible abnormal conditions to be part of the design basis of the facility. (Staff review regarding the definition of "unlikely" in the DCP, and "highly unlikely" for criticality events, is covered in DSER Section 6.1.4. While 10 CFR 70.61 does not have to be met for construction authorization, the applicant has identified criticality as a high-consequence event that will need to be made "highly unlikely", and stated that it will meet this by applying the DCP.) This commitment also described the process for flowing down NCSE requirements into the ISA as IROFS.

Compliance with the DCP will be demonstrated by identifying two or more process conditions which are relied on to ensure a subcritical configuration. The MFFF NCSEs will evaluate both normal and credible abnormal conditions and the associated potential failure paths for any common mode failures as well as examining the interaction effects between fissionable material units. The NCSEs will use geometry controls as the preferred controlled parameter for plant systems and may use fixed neutron absorbers as necessary.

These evaluations will also serve to promote a defense-in-depth philosophy in the facility design and plant layout and will adhere to a preferential hierarchy of controls. In order to enhance the inherent reliability of criticality controls, the following criteria will be used:

- Passive engineered features will be preferred over active engineered features.
- Engineered features will be preferred over administrative controls.
- Enhanced administrative controls will be preferred over simple administrative controls.

In instances where controlled parameters utilize physical measurements for compliance, representative sampling and analysis will be used. These sampling and analysis requirements will be included in the list of IROFS provided with the license application for possession and use of special nuclear material and will be flowed into facility management measures.

The staff, therefore, concludes that the commitment to the BDC is appropriate and provides reasonable assurance that the design of the MFFF will be in general agreement with the regulatory acceptance criteria. In addition, the description of the NCSE process provides reasonable assurance that an adequate safety basis will be established and documented.

6.1.3.2 MFFF CAAS

The staff reviewed the MFFF CAAS described in CAR Section 6.3.2 in light of the requirements in 10 CFR 70.24 and guidance in NUREG-1718 Section 6.3.3. Although the applicant has not listed the CAAS as a PSSC, it stated that the CAAS will be designed in accordance with 10 CFR 70.24, and the staff thus finds the CAAS to be part of the design basis of the facility. The CAR application must therefore include a description of the specifications of the CAAS and the management measures that ensure operability of the CAAS and emergency response procedures. (Additional requirements are found in the discussion on commitments to ANSI/ANS-8.3-1997 [DSER Section 6.1.4].)

CAR Section 6.3.2 states that "ANSI/ANS-8.3-1997, Criticality Accident Alarm System, is the main guidance for alerting personnel that an inadvertent criticality has occurred. The main requirement linked to the design of the system is the reliability of actuation of the alarm."

The applicant describes the CAAS as a monitoring system composed of groups of detectors called monitoring units (detector network, data processing, and alarm actuation units) that will activate audible and visual alarms (network of audible and visual alarms, an off-line processing facility) in case of a criticality accident. The CAAS will be designed to detect both gamma and neutron radiation, and to actuate within one-half second of detector recognition of a criticality accident. The range and design features of the alarm will also follow the guidance provided in ANSI/ANS-8.3-1997. In DSER Reference 6.3.3, the applicant further committed to two alarms in all non-exempt areas of the facility. In DSER Reference 6.3.3, the applicant commits that identification of any areas to be exempted and the basis for the exemption will be provided with the license application. Together with the commitment in CAR section 6.3.2 and ANSI/ANS 8.3-1997, this provides reasonable assurance that all the applicable regulatory requirements for the CAAS will be met.

The NRC staff considers that the material presented by the applicant in Section 6.3.2 of the CAR, and supplements, provides reasonable assurance that the design basis for the MFFF CAAS will be in broad agreement with the regulatory acceptance criteria, and therefore, the staff concludes that this section of Chapter 6 of the CAR is acceptable.

6.1.3.3 Criticality Safety Control Design Criteria

Section 6.3.3 of the CAR was reviewed by the staff according to NUREG-1718, Section 6.3.3, to ensure the safe design of operations for the proposed facility. Specifically, NUREG-1718 (Section 6.3.3.) includes the expectation that "the technical practices to ensure that sufficient NCS controls, developed in the criticality safety evaluations (CSEs) and flowed into the ISA, are identified for each process." The staff considers the description of the technical practices for each controlled parameter and the design criteria–including the preferred hierarchy of controls–to be part of the design basis of the facility.

In Section 6.3.3.2 of the CAR, the applicant discusses the criticality control modes, and available methods of control to be used in the MFFF. Under the "Criticality Control Modes" (CAR Section

6.3.3.1), the applicant discusses passive and active engineered controls, and enhanced and simple administrative controls. The staff asked for clarification of the conditions under which neutron interaction is to be considered. This information was necessary to ensure that the process remains subcritical under both normal and credible abnormal conditions in accordance with 10 CFR 70.61(d). The applicant explained in DSER Reference 6.3.2, RAI 76, that "the neutron interaction is considered for all conditions to ensure that the process remains subcritical under all normal and credible accident conditions." The applicant commits to modify CAR Section 6.3.3.1 to reflect that the potential for neutron interaction between units will be fully evaluated and guaranteed to remain subcritical under all normal and credible abnormal conditions. The NRC staff found the applicant's answer acceptable.

The staff reviewed the special status afforded to fixed neutron absorbers and whether the use of other type of neutron absorbers were considered. This information was necessary to ensure that the design bases adequately provide for criticality control, as required by 10 CFR 70.64(a)(9). The applicant explains in DSER Reference 6.3.2, RAI 77, that fixed neutron absorbers represent a very reliable means of criticality control often used in conjunction with geometry control. The design of the MFFF will incorporate fixed neutron absorbers along with the implementation of ANSI/ANS-8.21-1995 guidance. NUREG-1718 also allows the use of borosilicate glass raschig rings provided the applicant commits to ANSI/ANS 8.5-1996. The applicant does not consider other types of neutron absorbers as inherently reliable as fixed neutron absorbers and does not envision them to be employed in any MFFF process unit or area (see also DSER Section 6.1.4 on ANSI Standards). The applicant states that any applications of neutron absorber control in the final MFFF design will be clearly identified in the license application and ISA. The NRC staff found the applicant's answer acceptable.

In reviewing the controlled parameters for the MOX and Aqueous Polishing processes, the staff reviewed the applicant's use of the preferred design approach. In the CAR, Section 6.3.3.1, "Criticality Control Modes," the applicant committed to the preferred use of passive engineered over active engineered control, and preferred use of active engineered over administrative control, as well as the preferred use of geometry control. The applicant committed to the preferred use of dual over single-parameter control as part of the design approach[1]. Reliance on diverse control modes is important to minimize the potential for common-mode failure. However, the applicant's response (DSER REference 6.3.2, RAI 80) stated that "a criticality control scheme incorporating the preferred MFFF hierarchy of criticality controls is not feasible for a MFFF. Therefore, the MFFF design preference is to rely on passive geometry control as the preferred criticality safety control, followed by reliance on dual independent controls on control parameters." Even when geometry control is used, diverse control modes should be used and independent controls are required to meet the DCP. In the October 11, 2001, meeting, the applicant stated that even when geometry is the sole controlled parameter, the DCP will still be complied with, but that if no credible means is identified for changing the system geometry there is no need for additional controls. Accident sequences in which there are no credible means to achieve criticality do not require dual independent controls to comply with the

[1] While this commitment does not preclude the use of administrative controls or control modes other than geometry, these are to be preferred over other types of controls; the process descriptions in CAR sections 11.2 and 11.3, and the CAR Tables 6-1 and 6-2 were reviewed to confirm that the facility will be designed in accordance with this design philosophy to the greatest extent practical. (e.g., passive geometry control used on a majority of processes).

DCP. The staff considered this clarification adequate (as stated in Appendix B of DSER Reference 6.3.5). However, DSER Reference 6.3.6 stated:

> This preference or passive geometry control meets the DCP...In the case of geometry control, there are no credible changes in process conditions which can occur causing a criticality...the geometry of the unit is fixed by design and therefore can not change.

This response was unacceptable to the staff because there are credible scenarios under which the geometry of the fissionable material could change–either through degradation of the geometry barrier (e.g., bulging or corrosion), or through loss of containment and accumulation of material in an unsafe geometry. These possible abnormal conditions cannot be dismissed without being evaluated and it has not been demonstrated that they cannot occur at the MFFF. In DSER Reference 6.3.6, the applicant stated that it would evaluate credible means of changing the system geometry and establishing sufficient controls to ensure that the double contingency principle is met. Because geometry controls are the preferred method of control, and this commitment is necessary to meet the double contingency principle as stated in 10 CFR 70.64(a)(9), for cases that involve geometry control, this response adequately resolves the staff's concern.

Under the "Available Methods of Control" (CAR Section 6.3.3.2), the applicant discusses geometry, mass, density, isotopics, reflection, moderation, concentration, interaction, neutron absorber volume heterogeneity, and process variable controls. Technical practices associated with most of these controlled parameters were found to be acceptable, based on standard industry practice and a comparison with the acceptance criteria of NUREG-1718. The staff did raise issues with the technical practices for certain of the parameters, as discussed in the following paragraphs:

- CAR, Section 6.3.3.2.1, "Geometry Control," states that "tolerances on nominal design dimensions are treated conservatively"; however, it does not state that the most reactive combination of tolerances will be determined and used. (This observation also applies to all other controlled parameters.) When questioned in the RAI, the applicant responded (DSER Reference 6.3.2, RAI 78) stating that "the design approach with respect to criticality ... for each controlled parameter, assume the credible optimal condition (i.e., most reactive condition physically possible) for the parameter, or calculate the allowed range for the parameter. Criticality calculations and nuclear criticality safety evaluations are performed assuming the most reactive physical condition to ensure that the process remains subcritical under all normal and abnormal conditions, in accordance with 10 CFR 70.61(d)." The NRC staff considered the applicant's answers acceptable and no further action was necessary.

- DSER Reference 6.3.1 relates to two assumptions made by the applicant: (1) that the presence of ^{241}Pu can be neglected (CAR Section 6.3.3.2.4) and (2) that 1-inch of water can be used to conservatively represent reflection (CAR Section 6.3.3.2.5). The staff stated that it is necessary to demonstrate these assumptions during the design phase, to ensure that the process remains subcritical under both normal and credible abnormal conditions.

- In answer to the question about Assumption 1, the applicant stated in DSER Reference 6.3.2, RAI 79, that "specifications of plutonium isotopics used in the MFFF include 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," and that "a calculation will be prepared to show that 96 percent ^{239}Pu and 4 percent ^{240}Pu as used in criticality calculations bounds the presence of ^{240}Pu (5 percent to 9 percent) and ^{242}Pu (<0.02 percent) and offsets any contribution from ^{241}Pu (<1 percent) such that it can be neglected." The applicant also stated that clarifying statements will be added to CAR Section 6.3.3.2.4 indicating that these statements will be demonstrated in an MFFF criticality calculation and referenced in NCSEs.

- In the answer on Assumption 2, the applicant stated in DSER Reference 6.3.2, RAI 79, that "... at a minimum, reflection conditions equivalent to a 1-inch tight fitting water jacket are assumed to account for personnel and other transient incidental reflectors not evaluated in unreflected models. However, it is MFFF criticality calculation practice that the ranges of water reflection up to and including 12 inches (30 cm) be employed. The calculations as referenced by the NCSEs shall provide a demonstration of the full range of reflection. Any exceptions to this principle, such as in the moderation controlled, normally dry areas, shall be fully justified in the NCSEs." This approach is conservative and corresponds to common industry practice for modeling both incidental and full reflection.

- CAR, Section 6.3.3.2.6, "Moderation Control," states that in moderation-controlled areas, hydrogenous fire suppressants will not be used. The applicant's approach to balancing the combined risk of criticality and fire during the facility design is part of the design basis of the facility. This requires an integrated approach to safety, and while this is not limited conceptually to the interaction between fire protection and criticality safety, this has historically been one of the most significant areas of overlap. While the commitment to avoid moderating fire suppressants is acceptable to the criticality staff (adequacy from the standpoint of fire protection is outside the scope of this chapter), staff noted an inconsistency between this section and clarifications to commitments to ANSI/ANS-8.22-1997, "Nuclear Criticality Safety Based on Limiting and Controlling Moderators" (see DSER Section 6.1.4).

The staff considers that the information provided by the applicant, both in the CAR and in the responses to the RAIs, gives reasonable assurance that the design criteria to be used in the MOX criticality safety control, CAR Subsection 6.3.3, are in broad agreement with the regulatory acceptance criteria and concludes that this section of Chapter 6 of the CAR and the responses to the RAIs are acceptable.

6.1.3.4 Criticality Safety Process Description

The staff reviewed the process descriptions in CAR Sections 11.2, "MOX Process Description," and 11.3, "Aqueous Polishing Process Description," along with the tables of associated criticality control units (CCUs), Tables 6-1 and 6-2. CAR Chapter 11 provides an overview of both the MP and AP processes. Tables 6-1 and 6-2 list each of the 28 AP and 25 MP CCUs and their controlled parameters.

Although specific controls have not been identified during the construction phase, the design bases include the dominant controlled parameters and their approximate ranges relied upon in each major process area, as discussed in the January 2001 public meeting. (For most parameters, the staff did not consider specific numerical ranges necessary; e.g., acceptable commitments to technical practices for process evaluation provide reasonable assurance that appropriate dimensions will be determined for favorable geometry equipment.) During the January 3-4, 2001, meeting, the staff requested that the applicant submit, for each major process area, the dominant controlled parameters. This information is presented in Tables 6-1 and 6-2. Staff review in the following sections is based on the revised Tables 6-1 and 6-2 that were submitted in the DSER Reference 6.3.2, RAI 83.

6.1.3.4.1 NCS - AP Process

The AP Process, as described in the revised Table 6-1 and CAR Section 11.3.2, consists of 13 major areas that have been subdivided into 28 CCUs. PuO_2 powder is received from offsite in cans, and then batched to the dissolution unit where it is electrolytically dissolved into plutonium nitrate (primarily $Pu(NO_3)_4$) solution. Following this, it is purified in pulsed solvent extraction, scrubbing, and stripping columns. During these stages, the plutonium nitrate chemical form and process equipment favorable geometry are the dominant controlled parameters. Following purification, the plutonium nitrate is precipitated and calcined to form a purified PuO_2 powder used as feed for the MP, all still in favorable geometry process equipment. Unprecipitated PuO_2 is filtered and then recovered in the Oxalic Mother Liquor Recovery Unit. Additional process units recover solvent, nitric acid, silver, offgases, and liquid waste, and recycle raffinate produced in the solvent extraction. These units primarily rely on dual independent concentration controls as they are designed not to contain plutonium under normal process conditions.

In reviewing the AP process description, the staff identified a mismatch between the description in CAR Chapter 11 and Table 6-1. CAR Section 11.3.2, identified 13 different process units, whereas Table 6-1 contained 25 CCUs. The exact number was difficult to determine, because some units appear to have combined or have different names in the two locations. DSER Reference 6.3.2, RAI 83, revised Table 6-1, which resulted in a total of 28 CCUs. The applicant also provided a cross-reference between the 13 process units in Section 11.3.2 and the 28 CCUs in Table 6-1 (Note: Three of the process units and three of the CCUs are omitted from the table in the response, but could be matched up by inspection. These units do not normally contain fissile material and rely on dual independent concentration control). This cross-reference was sufficient to resolve the difference between the two descriptions of the AP Process.

For each CCU in CAR Table 6-1, the dominant controlled parameters are listed along with parameter ranges for some parameters. The chemical form through most of this process is primarily aqueous PuO_2 or $Pu(NO_3)_3$ solution[2], or plutonium oxalate ($Pu(C_2O_4)_2$, which the applicant assumes is bounded by plutonium oxyfluoride, PuO_2F_2); this is converted to PuO_2 during calcination. Since $Pu(NO_3)_3$ is less reactive than a PuO_2 or Pu-metal solution due to the neutron absorption of nitrogen, the chemical form must be controlled for criticality safety. NCSEs should demonstrate the bounding nature of these compounds during the design phase.

According to CAR Table 6-1, most of the process is conducted in favorable geometry equipment; although dimensions are not specified, the general geometrical shape is (i.e., cylindrical, slab, annular). It is anticipated that most of the AP processes will occur in cylindrical columns or slab/annular tanks. Dimensions will be based on validated calculational methods, standards (such as ANSI/ANS-8.1-1983 (R1988)), or handbook data. Dimensions are not provided, but the applicant's commitment to the DCP should ensure that favorable geometry units will be subcritical (i.e., $k_{eff} + 2\sigma \leq$ Upper Subcritical Limit) under normal and all credible abnormal conditions. While favorable geometry limits may be based on calculations assuming specific chemical and physical characteristics of the material, its application must be consistent with the applicant's commitment to the DCP. That is, if it is credible for the physicochemical

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[2]Although plutonium can exist in a variety of valence states, $Pu(NO_3)_4$ is the most common form. The applicant stated it will assume $Pu(NO_3)_3$ in the analysis because it is more reactive.

form of the material to change, this possibility must be considered as a process upset in evaluating compliance with the DCP.

CAR Table 6-1 assumes specific densities of PuO_2 powder and isotopic composition ($\geq$4 percent ^{240}Pu) but are not identified as controls. These bounding assumptions on the incoming materials are relied on for criticality control, but not justified in CAR Table 6-1. The applicant has agreed to provide additional justification for those parameter values which are assumed less than optimal but not specifically controlled. In DSER Reference 6.3.6, the applicant provided clarification sufficient to resolve the concern over physiochemical form, but stated that the justification for bounding densities would be submitted at a later date. Therefore, justification of these values has not been adequately resolved and is considered an open issue.

In several CCUs, the applicant identified neutron absorption as a means of control. The staff reviewed the application of neutron absorption and concluded that it is appropriate (in several cases neutron absorption is combined with geometry control–i.e., the presence of fixed absorbers is assumed in determining subcritical geometric limits). The use of fixed absorbers is recognized in industry practice as a highly reliable means of control, especially in conjunction with geometry control, and thus complies with the preferred hierarchy of controls. In each case, the presence of fixed neutron absorbers was used as the means of control.

The applicant proposed concentration control for those units not relying on favorable geometry, where fissile material is not expected under normal conditions. As indicated in its August 31, 2001, RAI response (Question 84), the applicant commits to dual independent controls on concentration as the presence of plutonium in these units could lead to inadvertent criticality, which is often used to meet the DCP and is therefore acceptable to the staff.

As discussed in DSER Section 8.1.1.5.1, “Dissolver Chemistry and Reactions,” high enriched uranium (HEU) is separated from the incoming plutonium stream in the Uranium Stripping and Diluent Washing Mixer Settlers of the Purification Unit (KPA). This HEU is isotopically diluted with depleted uranium in two stages, first to ~30wt percent ^{235}U assay, and then to $\leq$ 1wt percent ^{235}U assay. Dilution to ~30wt percent assay occurs in the Reception Tank of the Dissolution Unit (KDB) and dilution to $\leq$ 1wt percent assay occurs in one of two Dissolution Tanks in the Liquid Waste Reception Unit (KWD) before being transferred to high-alpha waste.

Although this two-stage isotopic dilution has been described elsewhere in the CAR, this has not been described in CAR Table 6-1. Isotopic abundance is controlled for the Uranium Stripping and Diluent Washing Mixer Settlers CCU, which states that the ^{235}U assay will be limited to $\leq$ 93.5wt percent. However, there is no mention of controlling the uranium assay in either the Reception Tank or Liquid Waste Reception Unit CCU, and there are no criticality controls identified for the high-alpha waste unit. Control of uranium assay is required because the high-alpha waste assay must be less than 2wt percent ^{235}U to be critically safe, and thus criticality controls must be defined for the isotopic dilution. In addition, the staff noted that there is the potential for accumulation of up to 3 kg of PuO_2 powder in the MFFF ventilation system, but there has been no criticality controls defined for the ventilation system. While CAR Tables 6-1 and 6-2 provide design basis controlled parameters for the main AP and MP steps, design bases are not defined for auxiliary systems, such as ventilation and waste processing. Therefore, definition of NCS design basis controlled parameters for AP and MP process auxiliary systems (specifically including process ventilation, isotopic dilution, and high-alpha waste) has not been adequately resolved and is considered an open item.

The staff reviewed the application of the preferred design approach (i.e., the preferred use of passive over active engineered over administrative control, and the preferred reliance on favorable geometry) in the AP process, and concluded that it was appropriate, because of the observed predominance of passive engineered controls. In addition, staff reviewed the application of the commitment to the preferred use of dual over single parameter control in CAR Table 6-1, and concluded that it was appropriate, because most processes had two or more parameters defined. Given the higher inherent risk of the AP process relative to the MP, due to the forms and types of materials being processed, the favorable geometry design of the facility is appropriate. Most of the process relies on favorable geometry, with the exception of the concentration-controlled units. Moreover, fixed neutron absorbers are used in the process as part of the design. The identified parameters provide reasonable assurance that the design of the facility will be based primarily on passive design features, in accordance with the preferred control hierarchy. The only notable exceptions to this are the density and isotopic composition, which have bounding values defined that are not passively controlled (the basis for these bounding values has been identified as an open issue). Coupled with the commitment to follow the preferred design approach during the design of actual IROFSs, this table provides reasonable assurance that the design of the AP process will be in broad agreement with the regulatory acceptance criteria, with the exception of the open items identified above.

6.1.3.4.2 NCS - MP

The MP, as described in the revised CAR Table 6-2 and CAR Section 11.2.2, consists of five major areas that have been subdivided into 25 CCUs. In the Receiving Area, depleted UO_2 powder as well as purified PuO_2 powder (in 3013 containers) is received and stored prior to being used as feed in the powder area. Depleted uranium poses no criticality concerns, while the PuO_2 powder is stored in closed 3013 containers.[3] Geometry and spacing in the PuO_2 3013 Storage Pit and Buffer Storage unit are controlled, along with the moderation inside containers. In the Powder Area, the PuO_2 containers are emptied inside a glovebox onto a conveyor supplying the dosing unit, where the depleted UO_2 and the PuO_2 powder is blended, homogenized, and milled to form the master blend. The relative proportion of UO_2 and PuO_2 is carefully controlled for product specification and criticality purposes. The master blend has a composition of $\leq$ 22wt percent Pu; the final blend consists of $\leq$ 6.3wt percent Pu. During this process, mass, isotopic composition, and moderation are the dominant controlled parameters. The Powder Process ends with the addition of a controlled amount of poreformer and pelletizing of the powder into green pellets. The remaining process areas (Pellet Process Area, Fuel Rod Process Area, and Assembly Area) consist of sintering and handling the green pellets, loading of pellets into rods, and loading of rods into assemblies, in a similar fashion to that in other fuel manufacturing processes. During these processes, the chemical and geometric form, isotopic composition, and moderation are the dominant controlled parameters.

In reviewing the MP description, the staff identified a mismatch between the description in CAR Chapter 11 and Table 6-2. CAR Section 11.2.2 identified 38 different process units, whereas CAR Table 6-2 only contained 20 CCUs. The exact number was difficult to determine, because some units appear to have been combined or have different names in the two locations. DSER

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[3] There are some inconsistencies between CAR Table 6-2 and CAR Section 11.2.2 (e.g., primary dosing and PuO_2 container handling are in the Receiving Area according to CAR Table 6-2, but in the Powder Area according to CAR Section 11.2.2). This is not safety-significant and the areas defined in CAR Section 11.2.2 are used in this discussion.

Reference 6.3.2, RAI 83, revised CAR Table 6-2, which resulted in a total of 25 CCUs. The applicant provided a cross-reference between 36 of the 38 process units in CAR Section 11.2.2 and the 25 CCUs in CAR Table 6-2. The remaining two process units involved UO_2 receiving and storage and UO_2 drum emptying. Since this is depleted uranium, there are no criticality controls in those areas.

For each CCU in CAR Table 6-2, the dominant controlled parameters are listed along with parameter ranges for various parameters. The chemical form throughout this process is either purified PuO_2 or MOX. Powder in the Receiving Area is handled in fixed geometry 3013 containers and stored in fixed storage arrays. Once the powder is removed from the containers, criticality safety is based on mass and moderation control in lieu of geometry control. Favorable geometry is also a dominant control in the processing and handling of pellets, rods, and fuel assemblies, once the fuel has been processed into a fixed configuration.

Following primary dosing, the isotopic composition of the MOX powder is controlled for criticality safety. The primary blend consists of MOX with $\leq$ 22wt percent Pu, and is stored in J60 jars; the final blend consists of MOX with $\leq$ 6.3wt percent Pu and is stored in J80 jars. These are controlled by geometry; in addition, the MOX operations are batch processes that rely largely on mass control for criticality safety.

Moderation is relied on for criticality safety in much of the MOX process, including primarily the powder handling and storage operations. Moderator is limited to $\leq$ 1wt percent H_2O for incoming PuO_2 powder, is limited by the controlled addition of organic additives following the primary dosing CCU, and is limited further downstream by controlling the amount of poreformer added.

Heterogeneity is controlled by ensuring a homogeneous mixture of UO_2 and PuO_2 powder to prevent the formation of high-plutonium “hot spots” in the fuel.

Certain parameters (such as density and isotopic composition) have bounding values listed that are relied on for criticality control, but are not specifically credited as criticality control or justified in CAR Table 6-1. The applicant has agreed to provide additional justification for those parameter values which are assumed less than optimal but not specifically controlled. In DSER Reference 6.3.6, the applicant provided clarification sufficient to resolve the question of controlling physiochemical form (i.e., $Pu(N0_3)_3$). DSER Reference 6.3.6 stated, however, that assumptions about bounding powder densities would be provided in a subsequent response. Therefore, justification of the values has not been adequately resolved and is considered an open issue.

The staff reviewed the application of the preferred design approach (i.e., the preferred use of passive over active engineered over administrative control, and the preferred reliance on favorable geometry) in the MP, and concluded that it was appropriate because of the observed predominance of passive engineered controls. In addition, staff reviewed the application of the commitment to the preferred use of dual over single parameter control and concluded that it was appropriate. The majority of CCUs in the MP utilize favorable geometry for criticality control. Where geometry control is not used, mass and moderator control is typically used. As stated above, these operations consist of batch processes inside gloveboxes designed to exclude moisture. The relative risk of the MP is low compared with that of the AP Process, especially following downblending of the PuO_2 into MOX powder and subsequent pelletizing. The identified parameters provide reasonable assurance that the design of the facility will be based primarily on passive design features, in accordance with the preferred control hierarchy. Coupled with the commitment to follow the preferred design approach during the design of actual IROFSs, this table provides reasonable assurance that the design of the MP will be in broad agreement with the regulatory acceptance criteria, with the exception of the open items identified above.

6.1.3.4.3 Single-Parameter Limits

In Tables 6-3 and 6-4, the applicant provided a list of permissible single-parameter limits for various parameters, such as subcritical masses, dimensions, and volumes for use in the MFFF design. In its August 31, 2001, response to Questions 103 and 104 concerning the bases for these limits, the applicant stated that they are order-of-magnitude estimates that will not be used directly in the design (*i.e*., without use of validated methods or approved standards). However, the table titles and text state that they are “permissible” or “allowable” values of applicable parameters. The applicant agreed in the October 11, 2001, public meeting to clarify the role and use of these tables. In its December 5, 2001, letter, the applicant committed to amend CAR Section 6.3.4.5 to state that Tables 6-3 and 6-4 will not be used as references in criticality calculations or other safety evaluations. Therefore, the staff did not evaluate the adequacy or derivation of the values in these tables as part of the CAR review.

6.1.3.5 Nuclear Criticality Analysis and Safety Evaluation Methods

The staff considers a description of the calculational methodology, including computer code validation methodology, to be part of the design basis of the facility. The staff also considers the design basis to include the maximum k_{eff}, or Upper Subcritical Limit (USL). ANSI/ANS-8.1-1983 (R1988) requires that processes be shown to be subcritical under both normal and credible abnormal conditions; this requires determination of the USL. The applicant has committed to the methodology described below to ensure that there will be an adequate margin of subcriticality for safety, which will later be demonstrated in accordance with 10 CFR 70.61(d) as part of its application for a 10 CFR Part 70 operating license. ANSI/ANS-8.1-1983 (R1988) is one of the basic documents used by the applicant in the discussion of CAR Section 6.3.5, "Nuclear Criticality Analysis and Safety Evaluation Methods." The applicant stated that ANSI/ANS-8.1-1983 (R1988) provides (1) single and multi-parameter limits to be referenced in criticality safety calculations, and (2) guidance to be used in the performance of criticality analysis method validation. Other documents used in support of the information presented in CAR Section 6.3.5 include NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," and NUREG/CR-6655, "Sensitivity and Uncertainty Analyses Applied to Criticality Safety Validation." The results and findings of the NRC staff review are described in the sections following. The applicant’s methodology for validation is described below.

In the CAR, the applicant discussed the criticality analysis methodology to be used in MFFF design activities and facility safety programs, as discussed in ANSI/ANS-8.1-1983 (R1988). The applicant stated that the computational methods of choice are the ones with Evaluated Nuclear Data File (ENDF) cross-section libraries (i.e., the Standardized Computer Analyses Licensing Evaluation (SCALE) 4.4 code system with the 238-group cross-section library derived from the ENDF/B-V file of evaluated nuclear physics data, and the Monte Carlo Neutron Photon (MCNP) code, with ENDF60 cross-section libraries.)

The applicant discussed the method of validation and calculated k_{eff} design limits to be used in the MFFF. As discussed in ANSI/ANS-8.1-1983, the applicant code validation methodology will use critical experiment benchmarks to determine bias and bias uncertainties. The critical experiments are to be selected to represent the neutronic characteristics of the systems analyzed in specific design applications. The derived biases are to be applied to calculations on systems whose neutronic characteristics are bounded by those of the selected set of experimental benchmarks. When the neutronic characteristics of the system analyzed are not bounded by those of the experimental benchmarks, additional adjustments for bias and uncertainties will be employed. The applicant states that criticality safety calculations will be performed for each of the stages of process operations at MFFF (ranging from receipt of PuO_2 and UO_2 powders through fabricated MOX fuel assembly storage and shipment).

The validation approach to be used by the applicant is similar to the ones already approved by NRC in previous license application approvals for a broad range of applications. This validation approach is described in NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," and NUREG/CR-6655, "Sensitivity and Uncertainty Analyses Applied to Criticality Safety Validation." The "International Handbook of Evaluated Criticality Safety Benchmark Experiments" (ICSBEP) provides a large compilation of benchmark criticality experiment descriptions. RAI 88 requested a description of statistical methods to be used to benchmark the criticality codes where there is insufficient statistical data to directly use methods similar to NUREG/CR-6361. The applicant outlined a variety of sensitivity methods that could be used, which will be more fully described in subsequent parts of the Validation Report; this information has not yet been submitted, and therefore has not been reviewed as yet by the staff and it remains an open item for construction authorization review.

The NRC staff determined that the code validation methodology described generally in CAR Section 6.3.5 was appropriate; validation of specific areas of applicability (AOAs) intended to cover anticipated AP and MP operations is covered in the following sections.

6.1.3.5.1 NCS Validation Report Parts I and II

10 CFR 70.64(a)(9) requires that the applicant provide for criticality control, including adherence to the DCP. ANSI/ANS-8.1-1983 (R1988) defines double contingency as requiring at least two independent, unlikely, and concurrent changes in process conditions before criticality is possible. ANSI/ANS-8.1 also requires that processes be subcritical under both normal and credible abnormal conditions. The applicant has chosen to commit to this standard to meet, in part, 10 CFR 70.64. Determining the bias, uncertainty in the bias, and administrative margin are all required to select a USL that ensures an adequate margin of subcriticality.

The applicant used the statistical methods in NUREG/CR-6361, which contains a description of two methods for determining the USL. In Method 1 (Confidence Band with Administrative Margin), the user chooses an administrative margin *a priori*, Δk_m, and the computer code USLSTATS calculates the bias as a function of some physical parameter (*e.g.*, neutron energy, moderation level) to which the k_{eff} is correlated. The code calculates both the bias, $\beta(x)$, and the uncertainty in the bias, $w(x)$, by linear regression of β as a function of the correlating parameter, x, and applies the maximum $w(x)$ over the entire AOA as a constant uncertainty in the bias, $W=\max(w(x)|x_A \leq x \leq x_B)$. The USL is then determined as follows:

$$\text{USL-1} = 1 - \Delta k_m - W + \beta(x)$$

In Method 2 (Lower Tolerance Band), the user computes the single-sided tolerance band, as follows:

$$USL\text{-}2 = 1 - (C_{\alpha/P}\, s_p) + \beta(x)$$

The term s_p is the pooled standard deviation for the set of critical benchmarks, and $C_{\alpha/P}$ is a statistical term for a given confidence α that some fraction P of future calculations below the USL are indeed subcritical. For instance, $C_{95/99}\, s_p$ represents the width of the confidence band that 99 percent of all future calculations below the USL are subcritical, with 95 percent confidence. USL-1 is a predictor for a single future calculation, whereas USL-2 is a predictor for some very high proportion (usually 95 or 99 percent) of all future calculations. As a result, USL-2 is normally significantly lower than USL-1, as it has additional margin resulting from the uncertainty with regard to the population variance (*i.e.*, the distribution of all future calculations). The administrative margin as applied in Method 1 is intended to account for any additional uncertainty not determined under Method 1, which includes this extra uncertainty accounted for in Method 2. Thus, by comparing USL-1 and USL-2, it can be determined whether there is sufficient administrative margin to bound the additional statistical uncertainty accounted for in Method 2. A determination that USL-2 < USL-1 indicates insufficient administrative margin. The converse is not necessarily true.

The administrative margin accounts for both statistical and non-statistical uncertainties; NUREG/CR- 6361 states that for applications involving the transportation and storage of light-water reactor fuel, a minimum administrative margin of 0.05 should be used for Method 1.

The applicant stated that the MOX Validation Report would consist of three parts covering five different Areas of Applicability (AOAs) at the MFFF. AOA(1), covering plutonium solutions, and AOA(2), covering MOX fuel pellets, rods, and assemblies, were submitted as Part I by letter dated June 25, 2001, and reviewed by NRC staff. The revised Part I and initial Part II (including AOA(3), covering PuO_2 powders, and AOA(4), covering MOX powders) were submitted by letter dated December 11, 2001. An additional Part III covering plutonium oxalate systems (AOA(5)) is scheduled to be submitted in early 2002, but has not yet been received. Therefore, the receipt of the Part III Validation Report is considered an open issue.

6.1.3.5.2 AOA(1): Plutonium Nitrate Solutions

The applicant used the methods of NUREG/CR-6361 (i.e., calculation of USL-1 and USL-2 using the computer code USLSTATS) to determine the USL for AOA(1). The applicant used the SCALE 4.4 code package installed and verified on the SGN SUN hardware platform. The applicant modeled 182 critical experiments using KENO-VI with the 238-group cross-section library, to calculate k_{eff} for systems similar to the operations to be conducted under AOA(1). The applicant's chosen experiments included: bare spheres, water reflected spheres, concrete reflected spheres, cadmium/concrete reflected spheres, and cadmium/water reflected spheres. The staff evaluated the applicant's defined AOA against the chosen benchmark experiments to determine that the range was appropriate. The applicant assumed an administrative margin of 0.05, took the bias and uncertainty in the bias into account, set the USL, and determined that the USL was appropriate for AOA(1).

The staff compared the range of parameters covered by the benchmark data to the applicant's AOA and determined that there were a sufficient number of experiments and that they provided adequate coverage of the defined AOA.

The applicant chose 0.05 as the administrative margin for AOA(1) in Method 1. Using that margin and the USLSTATS program, the applicant ran SCALE/KENO VI, calculated k_{eff}, and plotted the evaluated USL-1 and USL-2 values versus both the trending parameters of Energy of Average Lethargy causing Fission (EALF) and the moderator to fuel atomic ratio (H/Pu). Table 6-1 of the Validation Report showed that, for the range of applicability (*i.e.*, EALF from 0.05 to 0.35 and H/Pu ratio from 78 to 1208), the minimum USL-1 with that margin was 0.9399 and the minimum USL-2 was 0.9680. In addition, the applicant performed the same calculation and evaluated USL-1 and USL-2 values versus the trending parameter of ^{240}Pu content. The result showed that the bias was positive and that it increased slightly with an increase in ^{240}Pu content. The applicant determined this was conservative because the maximum considered ^{240}Pu content of 4.7 percent in the critical experiments chosen was bounding for the design value of 4 percent ^{240}Pu content. Table 6-1 showed that, for ^{240}Pu content, the minimum USL-1 with that margin was 0.9449 and the minimum USL-2 was 0.9867. Comparing the results of USL-2 with USL-1, for EALF, H/Pu ratio, and ^{240}Pu content, the USL-2 was less than USL-1 and therefore the applicant determined that the use of 0.05 as the administrative margin was appropriate.

Taking into account the trending analysis results, the applicant concluded that the minimum USL for the AOA(1) systems was 0.9399, which included taking into account an administrative margin of 0.05, as well as a calculational bias of 0.0101 (which includes an allowance for uncertainty in the bias). The applicant justified that the 0.05 administrative margin was adequate because: (1) comparison of the USL-1 and USL-2 values demonstrates that the USL-1 with that administrative margin was conservative; (2) historically, an administrative margin of 0.05 has commonly been applied by the NCS community; and (3) based on an adequate number of representative benchmarks covering the AOA(1) design conditions, NUREG-1718 recommended an administrative margin of 0.05.

NRC staff determined that the applicant could use USLSTATS to determine the USL for AOA(1). NRC staff determined that the applicant could use a properly installed and verified SCALE 4.4 code package to calculate k_{eff} with the Monte Carlo computer code KENO VI. NRC staff further determined that the applicant could use the chosen benchmark experiments run with the 238 group cross-section library using KENO VI to represent the operations to be conducted under AOA(1). Use of these codes and the cross section library is accepted industry practice, and has been recognized by the NRC as generally acceptable (as in NUREG/CR-6361). However, the staff did not consider the three justifications for the adequacy of the administrative margin to be sufficient.

The first justification (*i.e.*, comparison of the USL-1 and USL-2 values) was not valid because the condition that USL-1 < USL-2 is necessary but not sufficient to demonstrate adequacy of the administrative margin. If the assumed value of Δk_m is chosen incorrectly, then the results of Method 1 can demonstrate that it is not conservative. This condition is not sufficient to show the adequacy of Δk_m because Method 2 cannot be used to justify the appropriateness of inputs for Method 1. Methods 1 and 2 are different statistical treatments of the same data, and can only account for uncertainties that are statistical in nature. Because the administrative margin may contain allowances for non-statistical uncertainties associated with calculating k_{eff}, a comparison of the two methods is necessary but not sufficient to show the adequacy of the administrative margin.

The second justification (*i.e.*, historically, an administrative margin of 0.05 has commonly been applied by the NCS community) was not valid because that statement in the USLSTATS NUREG was only referring to using USLSTATS in the application of transportation of spent light water reactor uranium fuel. That application is not the same as the fabrication of MOX fuel nor is it the same as the specific application of AOA(1). In addition, the three types of NRC licensees and certificate holders fuel cycle facilities (i.e., gaseous diffusion plants, high enriched uranium fuel fabrication facilities, and low enriched uranium fuel fabrication facilities) have different administrative margins between facilities and have different administrative margins for different conditions at the same facility. The value of the administrative margin agreed to by both the facility and the NRC is based in part on the risk of the operation and performance of the licensee or certificate holder. A MOX fuel fabrication facility is different from any of those types of facilities because of the presence of plutonium and AOA(1) is not an operation at any of those other facilities. NUREG/CR-6361, Section 4.1.1, "USL Method 1: Confidence Band with Administrative Margin," describes a Δk_m of 0.05 as the recommended minimum (for applications involving transportation and storage of light water reactor fuel), but does not state that this is adequate.

The third justification (i.e., NUREG-1718 recommended an administrative margin of 0.05) was not valid because NUREG-1718 does not state any value for the administrative margin in the Acceptance Criteria. NUREG-1718, Section 6.4.3.3.4, "Requirements of Proposed 10 CFR 70.61 (Subcriticality of Operations and Margin of Subcriticality for Safety)," states that a margin of 0.05 is "typically considered acceptable for most cases" that are statistically well-represented, but that the applicant should justify the administrative margin chosen.

NRC staff determined that the applicant had not adequately demonstrated that the use of an administrative margin of 0.05 for AOA(1) was acceptable. NRC staff determined that the applicant was required to provide an adequate justification for the value of an administrative margin for AOA(1), which consists of relatively high risk plutonium solution systems. The NRC provided guidance to the applicant summarizing the regulatory requirements for other NRC-regulated facilities for the maximum allowed k_{eff}, in a letter dated November 9, 2001. The applicant submitted a complete revision to Part I as well as Part II to the Validation Report on December 11, 2001. The NRC staff has not yet completed its review of the Validation Report submitted on December 11, 2001, which contains additional justifications of administrative margins for each of the first four AOAs. While the review is in progress, the staff has previously identified concerns with the administrative margin for AOA(1), as discussed above. Therefore, determination of design basis USLs for AOA(1) has not been adequately resolved and is considered an open item.

Additional guidance regarding preparing an NCS Validation Report and the elements of that report can be found in NUREG-1718, Section 6.4.3.3.1, "Analytical Methodology." Therefore, justification of the administrative margin has not been adequately resolved and is considered an open item.

6.1.3.5.3 AOA(2): MOX Pellets, Fuel Rods, and Fuel Assemblies

The applicant also used the methods of NUREG/CR-6361 (i.e., calculation of USL-1 and USL-2 using the computer code USLSTATS) to determine the USL for AOA(2). To address AOA(2), the applicant selected five sets of mixed uranium/plutonium benchmark experiments from the ICSBEP-Handbook (Nuclear Energy Agency 1999). These five sets included 36 critical experiments performed with lattices of MOX fuel rods in water with various concentrations of plutonium and moderating ratios (v^m/v^f). The staff evaluated the applicant's defined AOA against the chosen benchmark experiments to determine that the range was appropriate. The applicant used the SCALE4.4 code system with the 238-group cross-section library, and the 36 critical experiments were modeled for the calculation of k_{eff} using KENO VI.

The staff compared the range of parameters covered by the benchmark data to the applicant's AOA and determined that there were a sufficient number of experiments and that they provided adequate coverage of the defined AOA.

The applicant analyzed the 36 critical experiments and determined the USL for AOA(2). The applicant calculated USL-1 and USL-2 with the USLSTATS computer code, as discussed in NUREG/CR-6361. The minimum USL-1 with a 0.05 administrative margin was 0.9350 for trending with EALF from 0.08 to 0.91, and v^m/v^f ratio from 0.91 to 10.75. The minimum USL-2 for the experiments was 0.9711. The applicant concluded that the results showed that the administrative margin applied to the USL-1 value was adequate for the AOA(2) application, as long as the EALF and v^m/v^f ratios fall within the applicable range.

The applicant also analyzed the 36 critical experiments as a function of the PuO_2/ (UO_2+PuO_2) ratio to determine the effect of the PuO_2 content on the calculational bias. The calculations covered a range between 1.5 wt. percent and 6.6 wt. percent in $PuO_2/(UO_2+PuO_2)$. The calculations show an increase in the USL-1, and USL-2 values with the increase of the PuO_2 proportion. With an administrative margin of 0.05, the minimum value for USL-1 is 0.9386, and 0.9753 for USL-2. The applicant determined on this basis that the 0.05 administrative margin was appropriate.

The applicant stated that the benchmark experiments for clad fuel pellets were applicable to the production of fuel assemblies and configurations involving loose rods. The applicant also stated that these benchmarks are also applicable to unclad fuel pellets or loose pellets because the cladding effects due to neutron absorption in the epithermal and thermal regions are negligible. The applicant's analysis showed that the cladding configuration (i.e., material, position, thickness) changes the epithermal and thermal neutron flux distribution by less than 1 percent at the surface of the pellet. This effect in k_{eff} was on the order of the KENO variance, and the AOA for clad fuel pellets, fuel assemblies, and loose rods is, therefore, directly applicable to unclad pellets.

The staff reviewed the applicant's calculations and, therefore, agrees that the cladding effects are negligible and may be ignored. Pellet configurations and application calculations within the AOA range for clad rods were, therefore, directly applicable to pellets and also validated. In addition, the staff ran a number of the applicant's benchmark cases and obtained good agreement with the applicant's results.

The staff concluded that the applicant's use of the NUREG/CR-6361 methodology (including USLSTATS) and the SCALE 4.4 code system was appropriate. Use of these codes and the cross section library is accepted industry practice, and has been recognized by the NRC as generally acceptable (as in NUREG/CR-6361). The applicant submitted the same justification for the administrative margin for AOA(2) as for AOA(1). For the reasons stated in Section 6.1.3.5.2, these justifications are invalid. However, upon completion of its review, the staff has determined that an administrative margin of 0.05 is adequate to provide reasonable assurance of subcriticality for the reasons stated below.

First, the risk of operations involving MOX fuel pellets, rods, and assemblies is very low relative to the risk of other proposed operations at the MFFF. Fuel covered by this AOA will have a plutonium content of ~6.3wt percent Pu, and thus is much less sensitive to changes in system parameters than the other AOAs. In terms of critical masses and dimensions, these systems are more comparable to low-enriched uranium (LEU) systems than high-enriched uranium (HEU) systems. As shown in the table of DSER Reference 6.3.8, the NRC has historically allowed a smaller margin for lower-risk LEU operations than for HEU operations. In addition, the neutronic characteristics of MOX fuel must be well-known and has been well-studied in order to enable it to be used as reactor fuel. The configuration of this material is fixed, and therefore is not readily subject to change (which translates both into a well-known and tightly-controlled configuration and lower risk). Moreover, the chosen benchmarks have configurations unusually similar to those anticipated for the design applications (i.e., MOX fuel rod lattices), which provides a high degree of certainty that the code can accurately model these systems. These benchmarks are all well-moderated thermal systems, and the neutron cross sections in the thermal energy range have been measured with a high degree of accuracy.

Therefore, based on the inherent low risk posed by MOX fuel pellets, rods, and assemblies; the historical precedent for facilities with similar qualitative risk; the well-characterized nature of the fuel; similarity of the design applications to benchmark experiments; and well-understood physics of the applicable cross sections in the thermal range, the staff finds the proposed benchmark experiments, defined area of applicability, and methodology (including the use of administrative margin of 0.05) acceptable for construction authorization.

6.1.3.5.4 AOA(3) and AOA(4): Plutonium Oxide and MOX Powders

The applicant submitted Part II of the MOX Validation Report on December 11, 2001, for AOA(3) and AOA(4). The staff is in the process of reviewing Part II to determine the adequacy of the design basis USLs for AOA(3) and AOA(4). The staff has not completed its review of these areas as of the date of this DSER.

6.1.3.6 NCSE Criticality Controls

The applicant stated that the criticality controls credited in NCSEs will be identified and evaluated during development of the ISA, and will later be identified as IROFS in its ISA summary to be submitted as part of its application for a 10 CFR Part 70 operating license. Since the specific controls relied on as NCS IROFS are not design information, the identification of such controls is not needed to make a regulatory determination for construction, and therefore this approach is acceptable to the staff.

6.1.4 Design Bases of the PSSCs

6.1.4.1 Description of PSSCs

Section 70.23(b) of 10 CFR requires in part, that before approving construction of the MFFF, the NRC must find that “design bases of the principal structures, systems, and components” provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents. CAR Section 6.4, “Design Bases,” states that the “[p]rincipal SSCs are described in Chapter 5 of this document. Specific IROFS associated with criticality safety will be identified in the ISA.” However, CAR Section 5.6.1, “Description of Principal SSCs and Required Support Systems,” states that PSSCs are described in CAR Chapter 6. Thus, the CAR does not identify and specific PSSCs as being relied on for criticality safety. CAR Table 5.6-1, “MFFF Principal SSCs,” does identify “criticality control” as the PSSCs for criticality hazards. Thus, the

CAR does not describe any specific PSSCs credited for preventing criticality as is stated in CAR Chapters 5 and 6.

Because the applicant can choose any criticality control methods described in this CAR Chapter to prevent criticality, there are no specific PSSCs for nuclear criticality safety (the criticality accident alarm system required in 10 CFR 70.24 has not been identified as a PSSC). However, the criticality controls must be sufficiently reliable and available to adequately provide for criticality control and comply with the DCP (10 CFR 70.64(a)(9)). While specific controls have not been identified, the controls that will be chosen during performance of the ISA must further meet the commitments to technical practices for each controlled parameter as discussed in CAR Chapter 6. The dominant controlled parameters have been included in CAR Tables 6-1 and 6-2 and the staff finds that these parameters comply with the preferred control hierarchy. Therefore, based on these regulatory requirements and the commitments contained in CAR Chapter 6 (reviewed in DSER Section 6.1.3.3), the staff has reasonable assurance that the design bases of the PSSCs (with the exception of the open items in this DSER) will provide adequate protection against the consequences of a criticality accident (as required by 10 CFR 70.23(b)), even though the specific IROFSs will not be defined until performance of the ISA. Identification of the PSSCs as “criticality control” is defined at a sufficiently detailed level to support the construction authorization, given the design bases reviewed in this chapter of the DSER.

6.1.4.2 Commitment to DCP

The staff considers the commitment to the DCP, including the corresponding definition of “unlikely”, as part of the design basis. The DCP requires (ANSI/ANS-8.1-1983 (R1988)) that “at least two unlikely, independent, and concurrent changes in process conditions” must occur before criticality is possible. The staff considers the commitment to the DCP generally acceptable, but in order to provide reasonable assurance that the design will result in controls that are sufficiently reliable and available when needed to perform their safety function, the meaning of “unlikely” for each contingency must be unambiguously defined. The definition of “unlikely” for meeting the DCP has been resolved, as discussed in DSER Section 6.1.4.3.

CAR Section 5.4.3, “Likelihood Definitions,” states that “...application of the DCP and/or single-failure criteria (in accordance with traditional engineering methods) is sufficient to satisfy the performance requirements of 10 CFR 70.61.” (Note: This statement applies only to 70.61(b).) The applicant has committed to ensure that criticality is “highly unlikely” as specified in 10 CFR 70.61(b), in performing its ISA Summary to be submitted as part of its application for a 10 CFR Part 70 operating license. In CAR Section 5.4.3, the applicant has stated that this commitment is also sufficient to meet 10 CFR 70.64(a)(9). The staff has determined that the definition of the DCP (taken from ANSI/ANS-8.1-1983 (R1988)) is insufficiently detailed to support the conclusion that criticality is necessarily “highly unlikely” to occur.

Moreover, the definition of “highly unlikely” in CAR Section 5.4.3 (“Events originally classified as Not Unlikely or Unlikely to which sufficient principal SSCs are applied to further reduce their likelihood to an acceptable level”) is circular and, therefore, unacceptable. This definition does not define what is meant by an “acceptable level” or what is meant by “sufficient” PSSCs. In DSER Reference 6.3.2, RAI 39, the applicant committed to an index likelihood method for determining likelihoods for hazards other than criticality. The applicant has not, however, committed to the index or an analogous method for criticality hazards. The staff provided the applicant with additional guidance concerning the relationship between being “highly unlikely” and meeting the DCP, by letter dated December 5, 2001. In the public meeting on March 27, 2002, the applicant presented a summary description of a methodology for determining acceptable likelihoods for criticality accidents. While the method appears to approximately agree with the appropriate acceptance criteria, the staff requested that a more detailed description of the plan be submitted. The applicant has not yet submitted this description. Therefore, the definition of "highly unlikely," and the appropriate level of protection against accidental criticality in 10 CFR 70.64, has not been adequately resolved and is considered an open item.

6.1.4.3 Commitment to ANSI Standards

The staff reviewed the applicant's commitment to the ANSI/ANS-8 series consensus standards in criticality safety, as described in CAR Section 6.4 and theDSER Reference 6.3.2. In CAR Section 6.4, the applicant commits to the use the ANSI/ANS-8 series standards endorsed in Regulatory Guide (RG) 3.71 in the design of the MFFF. In DSER Reference 6.3.2, RAI 90, concerning the specific provisions of these standards being committed to, the applicant stated that it was, in general, committing to all of the requirements ("shall" statements) and recommendations ("should" statements) in the applicable standards, but identified several clarifications to specific commitments within certain of the standards. Moreover, RG-3.71 endorses several of these standards conditionally. In those cases in which the regulations or regulatory guidance disagreed with the standard, the applicant was requested to clarify or modify the commitment. Specific commitments to these standards are summarized below:

- **ANSI/ANS-8.1-1983, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors":** The applicant commits to comply with the requirements of ANSI/ANS-8.1-1983 (R1988) and implement the recommendations, with clarification of three provisions: (1) in Section 4.2.2 of the ANSI standard, the applicant commits to follow the DCP, which requires that at least two unlikely, independent, and concurrent changes in process conditions must occur before criticality is possible. For the purposes of meeting this commitment, "unlikely" is defined in DSER Reference 6.3.2, RAI 90, as "events or event sequences that are not expected during the facility lifetime, but are considered credible." This commitment will be met for those processes and areas in which criticality is determined to be credible. Staff notes that assessment of credibility will be determined during the performance of the ISA (covered in CAR Chapter 5). Staff further notes that a definition of "unlikely" that is qualitatively consistent with a probability of failure on the order of 10^{-2} per year is considered to be acceptable in NUREG-1718. The phrase "not expected during the lifetime of the facility" is, therefore, acceptable provided the lifetime of the facility is assumed to be greater than approximately 100 years.[4] In DSER Reference 6.3.7, the applicant clarified that while this is a qualitative determination, it is consistent with a failure probability on the order of once in 100 years. Therefore, this is acceptable to the staff. (2) In Section 4.2.3 of the ANSI standard, the applicant commits to follow the standard (which lists several different types of control methods including engineered and administrative controls), but commits to relying on engineered features whenever practical and to justify the use of administrative controls. This is consistent with the preferred design approach and therefore acceptable to the staff. (3) In Section 4.3.2 of the ANSI standard, the applicant committed that where a large extension to the area(s) of applicability is required, the calculational method will be supplemented by other calculational

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[4]The term "unlikely" as used in the DCP should not be confused with the term "unlikely" in the performance requirements of 10 CFR 70.61(c) for intermediate-consequence events. SRP Appendix A (p. A-6) attaches a numerical value of 4×10^{-4}/yr to the 10 CFR 70.61(c) definition of "unlikely."

methods to provide a better estimate of bias in the extended area(s), or through an increase in the margin of subcriticality. Where a large extension is made beyond the range of available benchmark data, the approved margin of subcriticality must account for the increased uncertainties. The applicant did not define what was meant by a “large” extrapolation to the AOA.

In DSER Reference 6.3.3, the applicant committed that where there is *any* extension to the area(s) of applicability, the method will be supplemented by other calculational methods or other justification, or the margin of subcriticality will be increased. NUREG-1718, Section 6.4.3.3.1, “Analytical Methodology,” states that “any extrapolation of the AOA beyond the physical range of the data should be supported by an established mathematical methodology.” ANSI/ANS-8.1-1983, Section 4.3.2, states that where the extension to the AOA is large, “the method should be supplemented by other calculational methods to provide a better estimate of the bias in the extended area(s).” The clarification that the methods prescribed in ANSI/ANS-8.1-1983 would be used for *any* extension of the AOA is acceptable. However, in DSER Reference 6.3.6, the applicant stated that the justification would be provided in the specific calculations and NCSEs. Since the ANSI/ANS-8 additional standards are considered part of the design basis, this does not constitute an acceptable commitment. SRP Section 6.4 states that any variations from the requirements of a standard should be justified in the application. Therefore, what is meant by “other justification” has not been adequately resolved and is considered an open issue.

While the applicant is not required to meet 10 CFR 70.61 at the construction authorization stage, the standard requires that all operations be subcritical under both normal and credible abnormal conditions.

With regard to the subcritical limits in ANSI/ANS-8.1-1983 (R1988), the staff considers the use of subcritical limits from the standard in lieu of explicit calculation to be an acceptable practice. These results have been endorsed in RG-3.71 and as single parameter limits, are very conservative. In addition, since the standard was merely reaffirmed in the 1998 version, the staff considers the use of the 1998 version acceptable.

- **ANSI/ANS-8.3-1997, “Criticality Accident Alarm System”:** The applicant commits to comply with the requirements of ANSI/ANS-8.3-1997 and implement the recommendations, with clarification of two provisions: (1) in Section 4.1.3 of this standard, the applicant commits to evaluate the overall risk to personnel, specifically with respect to the risk from operational interruption and relocation following false alarms. Staff considers it appropriate to consider overall risk, but notes that 10 CFR 70.24(a) requires a criticality alarm in all areas of the facility where more than the specified quantities of fissionable materials are handled or stored. In DSER Reference 6.3.3, RAI 74, the applicant committed to identify and justify, in the license application, any specific areas for which the criticality alarm exemption was being sought. The justification should be included in any request for criticality alarm exemption (such as if the overall risk would be minimized by not having criticality alarms). (2) In Section 5.3 of this standard, the applicant commits to design the criticality alarm system to remain operational in the event of the site-specific design basis earthquake. Since this is the earthquake that the facility as a whole will be designed to withstand, this is acceptable to the staff.

  Staff notes that RG-3.71 endorses ANSI/ANS-8.3-1997 with exceptions required by the provisions of 10 CFR 70.24(a). The staff, therefore, requested and received clarification that where the standard disagrees with 10 CFR 70.24, the applicant will follow the rule (unless an exemption request is submitted and granted pursuant to 10 CFR 70.17).

- **ANSI/ANS-8.7-1975, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials":** The applicant commits to comply with the requirements of ANSI/ANS-8.7-1975 and implement the recommendations, with clarification of two provisions: (1) In Section 4.2.4 of ANSI/ANS-8.7-1975, the applicant commits that the design of storage structures will rely on engineered instead of administrative control to the extent practical, and that the use of administrative controls will be justified. This is consistent with the preferred design approach and, therefore, acceptable to the staff. (2) In Section 4.2.6, the applicant commits to administrative controls to limit combustible loading for fire protection, and to justify fire protection provisions in all fissile material processing, handling, or storage areas. This approach is acceptable to the criticality staff since this mainly affects fire protection. Wherever a fire or initiation of fire protection measures (including engineered systems and administrative responses) is considered credible, the effects on the system for criticality purposes should be evaluated.

  With regard to the subcritical limits in ANSI/ANS-8.7, the staff considers the use of subcritical limits from the standard in lieu of explicit calculation to be an acceptable practice. These results have been endorsed in RG-3.71 and as single parameter limits, are very conservative.

  Staff notes that the applicant has committed to the 1998 version of the standard, whereas RG-3.71 endorses the 1975 version. NUREG-1718 Section 6.4.2, "Regulatory Guidance," states that the responsibility for demonstrating the acceptability of unendorsed standards rests with the applicant. The staff, therefore, requested that the applicant identify the differences between the endorsed and unendorsed version of this ANSI/ANS-8.7 and justify those portions of the 1998 version which differ from the 1975 version. In its December 5, 2001, letter, the applicant committed to providing a justification to the NRC demonstrating the acceptability of the more recent version of the standard, if it is to be referenced as guidance for the MFFF. This approach is consistent with NUREG-1718 Section 6.4.2; actual determination of the acceptability of the more recent version of any of these standards is contingent on NRC review of this justification.

- **ANSI/ANS-8.9-1975, "Guide for Nuclear Criticality Safety for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Materials":** The applicant states that ANSI/ANS-8.9-1975 has been withdrawn by the ANS-8 working group, and will not be used in the design of the MFFF. Piping configurations containing aqueous solutions of fissile material will be evaluated by calculation in accordance with ANSI/ANS-8.1-1983 (R1988). Because using validated methods to determine subcritical limits is an acceptable methodology, the staff determined that this approach was acceptable.

- **ANSI/ANS-8.10-1983, "Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement":** The applicant states that the requirements of ANSI/ANS-8.10-1983 are, in general, not applicable to the design of the MFFF, since the approach used for the MFFF is to prevent criticality in accordance with the DCP rather than rely on shielding and confinement for dose mitigation. However, the applicant states that this standard may be referenced in crediting shielding for compliance with the performance requirements of §70.61(b) during the design stage. While the applicant is required to comply with the DCP and rely on prevention rather than mitigation, the staff also considers it appropriate to consider the effects of shielding when evaluating maximum doses in 10 CFR 70.61 during the design stage;

for high-consequence events, the rule allows the applicant to either establish controls to ensure the event is highly unlikely, or reduce the consequences to less than those in the high-consequence category. The staff, therefore, considers this application of ANSI/ANS-8.10-1983 appropriate.

- **ANSI/ANS-8.12-1987, "Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors":** With regard to the subcritical limits in ANSI/ANS-8.12-1987, the staff considers the use of subcritical limits from the standard in lieu of explicit calculation to be an acceptable practice. These results have been endorsed in RG-3.71 and as single parameter limits, are very conservative. As with other standards that primarily contain subcritical limits, this standard does not contain any administrative requirements to which the applicant should commit. The staff notes that this standard only contains subcritical limits for certain plutonium-uranium mixtures; however, the requirement to ensure subcriticality under normal and credible abnormal conditions requires that when characteristics of MOX fuel do not exactly meet the compositions that are evaluated in the standard, conservative limits from the standard should be used or the system should be evaluated by other accepted means (e.g., process-specific calculations should be performed or other references used.

- **ANSI/ANS-8.15-1981, "Nuclear Criticality Control of Special Actinide Elements":** The applicant states that although the MFFF processes will contain low levels of special actinide materials, their concentration will always be "relatively low" compared with the ^{235}U or ^{239}Pu concentration, and therefore, ANSI/ANS-8.15-1981 will not be used in the facility design. Instead, explicit calculations will be performed or reference will be made to the subcritical limits in ANSI/ANS-8.1-1983 (R1988). However, what was meant by "relatively low" in terms of being sufficiently low to permit application of the limits of ANSI/ANS-8.1 was not defined. In DSER Reference 6.3.3, the applicant removed the statement about the "relatively low" concentration of special actinide elements, but did not clarify the criteria for applying the limits of ANSI/ANS-8.1. This was also not clarified in DSER Reference 6.3.6. Therefore, the applicability of ANSI/ANS-8.1 limits to mixtures involving special actinide elements at the MFFF has not been adequately resolved and is considered an open issue.

- **ANSI/ANS-8.17-1984, "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors":** The applicant commits to comply with the requirements of ANSI/ANS-8.17-1984 and implement the recommendations, with clarification of two provisions: (1) For Section 4.11, the applicant commits to use of the DCP for the handling, storage, and transportation of fuel units and rods. Compliance with the DCP is required for the MFFF by 10 CFR 70.64(a)(9) and thus this commitment is acceptable to the staff. (2) For Section 5.1, the applicant committed that where a large extension to the area(s) of applicability is required, the calculational method will be supplemented by other calculational methods to provide a better estimate of bias in the extended area(s), or through an increase in the margin of subcriticality. Where a large extension is made beyond the range of available benchmark data, the approved margin of subcriticality must account for the increased uncertainties. The applicant did not define what was meant by a "large" extrapolation to the AOA.

  In DSER Reference 6.3.3, the applicant committed that where there is *any* extension to the area(s) of applicability, the method will be supplemented by other calculational methods or other justification, or the margin of subcriticality will be increased. NUREG-1718, Section 6.4.3.3.1, "Analytical Methodology," states that "any extrapolation of the AOA beyond the physical range of the data should be supported by an established mathematical methodology."

ANSI/ANS-8.1-1983, Section 4.3.2, states that where the extension to the AOA is large, "the method should be supplemented by other calculational methods to provide a better estimate of the bias in the extended area(s)." The clarification that the methods prescribed in ANSI/ANS-8.1-1983 would be used for *any* extension of the AOA is acceptable. However, in DSER Reference 6.3.6, the applicant stated that the justification would be provided in the specific calculations and NCSEs. Since the ANSI/ANS-8 additional standards are considered part of the design basis, this does not constitute an acceptable commitment. SRP Section 6.4 states that any variations from the requirements of a standard should be justified in the application. Therefore, what is meant by "other justification" has not been adequately resolved and is considered an open issue. Therefore, what is meant by "other justification" has not been adequately resolved and is considered an open issue.

- **ANSI/ANS-8.19-1996, "Administrative Practices for Nuclear Criticality Safety":** The applicant commits to comply with the requirements of ANSI/ANS-8.19-1996 and implement the recommendations, with the exception that no commitments are made to Section 10 regarding emergency response to criticality accidents. ANSI/ANS-8.19 is currently undergoing review by the working group and has the potential to be changed in the near term. However, staff notes that the applicant has committed to ANSI/ANS-8.23-1997, which contains many of the same requirements.

  Additionally, the staff notes that emergency response procedures are not at issue in deciding whether to approve the CAR. These procedures will be evaluated if DCS submits a license a license application. The staff, therefore, recommends deferring review of ANSI/ANS-8.19 commitments in this area until the standard is reaffirmed or revised.

- **ANSI/ANS-8.20-1991, "Nuclear Criticality Safety Training":** The applicant commits to comply with the requirements of ANSI/ANS-8.20-1991 and implement the recommendation without exception or clarification. This standard has been endorsed by RG-3.71, and therefore, this commitment is acceptable to the staff.

- **ANSI/ANS-8.21-1995, "Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors":** The applicant commits to comply with the requirements of ANSI/ANS-8.21-1995 (this standard contains no recommendations) without exception or clarification. This standard has been endorsed by RG-3.71, and therefore this commitment is acceptable to the staff.

- **ANSI/ANS-8.22-1997, "Nuclear Criticality Safety Based on Limiting and Controlling Moderators":** The applicant commits to comply with the requirements of ANSI/ANS-8.22-1997 and implement the recommendations, with clarification of two provisions: (1) for Section 4.1.7, the applicant commits to administrative controls to limit combustible loading for fire protection, and to justify fire protection provisions in all fissile material processing, handling, or storage areas. This approach is acceptable to the criticality staff since this mainly affects fire protection, although the effects on criticality safety of a fire or initiation of fire protection measures (including both engineered systems and administrative responses) should be evaluated. (2) For Section 5.4.1, the applicant stated that wherever fire suppression is to be used in moderator control areas, the use of non-moderating fire suppressants will be considered. However, CAR Section 6.3.3.2.6 states "in the MFFF moderation-controlled areas, hydrogenous fire-fighting materials are not allowed." The staff identified this as an inconsistency in the October 11, 2001, public meeting.

In its December 5, 2001, letter, the applicant removed the clarification to Section 5.4.1 of ANSI/ANS-8.22-1997. The original Section 5.4.1 stated, with regard to this issue, that only the "use of non-moderating fire suppressant media should be considered." Based on the commitment in CAR Section 6.3.3.2.6, the staff understands that moderating fire suppressant media will not be used in moderation-controlled areas. This approach to criticality prevention in moderation-controlled areas is acceptable to the criticality staff; the adequacy of fire suppressant media will be reviewed in accordance with NUREG-1718 Chapter 7.

- **ANSI/ANS-8.23-1997, "Nuclear Criticality Accident Emergency Planning and Response":** The applicant commits to comply with the requirements of ANSI/ANS-8.23-1997 and implement the recommendations without exception or clarification. This standard has been endorsed by RG-3.71, and therefore, this commitment is acceptable to the staff. Additionally, the staff notes that DCS, if later authorized to operate the MFFF, would be subject to the applicable criticality accident requirements stated in 10 CFR 70.24.

- **Additional ANSI/ANS-8 Series Standards**: In addition to those standards to which the applicant has committed above, other criticality safety standards are referenced in NUREG-1718, as follows:

    1.) ANSI/ANS-8.5-1996, "Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material."

    2.) ANSI/ANS-8.6-1983 (R1987), "Safety in Conducting Subcritical Neutron-Multiplication Measurements In Situ."

As indicated in DSER Reference 6.3.2, RAI 77, the applicant does not envision using raschig rings for criticality control in MFFF operations, but will instead rely only on fixed neutron absorbers in accordance with ANSI/ANS-8.21-1995. The applicant also does not intend to conduct subcritical neutron multiplication measurements at the facility. Therefore, the staff concurs that commitments to these two standards are not applicable to the design of the MFFF and, therefore, are unnecessary.

6.2 EVALUATION FINDINGS

In Section 6.4 of the CAR, DCS provided design basis information for nuclear criticality safety PSSCs that it identified for the MFFF. Based on that the staff's review of the CAR and supporting information provided by the applicant relevant to nuclear criticality safety, the staff finds that, due to the open items discussed above and listed below, DCS has not met the BDC set forth in 10 CFR 70.64(a)(9). Further, until the open items are closed, the staff cannot conclude, pursuant to 10 CFR 70.23(b), that the design bases of the PSSCs identified by the applicant will provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents.

The results of the assessment were found to be acceptable with the exception of the following open items:

- A discussion of the specific Pu/MOX experience for NCS staff involved in the design phase (DSER Section 6.1.1) (NCS-1).

- Justification for the bounding density values assumed in Tables 6-1 and 6-2 (DSER Sections 6.1.3.4.1 and 6.1.3.4.2) (NCS-3).

- Definition of NCS design basis controlled parameters for AP and MP process auxiliary systems (specifically including process ventilation, isotopic dilution, and high-alpha waste) (DSER Section 6.1.3.4.1) (NCS-2).

- Determination of design basis USLs for each process type, and justification for the administrative margin (DSER Section 6.1.3.5.2); description of sensitivity methods to be provided in Part III of the Validation Report (DSER Section 6.1.3.5) (NCS-4).

- The definition of “highly unlikely,” and how the performance requirements of 10 CFR 70.61(b) will be met for criticality hazards (DSER Section 6.1.4.2) (NCS-5).

- For ANSI/ANS-8.1-1983 (R1988), what is meant by “other justification” in the means for extending the code’s area(s) of applicability beyond experimental data (DSER Section 6.1.4.3) (NCS-6).

- For ANSI/ANS-8.15-1981, the applicability of ANSI/ANS-8.1 limits to mixtures involving special actinide elements at the MFFF (DSER Section 6.1.4.3) (NCS-7)

- For ANSI/ANS-8.17-1984, what is meant by “other justification” in the means for extending the code’s area(s) of applicability beyond experimental data (DSER Section 6.1.4.3) (NCS-8)

6.3 REFERENCES

6.3.1 Giitter, J.G., U.S. Nuclear Regulatory Commission, letter to P.S. Hastings, Duke Cogema Stone & Webster, RE Mixed Oxide Fuel Fabrication Facility Construction Authorization - Request for Additional Information, June 21, 2001.

6.3.2 Hastings, P.S., Duke Cogema Stone & Webster, letter to U.S. Nuclear Regulatory Commission, RE Response to Request for Additional Information - Construction Authorization Request, August 31, 2001.

6.3.3 Hastings, P.S., Duke Cogema Stone & Webster, letter to U.S. Nuclear Regulatory Commission, RE Clarification of Responses to NRC Request for Additional Information. December 5, 2001.

6.3.4 Hastings, P.S., Duke Cogema Stone & Webster, letter to U.S. Nuclear Regulatory Commission, RE Clarification of Responses to NRC Request for Additional Information, January 7, 2002.

6.3.5 Hastings, P.S., Duke Cogema Stone & Webster, letter to U.S. Nuclear Regulatory Commission, RE Clarification of Responses to NRC Request for Additional Information, February 11, 2002.

6.3.6 Hastings, P.S., Duke Cogema Stone & Webster, letter to U.S. Nuclear Regulatory Commission, RE Clarification of Responses to NRC Request for Additional Information, March 8, 2002.

6.3.7 Persinko, A., U.S. Nuclear Regulatory Commission, letter to P.S. Hastings, Duke Cogema Stone & Webster, RE Duke Cogema Stone & Webster Quality Assurance Program for Construction of the Mixed Oxide Fuel Fabrication Facility, October 1, 2001.

6.3.8 Persinko, A., U.S. Nuclear Regulatory Commission, letter to P.S. Hastings, Duke Cogema Stone & Webster, RE Nuclear Criticality Safety Staff Qualifications and Administrative Margins for Fuel Fabrication Facilities, November 9, 2001.