6.0 NUCLEAR CRITICALITY SAFETY

6.1 CONDUCT OF REVIEW

This chapter of the revised 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 revised 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 or the facility) 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 revised CAR, other sections of the revised 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 Basis (see Chapter 5.0 of this revised DSER), and the review of other plant systems.

The staff reviewed how the NCS information in the revised 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 facility:

- 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 70.61(d) of 10 CFR requires that the risk of nuclear criticality accidents be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety. Preventive controls and measures must be the primary means of protection against nuclear criticality accidents.

- 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 bases 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, Sections 6.3.1 and 6.5.1, state that staff should review the applicant's commitment to establish organization and administration methods to determine whether the applicant has identified the responsibilities and authorities for organizations and individuals implementing the NCS Program. For construction authorization, the design bases include a description of the roles and responsibilities of the NCS Function involved in the design of the facility, and NCS staff education and experience levels. Certain aspects of the NCS Program would only be relied on during operation of the facility–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) need not be 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 facility is licensed to operate.

Revised CAR Section 6.1.1 contains information relating to the organization and administration of the NCS Program during the design phase of the proposed facility, including a description of its relevant roles and responsibilities. The roles and responsibilities during design basically consist of establishing design criteria, supporting the integrated safety analysis (ISA) process, validating calculational methodologies, performing analyses, and designing criticality control systems. NUREG-1718, Section 6.4.3.1, "Organization and Administration," states that the NCS Function should be independent of operations to the extent 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 proposed facility.

Revised CAR Section 6.1.2 deals with the NCS Program during the operations phase, which is not part of the design basis of the facility.

Commitments to specific NCS American National Standards Institute/American Nuclear Society (ANSI/ANS) -8 standards (ANSI/ANS-8.1-1983 (R1988) and ANSI/ANS-8.19-1996) are discussed in the revised DSER Section 6.1.4, "Design Bases of the PSSCs."

In revised DSER Reference 6.3.9, as amended by Reference 6.3.12, the applicant provided qualifications for key NCS positions during design. The staff evaluated the proposed qualifications (education and experience levels) for the NCS Function Manager, Senior NCS Engineer, and NCS Engineer. The staff concluded that the education levels for these positions were acceptable based on a knowledge of comparable education requirements at other fuel facilities.

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. Moreover, comparable industry experience is likely to consist of experience at low- and high-enriched uranium fuel fabrication and enrichment plants. The staff also recommended that plutonium (Pu) or MOX-specific experience be sought for at least senior NCS positions. In revised DSER Reference 6.3.11, the applicant provided a summary of its criticality safety experience base involved in the design work. This included significant participation by COGEMA and SGN staff with over twenty years experience with plutonium and MOX processing at MELOX and LaHague, as well as staff with experience in the domestic nuclear industry. In addition, the applicant has committed to training the criticality staff in the appropriate characteristics of plutonium and MOX materials. This commitment, combined with the significant in-house

experience with criticality safety associated with plutonium and MOX processing should ensure a sufficient knowledge base suitable for the design of the facility. Therefore, the staff finds the qualifications and training requirements for new NCS staff, and the experience of current NCS staff, adequate for the design of the facility.

6.1.2 Management Measures

NUREG-1718, Sections 6.3.2 and 6.5.1, state that staff should review the applicant's commitment to establish 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 IROFS. The applicant is not required to identify specific IROFS in the revised CAR, and therefore, the specific management measures to be applied to them cannot yet be specified. However, the staff reviewed the quality level definitions and the associated statements pertaining to quality levels of individual IROFS, in Section 2.2, "Graded Quality Assurance," of the applicant's MOX Process Quality Assurance Plan (MPQAP). As discussed in revised DSER Chapter 15, the staff approved rev. 2 of the MPQAP in its Safety Evaluation Report issued on October 1, 2001, and approved rev. 3 of the MPQAP in its letter to DCS dated January 10, 2003. 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 revised DSER Reference 6.3.7; questions remained, however, concerning the examples of how these quality levels would be applied in practice (i.e., application to different classes of IROFS), as discussed below.

The controlled parameters have been identified as part of the design basis of the proposed facility for NCS, but the specific controls to be relied on have not. They will be specified later as IROFS by the applicant and identified in the ISA Summary as part of the application for an SNM possession and use license (to be submitted if the revised CAR is approved). Therefore, the means of providing for criticality control (as required by 10 CFR 70.64(a)(9)) are described at the parameter level for the construction review. The only specific structures, systems, and components (SSC) addressed in the revised CAR is the criticality alarm system required by 10 CFR 70.24 (discussed in revised DSER Section 6.1.3.2). Because there are no specific controls identified, the application of specific management measures (with quality levels as approved in Section 2.2 of the MPQAP) to IROFS will be reviewed later as part of any SNM possession and use license application.

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

6.1.3 Technical Practices

NUREG-1718, Sections 6.3.3 and 6.5.1, state that staff should review the applicant's commitment to designing and operating the proposed facility in accordance with NCS technical practices, to ensure that if the facility is later authorized to operate, NCS requirements will be met. 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(d) 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 methods;

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 revised CAR section contains a description of which codes and standards the applicant is committing to. Commitments to specific ANSI standards (ANSI/ANS-8.1-1983 (R1988) and ANSI/ANS-8.19-1996) are discussed in this revised DSER, Section 6.1.4, "Design Bases of the PSSCs".

6.1.3.1 Commitment to Baseline Design Criteria

The revised CAR stated that the design of the proposed facility will adhere to the DCP as required by 10 CFR 70.64(a)(9). The DCP will be used to meet the performance requirements in an ISA Summary supporting an SNM possession and use license application. This revised CAR Section also described the process for flowing down nuclear criticality safety evaluation (NCSE) requirements into the ISA as IROFS. These commitments are part of the design basis of the facility.

Compliance with the DCP will be demonstrated in an ISA Summary supporting an SNM possession and use license application by identifying two or more process conditions which are relied on to ensure a subcritical configuration. The facility NCSEs will evaluate both normal and credible abnormal conditions. Common mode failures and system interactions will also be considered. The NCSEs will use geometry controls (with fixed neutron absorbers, as necessary) as the preferred controlled parameter for plant systems. In order to enhance the reliability of criticality controls, the following preferred hierarchy of controls 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.

Where controlled parameters rely on physical measurements, representative sampling and analysis will be used. The sampling and analysis requirements will be identified as an IROFS along with appropriate management measures. The staff concludes that these programmatic commitments provide reasonable assurance that an adequate safety basis will be established and documented.

6.1.3.2 MFFF CAAS

The staff reviewed the facility CAAS described in revised 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. Specific technical commitments regarding the CAAS are found in the discussion of ANSI/ANS-8.3-1997 in revised DSER Section 6.1.4.3.

The revised CAR 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. Each area in which alarms are required will be covered by two alarm units. In revised DSER Reference 6.3.12, the applicant committed that any areas requiring exemption from CAAS coverage would be identified and justified in the SNM possession and use license application or separate exemption request. These commitments are consistent with the requirements of 10 CFR 70.24 and ANSI/ANS-8.3-1997 (as endorsed by RG-3.71), and are therefore, acceptable.

6.1.3.3 Criticality Safety Control Design Criteria

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 proposed facility.

Revised CAR Section 6.3.3.1 defines passive and active engineered controls, and enhanced and simple administrative controls, and describes the preferred hierarchy of controls. Tolerances will be conservatively taken into account and neutron interaction fully evaluated. Revised CAR Section 6.3.3.2 discusses the criticality control modes and available methods of control to be used in the proposed facility.

Revised CAR Section 6.3.3.2, “Available Methods of Control”, describes the requirements for geometry, mass, density, isotopics, reflection, moderation, concentration, interaction, neutron absorber, volume, heterogeneity, and process variable controls. Technical practices associated with these controlled parameters were found to be acceptable, based on standard industry practice and a comparison with the acceptance criteria of NUREG-1718. Additional clarifications regarding specific controlled parameters follow:

- Revised CAR, Section 6.3.3.2.1, "Geometry Control," states that "tolerances on nominal design dimensions are treated conservatively". Expanding on this, revised DSER Reference 6.3.2 (and revised DSER Reference 6.3.14) states 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 Nuclear Regulatory Commission (NRC) staff considered the applicant’s commitment acceptable to ensure that the most reactive combination of tolerances will be used.

- Revised CAR Section 6.3.3.2.4 states that “isotopics control includes both the ^{235}U/U concentration (enrichment) and the concentration of fissile and nonfissile plutonium isotopes...as well as the relative abundance of plutonium to uranium.” The plutonium isotopics will be represented as 96wt% ^{239}Pu and 4wt% ^{240}Pu in criticality models. The revised CAR further states that “The presence of ^{240}Pu (5% to 9%) and ^{242}Pu (<0.02%) offsets any contribution from ^{241}Pu (<1%) such that it can be neglected for ^{239}Pu ranges from 90% to 95%....This will be demonstrated in the criticality calculation to be referenced in the NCSEs. Justification will be provided in NCSEs and ISA summary.” The applicant also indicated that the bounding isotopics would be sufficient to bound the slight amount (^{235}U/Pu <2wt%) of high-enriched uranium in the incoming plutonium stream, in a footnote to revised CAR Table 6-1. To confirm that the bounding assumption (96wt% ^{239}Pu, 4wt% ^{240}Pu) is more

reactive than any combination of incoming feed material, the staff performed a series of confirmatory calculations. These calculations were done using a near-critical sphere of plutonium nitrate solution with full water reflection. The isotopic abundance of ^{239}Pu, ^{240}Pu, and ^{241}Pu was varied within the specification range above (the small presence of ^{242}Pu was neglected). The calculations showed that the most reactive combination occurred when ^{240}Pu and ^{241}Pu were at their maximum values (i.e., 94wt% ^{239}Pu, 5wt% ^{240}Pu, and 1wt% ^{241}Pu). This was still less reactive than the bounding assumption. Further, the effect of increasing both ^{240}Pu and ^{241}Pu by the same amount (1wt%) was a net decrease in k_{eff} ($|\partial k/\partial^{240}\text{Pu}| > |\partial k/\partial^{241}\text{Pu}|$). Although these calculations were limited in scope, the staff considers the applicant's arguments to be plausible based on this study.

- Revised CAR Section 6.3.3.2.5 states that a minimum 1-inch tight-fitting water reflector would be assumed, with the amount of reflection justified in the NCSE. In addition, revised DSER Reference 6.3.2 (and revised DSER Reference 6.3.3.2.5) states "...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.

- Revised CAR, Section 6.3.3.2.6, "Moderation Control," states that in moderation-controlled areas, hydrogenous fire fighting materials will not be allowed. 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. The commitment to avoid moderating fire suppressants is acceptable to the criticality staff.

- Revised CAR Section 6.3.3.2.9 contains commitments regarding the use of fixed neutron absorbers. 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 facility process unit or area (see also revised DSER Section 6.1.4 on ANSI Standards). Moreover, the applicant states that wherever neutron absorber control will be used, it will be part of the geometry and fixed by design. Thus, this excludes the use of removable (raschig rings) and soluble absorbers. Because fixed neutron absorbers are inherently more reliable than removable or soluble absorbers, the NRC staff finds this acceptable.

The staff considers the information in this section of the revised CAR to be in broad agreement with the regulatory acceptance criteria in the SRP, and finds, in accordance with 10 CFR 70.23(b), that it provides an acceptable basis for design.

6.1.3.4 Criticality Safety Process Description

Revised CAR Section 6.3.4 contains a description of the applicable safety principles to be applied to the design, as well as a preliminary description of how the controlled parameters will be applied to the proposed facility. The staff reviewed this material, as well as the process descriptions in revised CAR Sections 11.2, "MOX Process Description," and 11.3, "Aqueous Polishing Process Description," along with the tables of associated criticality control units (CCUs), revised CAR Tables 6-1 and 6-2. Revised CAR Chapter 11 provides an overview of

both the MP and AP processes. Revised CAR Tables 6-1 and 6-2 list each of the 55 AP and 25 MP CCUs and their controlled parameters.

In reviewing the controlled parameters for the MOX and Aqueous Polishing processes, the staff reviewed the applicant's use of the preferred design approach. Revised CAR Section 6.3.3.1, "Criticality Control Modes," commits to the preferred use of engineered over administrative controls, and the preferred use of geometry control, where practicable. Revised CAR Section 6.3.4.2, "Applicable Safety Principles", commits to the preferred use of passive engineered over active engineered control, engineered over administrative control, and enhanced over simple administrative control, as well as dual over single-parameter control.[1] Reliance on diverse control modes is important to minimize the potential for common-mode failure. The applicant's commitment to the DCP ensures that, regardless of the parameter or type of control, at least two independent and unlikely process changes must occur before criticality is possible. However, accident sequences in which there are no credible means to achieve criticality (as may occur in certain cases involving geometry control) do not require dual independent controls to comply with the DCP.

Although specific controls (i.e., IROFS) have not been identified in the revised CAR, the design bases include the dominant controlled parameters. Specific values for these parameters were not considered necessary in most cases, although some assumed values were provided (e.g., powder density and isotopics) when they represented bounding constraints on the design. This information is presented in revised CAR Tables 6-1 and 6-2. Staff review in the following sections is based on revised CAR Tables 6-1 and 6-2, as amended by revised DSER References 6.3.10 and 6.3.12.

6.1.3.4.1 NCS - AP Process

The AP Process, as described in revised CAR Table 6-1 (as amended by revised DSER References 6.3.10 and 6.3.12) and revised CAR Section 11.3.2, consists of 16 major areas that have been subdivided into 55 CCUs. PuO_2 powder is received from offsite in cans, and then batched to the dissolution unit and/or dissolution/dechlorination 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, offgases, and liquid waste, and recycle raffinate produced in the solvent extraction. These latter units primarily rely on dual independent concentration controls as they are designed not to contain plutonium under normal process conditions.

---

[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 revised CAR sections 11.2 and 11.3, and the revised 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).

For each CCU in revised CAR Table 6-1, the dominant controlled parameters are listed along with some parameter ranges. The chemical form through most of this process is primarily aqueous plutonium oxide or nitrate solution[2], or plutonium oxalate. Plutonium oxide is considered more reactive than plutonium nitrate. (The applicant has also determined that plutonium oxalate is bounded by plutonium oxyfluoride.) The solution is converted to PuO_2 during calcination. For those areas in which plutonium nitrate is the chemical form, the applicant has identified physicochemical form as a criticality control mode. For units relying on chemical form, the applicant has committed to take into account both nominal conditions and possible process upsets in accordance with the DCP. The bounding nature of these compounds will be demonstrated in NCSEs.

According to revised CAR Table 6-1, most of the process is conducted in favorable geometry equipment, with the general geometrical shape, but not the dimensions, specified. Typically, units in the AP processes will consist of 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. This will ensure that favorable geometry units will be subcritical under normal and all credible abnormal conditions. When using methods that determine favorable dimensions for a specific material composition (chemical form, isotopics, etc.), the commitment to the double contingency principle provides assurance that process upsets capable of changing the form of the material will be considered.

Revised CAR Table 6-1 also assumes specific powder densities and isotopic compositions. The AP process downstream of the PuO_2 filter assumes a chemical form of $Pu(NO_3)_3$, which is less reactive than PuO_2. Dual independent filters preclude the presence of PuO_2 beyond this point. For powder density, different bounding values are presented. In the Decanning and Milling Units, powder density is controlled by upstream measurement by a variety of means (revised CAR Sections 11.3.2.1.2 and 11.3.2.2.2). For these units, density must be confirmed to be less than 7 g/cm^3. This is considered very high for PuO_2, due to its highly porous nature.

Downstream of the calcination furnace in the Oxalic Precipitation and Oxidation Unit, density is limited to less than 3.5 g/cm^3. These values were derived based on LaHague experience and will be confirmed during start-up testing. Revised DSER Reference 6.3.11 states that, wherever the physical form of the material changes, density will be confirmed by direct measurement “or otherwise justified”. Based on domestic nuclear industry experience, the staff accepts that a density of 7 g/cm^3 is very high for PuO_2 or MOX powder, and that the described measurement techniques are reasonable to confirm this. For lower density values, appropriate controls on density, and on any process variables that can affect powder density non-conservatively, will be specified and justified in the NCSEs. The staff will review the justification in the NCSEs supporting an SNM possession and use license application.

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. For 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

---

[2] $Pu(NO_3)_4$ is the most common nitrate, but the applicant stated it will assume $Pu(NO_3)_3$ in the analysis because it is more reactive.

with geometry control, and thus complies with the preferred hierarchy of controls and is acceptable.

The applicant proposed concentration control for those units not relying on favorable geometry, where fissile material is not expected under normal conditions. Revised CAR Table 6-1 indicates only that concentration is controlled upstream of these processes. However, a previous communication (revised DSER Reference 6.3.2, RAI 84 and revised DSER Reference 6.3.14) had stated that this would be through in-line monitors and/or sampling measurements. Revised CAR Section 6.3.3.2.7 contains the requirements for both sampling and instrumentation, including the use of dual independent sampling methods. This method for preventing plutonium intrusion into these units is in accordance with established industry practice and the DCP, and is therefore, acceptable to the staff.

Revised CAR Table 6-1 also contains a description of design bases for auxiliary systems connected to the main AP process. The Offgas Treatment Unit (KWG) removes plutonium from the gaseous effluents in the AP process. The Liquid Waste Reception Unit (KWD) treats liquid effluents, including high-alpha waste. High-enriched uranium is present in the incoming plutonium stream (assumed to be 100wt% ^{235}U). This is isotopically diluted with depleted uranyl nitrate in two steps, first to $\leq$ 30wt% ^{235}U (evaluated at 35wt% ^{235}U), and then to $\leq$ 1wt% ^{235}U, before being separated from the plutonium in the Purification Unit. Control of uranium assay is required because the high-alpha waste assay must be less than 2wt% ^{235}U for criticality safety. The applicant has identified uranium isotopics as a control mode for the downstream processes in revised CAR Table 6-1, and the staff, therefore, finds this acceptable.

For other auxiliary systems for which plutonium intrusion must be precluded (such as chemical and water addition), the revised DSER Reference 6.3.11 states that a similar approach will be used. The control methods will include passive design features and dual, independent sampling. These were not included in the revised CAR tables as they are not process units, but will be addressed in an auxiliary system NCSE. This conforms to common industry practice and is, therefore, acceptable to the staff. (Note: This will apply to auxiliary systems supporting both the AP and MP processes.)

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, there were two or more parameters identified for almost every CCU. This diversity in the control modes is desirable because it reduces the potential for common-mode failure. 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 bounding densities will be provided with any SNM possession and use license application). Coupled with the commitment to follow the preferred design approach during the design of actual IROFS, Revised CAR Table 6-1 provides reasonable assurance that the design of the AP process will be in agreement with the regulatory acceptance criteria.

6.1.3.4.2 NCS - MP

The MP, as described in revised CAR Table 6-2 (as amended by revised DSER References 6.3.10 and 6.3.12) and revised 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. 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 $\leq$ 22 wt% Pu; the final blend consists of $\leq$ 6.3 wt% Pu. These values are the conservative modeled values for maximum Pu content that are used in the criticality safety analyses. 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 latter processes, the chemical and geometric form, isotopic composition, and moderation are the dominant controlled parameters.

For each CCU in revised 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 $\leq$ 22 wt% Pu, and is stored in J60 jars; the final blend consists of MOX with $\leq$ 6.3 wt% 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 $\leq$ 1 wt% 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.

---

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

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.

Revised CAR Table 6-2 also assumes specific powder densities and isotopic compositions. For powder densities, different bounding values are presented. Powder density is controlled by upstream measurement.

There are several other units, however, where the density of powder is limited to less than 3.5 g/cm^3 or 5.5 g/cm^3, or the density of scrap to less than 4.6 g/cm^3. These values were derived based on MELOX experience and will be confirmed during start-up testing. Following pelletizing, full theoretical density (11.46 g/cm^3) is assumed for NCS purposes. Revised DSER Reference 6.3.11 states that wherever the physical form of the material changes, density will be confirmed by direct measurement “or otherwise justified”. Appropriate controls on density, and on any process variables that can affect powder density non-conservatively, will be specified and justified in the NCSEs. The staff will review the justification in the NCSEs supporting an SNM possession and use license application.

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, there were two or more parameters identified for almost every CCU. This diversity in the control modes is desirable because it reduces the potential for common-mode failure. 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, (the risk is especially low 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 IROFS, the information presented in revised CAR Table 6-2 provides reasonable assurance that the design of the MP will be in agreement with the regulatory acceptance criteria.

Although diverse control modes were described for both the AP and MP processes in revised CAR Tables 6-1 and 6-2, the extent to which diverse control modes were used in meeting the double contingency principle could not be determined. This was simply because the tables contained no information on the structure of individual accident sequences. While it appeared that dual parameter control had been implemented, during the January 16, 2003, public meeting on criticality safety, the applicant made the following statement:

> “For the MFFF, double contingency, in most cases, is based on 2 controls or barriers to prevent a change in one controlled parameter. A loss of one of these controls or barriers does not cause a change in the controlled parameter and therefore does not change the k_{eff} value.”

The applicant has committed to provide additional information regarding its commitment to the preferred use of dual parameter control and thus, this is considered an open item as part of NCS-4.

6.1.3.4.3 Single-Parameter Limits

In revised CAR Tables 6-3 and 6-4, the applicant provided a list of single-parameter limits for various parameters, such as subcritical masses, dimensions, and volumes for use in the facility design. Revised CAR Section 6.3.4.5 and the footnote to these tables indicates that they are not to be used in support for criticality calculations or NCSEs. Therefore, the staff did not evaluate the adequacy or derivation of the values in these revised CAR tables as part of the revised 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). The applicant has committed to ANSI/ANS-8.1-1983 (R1988) as part of the design basis of the facility; the standard requires that processes be shown to be subcritical under both normal and credible abnormal conditions, and contains requirements for the validation of calculational methodologies. As discussed below, the staff is analyzing the validation report submitted by the applicant on January 8, 2003, which sets forth the applicant's proposed methodology by which the margin of subcriticality for safety can be calculated. To approve the revised CAR the staff must, pursuant to 10 CFR 70.61(d), approve this proposed methodology. The applicant has committed to the methodology described below to ensure that there would be an adequate margin of subcriticality for safety during any operation of the proposed facility. Since the proposed methodology has not yet been approved, this is an open item.

ANSI/ANS-8.1-1983 (R1988) provides single and multi-parameter limits to be referenced in criticality safety calculations, as well as guidance used in the performance of criticality analysis method validation. In revised CAR Section 6.3.5, the applicant described the criticality analysis methodology to be used in facility design activities and facility safety programs, as also discussed in ANSI/ANS-8.1-1983 (R1988). The applicant stated that the chosen computational methods are those using the Evaluated Nuclear Data File (ENDF) cross-section libraries (i.e., the Standardized Computer Analyses Licensing Evaluation (SCALE) 4.4 and 4.4a 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. Use of these codes and cross section libraries is accepted industry practice, and has been recognized by the NRC as generally acceptable (as in NUREG/CR-6361).

In the revised CAR, the applicant discussed the method of validation and calculated k_{eff} design limits to be used in the facility. Consistent with ANSI/ANS-8.1-1983 (R1988), the applicant's code validation methodology will use critical experiment benchmarks to determine bias and bias uncertainties. The critical experiments should be selected to represent the neutronic and physical characteristics of the systems analyzed in specific design applications. The derived biases should then be applied to calculations on systems whose neutronic and physical characteristics are bounded by those of the selected set of experimental benchmarks. When the neutronic and physical 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 stated that criticality safety calculations would be performed for each of the stages of process operations at facility (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 those 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", NUREG/CR-6689, “Guide for Validation of Nuclear Criticality Safety Calculational Methodology”, 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.

The staff reviewed the information contained in revised CAR Section 6.3.5, "Nuclear Criticality Analysis and Safety Evaluation Methods”, and is reviewing a code Validation Report (Reference 6.3.15) submitted separately, in order to determine that an acceptable subcritical margin will be provided for the design of the facility. The following sections describe the validation methodology and the application of that methodology to determine design basis k_{eff} limits for each of the five areas of applicability (AOAs).

6.1.3.5.1 NCS Validation Report

10 CFR 70.64(a)(9) requires that the design of the proposed facility provide for criticality control, including adherence to the DCP. ANSI/ANS-8.1 requires both adherence to the DCP, and 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, and has identified it as a part of the design basis of the facility. Determining the bias, uncertainty in the bias, and an administrative margin are all necessary tasks in setting 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: (1) Confidence Band with Administrative Margin; and (2) Lower Tolerance Band, also known as the Single-Sided Uniform Width Closed Interval Approach. These approaches are summarized in the validation report.

Method 1 requires the choice of an administrative margin, which the applicant set to 0.05. Method 2 requires the choice of the confidence, α that some fraction, P, of all future calculations below the USL are subcritical. The applicant used values of α=0.95 and P=0.999 throughout the validation. These values are conservative with respect to the guidance in NUREG/CR-6698, and therefore, acceptable to the staff. The choice of administrative margin is discussed in revised DSER Section 6.1.3.5.2.

An important caveat is that the aforementioned techniques can only be used on data that is normally distributed. To verify normality, the applicant’s validation methodology uses the χ^2 (chi-squared) test for normality included with the USLSTATS code. NUREG/CR-6361 advises that normality can usually only be assured when there are more than about 25 data points. In those cases when data is not normally distributed, the applicant committed to use non-parametric methods, as discussed in NUREG/CR-6698. This requires basing the USL on the lowest observed value, with additional margin taken from a table in NUREG/CR-6698.

The methodology also describes how experimental uncertainties and statistical uncertainties are combined in a total uncertainty in the calculated k_{eff} values of benchmarks. In some cases, the benchmark configuration is not precisely critical, and the calculated k_{eff} must be normalized to

account for this ($k_{norm} = k_{calc}/k_{experiment}$). While this correction is approximate, the normalization is expected to be slight as the experiment will typically only deviate slightly from critical.

The aforementioned methodology is acceptable to the staff, as it entails techniques that have been endorsed in various NUREG publications and have been used to validate criticality codes at fuel facilities and across the domestic nuclear industry.

The applicant submitted its Validation Report in three separate parts by letter dated January 8, 2003, each addressing one or more separate AOAs. Part I included AOA(1), covering plutonium nitrate solutions, and AOA(2), covering MOX fuel pellets, rods, and assemblies. Part II included AOA(3), covering PuO_2 powder systems, and AOA(4), covering MOX powder systems. Part III included AOA(5), covering solutions of plutonium compounds such as plutonium oxalate and plutonium oxyfluoride. For each AOA, the applicant used the validation methodology described in this section.

The NRC staff determined that the code validation methodology described generally in revised CAR Section 6.3.5 was appropriate; validation of specific AOAs intended to cover anticipated AP and MP operations is covered in the following sections. For the purposes of construction authorization review, only the results of the validation (i.e., determination of the benchmark experiments and AOA, and computation of the bias and uncertainty) are being reviewed. The accuracy of benchmark and statistical calculations was not reviewed. The staff expects that the applicant will demonstrate the applicability of the validation results to specific design applications in NCSEs and will ensure that the design calculations use the code in a manner consistent with that used in the validation (e.g., similar statistics and use of code options, such as albedo and biasing). This will be reviewed as part of the review of any SNM possession and use license application.

6.1.3.5.1.1 AOA(1): Plutonium Nitrate Solutions

To validate AOA(1), the applicant modeled 191 critical experiments using the KENO-VI code with the 238-group cross section library. The bias was determined for both the SCALE-4.4 code package on the SGN Sun hardware platform and the SCALE-4.4a code package on the SGN PC hardware platform. The applicant's chosen experiments included plutonium nitrate systems modeled as spheres, cylinders, and slabs with a variety of reflector conditions (no reflector, water, concrete, and cadmium).

The applicant ran SCALE/KENO-VI for the chosen experiments, calculated k_{eff}, and plotted the values of USL-1 and USL-2 (determined by USLSTATS) versus the trending parameters of Energy of Average Lethargy causing Fission (EALF), the moderator to fuel ratio (H/Pu), and the ^{240}Pu content. Over the range of applicability, the minimum USL-1 and USL-2 were calculated as a function of these parameters, and the lowest such value chosen as the USL for the entire AOA. For each trending parameter (EALF, H/Pu, and ^{240}Pu content), USL-1 was less than USL-2 for an administrative margin of 0.05. Because the data were found to be not normally distributed, non-parametric estimates of the USL were also made. Based on the calculated value of β, no additional non-parametric margin was needed. The lowest value determined using USLSTATS was more conservative than that using the non-parametric margin, so this was used as the final USL for the AOA. For certain materials for which sufficient benchmark experiments were not available (plutonium nitrate systems with borated concrete and cadmium), the applicant presented supplemental analysis to show that the effect of these absorbers on the bias would be negligible.

The staff has reviewed this section of the validation report, and is continuing to evaluate the applicant's analysis. This is considered part of the open item in revised DSER Section 6.1.3.5.

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

To validate AOA(2), the applicant modeled 36 critical experiments using the KENO-VI code with the 238-group cross section library. The bias was determined for both the SCALE-4.4 code package on the SGN Sun hardware platform and the SCALE-4.4a code package on the SGN PC hardware platform. The applicant's chosen experiments included lattices of MOX fuel rods in water, with a variety of different moderator-to-fuel ratios (v^m/v^f) and plutonium contents.

The applicant ran SCALE/KENO-VI for the chosen experiments, calculated k_{eff}, and plotted the values of USL-1 and USL-2 (determined by USLSTATS) versus the trending parameters of EALF, the moderator to fuel ratio (v^m/v^f), and the plutonium content ($PuO_2/(PuO_2 + UO_2)$). Over the range of applicability, the minimum USL-1 and USL-2 were calculated as a function of these parameters, and the lowest such value chosen as the USL for the entire AOA. For each trending parameter (EALF, v^m/v^f, and Pu content), USL-1 was less than USL-2 for an administrative margin of 0.05. The benchmark data tested normal, though USLSTATS indicated that the amount of data may not be sufficient for this normality test. However, no anomalies in the k_{eff} histogram indicating a deviation from normality were apparent.

The staff has reviewed this section of the validation report, and is continuing to evaluate the applicant's analysis. This is considered part of the open item in revised DSER Section 6.1.3.5.

6.1.3.5.1.3 AOA(3): Plutonium Oxide Powder

To validate AOA(3), the applicant modeled 90 critical experiments using the KENO-VI code with the 238-group cross section library. The bias was determined for the SCALE-4.4a code package on the SGN PC hardware platform. The applicant's chosen experiments included reflected and bare plutonium metal systems and arrays of PuO_2- and MOX-polystyrene compacts. This set of experiments was selected from a much larger suite of candidate benchmarks.

The applicant used the Sensitivity/Uncertainty (S/U) method recently developed by Oak Ridge National Laboratory (ORNL) to determine which benchmarks were applicable to a validation of PuO_2 powder systems. To do this, three "design applications" were chosen, and the entire set of benchmark experiments compared against these three applications, using ORNL's S/U codes (SEN1, SEN3, and CANDE analytical sequences). As described in NUREG/CR-6655, "Sensitivity and Uncertainty Analysis Applied to Criticality Safety Validation", an integral measure of applicability of selected benchmarks to specific design systems is the correlation coefficient, c_k. A value of $c_k \geq 0.8$ was established as a criterion for applicability, though several other integral parameters have been defined. The experiments with $c_k \geq 0.8$ were included in the validation database applicable to each design application, labeled AOA 3-1, 3-2, and 3-3. The reduced set of experiments then comprised the experiments on which the statistical methodology was performed for the entire AOA. The results of this analysis are described in Part II, and have also been reported in ORNL/TM-2001/262, "Investigations and Recommendations on the Use of Existing Experiments in Criticality Safety Analysis of Nuclear Fuel Cycle Facilities for Weapons-Grade Plutonium."

The applicant ran SCALE/KENO-VI for the chosen experiments, calculated k_{eff}, and plotted the value of k_{eff} versus the trending parameters of EALF, the moderator to fuel ratio (H/Pu), and the ^{240}Pu content. Because the data were found to be not normally distributed, a non-parametric estimate of the USL was made, assuming an administrative margin of 0.05. Based on the calculated value of β, no additional non-parametric margin was needed. The non-parametric estimate was used as the USL for AOA(3).

The staff has reviewed this section of the validation report, and is continuing to evaluate the applicant's analysis. This is considered part of the open item in revised DSER Section 6.1.3.5.

6.1.3.5.1.4 AOA(4): Mixed Oxide Powder

To validate AOA(4), the applicant modeled 66 critical experiments using the KENO-VI code with the 238-group cross section library. The bias was determined for both the SCALE-4.4 code package on the SGN Sun hardware platform, and the SCALE-4.4a code package on the SGN PC hardware platform. The applicant's chosen experiments included arrays of PuO_2- and MOX-polystyrene compacts and lattices of MOX fuel rods in water. This set of experiments was selected from a much larger suite of candidate benchmarks.

The applicant used ORNL's S/U method to determine which benchmarks were applicable to a validation of MOX powder systems. To do this, several "design applications" were chosen, and the entire set of benchmark experiments compared against the applications, using ORNL's S/U codes (SEN1, SEN3, and CANDE analytical sequences). The experiments with $c_k \geq 0.8$ were included in the validation database applicable to each design application, labeled AOA 4-1, 4-2, 4-3, 4-4-Critical, 4-4-P163, 4-4-P40, and 4-4-P8. For design applications 4-4-Critical and 4-4-P163, the criterion was relaxed to require $c_k \geq 0.7$. The applicant defended this on the basis that these represent plutonium masses far in excess of anticipated conditions at the facility, and that the S/U analysis did not identify any benchmarks that were not also applicable to other design applications. The reduced set of experiments then comprised the experiments on which the statistical methodology was performed for the entire AOA.

The applicant ran SCALE/KENO-VI for the chosen experiments, calculated k_{eff}, and plotted the value of k_{eff} versus the trending parameters of EALF, the moderator to fuel ratios (H/(U+Pu) and H/Pu), and the ^{240}Pu and total Pu content. Because the data were found to be not normally distributed, a non-parametric estimate of the USL was made, assuming an administrative margin of 0.05. Based on the calculated value of β, no additional non-parametric margin was needed. The non-parametric estimate was used as the USL for AOA(4).

The staff has reviewed this section of the validation report, and is continuing to evaluate the applicant's analysis. This is considered part of the open item in revised DSER Section 6.1.3.5.

6.1.3.5.1.5 AOA(5): Solutions of Plutonium Compounds

To validate AOA(5), the applicant modeled 119 critical experiments using the KENO-VI code with the 238-group cross section library. The bias was determined for the SCALE-4.4a code package on the SGN PC hardware platform. The applicant's chosen experiments included arrays of bare and reflected PuO_2-polystyrene compacts and plutonium nitrate solutions.

The applicant separated these benchmark experiments into two groups for the purpose of determining the bias independently. Group 1 consisted of PuO_2-polystyrene compacts, while

Group 2 consisted of highly thermal plutonium nitrate solution systems. Thus, Group 1 (with 32 experiments) covered the high energy, low moderation portion of AOA(5), while Group 2 (with 87 experiments) covered the low energy, highly moderated portion of AOA(5). To demonstrate the applicability of the selected benchmarks in these two groups to postulated design applications, several sensitivity studies were conducted to determine the effects of different chemical forms, geometrical shapes, and reflector conditions, on the average neutron energy (EALF). The chemical compositions considered included PuO_2, hydrides of Pu(III) nitrate ($Pu(NO_3)_3 \cdot 5H_2O$) and plutonium oxalate ($Pu(C_2O_4)_2 \cdot 6H_2O$), and plutonium oxyfluoride (PuO_2F_2), also known as "standard salt". The geometrical shapes considered included infinite critical slabs and cylinders. The reflectors considered included water, plexiglass, colemanite (borated) concrete, regular concrete, and cadmium/steel. Each of these sensitivity studies plotted EALF versus H/Pu for various differing configurations. The results demonstrated that there was an increasing effect on EALF for low H/Pu as a function of chemical form, geometry, and reflector materials. In general, the most significant differences occurred at an H/Pu $\lesssim$ 50. The most reactive chemical forms were PuO_2 and PuO_2F_2 over the entire studied moderation range (although the calculations did not extend over the lowest H/Pu range). Based on this, the applicant concluded that these compounds bounded all the other chemical forms within AOA(5) and that the PuO_2-polystyrene cases were applicable to validate these materials, which bounded the various miscellaneous plutonium compounds in the low H/Pu range. The geometry with the highest leakage (infinite slab) displayed a greater sensitivity to chemical form and reflector materials. Reflectors were found to be categorized into two groups–those with strongly absorbing materials (boron and cadmium), and hydrogenous reflectors (water and plexiglass). The EALF was found to depend strongly on which class of reflectors was present, but only slightly on the particular reflector material.

The applicant ran SCALE/KENO-VI for the chosen experiments, calculated k_{eff}, and plotted the values of USL-1 and USL-2 (determined by USLSTATS) versus the trending parameters of EALF and the moderator to fuel ratio (H/Pu) for each group. Over the range of applicability, the minimum USL-1 and USL-2 were calculated as a function of these parameters, and the lowest such value chosen as the USL for the entire AOA. The lowest USL (for Group 1) was used as the USL for the entire AOA. For each trending parameter (EALF and H/Pu), USL-1 was less than USL-2 for an administrative margin of 0.05. The benchmark data tested normal, though USLSTATS indicated that the amount of data may not be sufficient for this normality test.

The staff has reviewed this section of the validation report, and is continuing to evaluate the applicant's analysis. This is considered part of the open item in revised DSER Section 6.1.3.5. The staff is continuing to evaluate the adequacy of the USL for each AOA covered in the validation report, and therefore, as stated above, this is considered an open item.

6.1.3.5.2 Determination of Normal and Abnormal Subcritical Margins

The applicant selected an administrative margin of 0.05 (for use with Method 1) for both normal and credible abnormal conditions, for each AOA covered in the Validation Report. Each part of the Validation Report contained a justification for this administrative margin. This justification depended on past nuclear industry practice and comparison with NRC guidance, as well as comparison of USL-1 (including administrative margin) with USL-2. The staff reviewed this and determined that past nuclear industry practice does not support a minimum subcritical margin of 0.05 because the currently licensed facilities most similar to the facility require greater margin in the normal condition. Both high enriched uranium (HEU) and 96wt% ^{239}Pu material exhibit greater sensitivity to small changes in physical and neutronic parameters than low enriched

uranium (LEU). NRC guidance (NUREG-1718) does state that a margin of 0.05 is "generally considered to be acceptable without additional justification when both the bias and its uncertainty are determined to be negligible". However, the applicant has not sufficiently demonstrated that these conditions are met. Finally, the comparison of USL-1 with USL-2 does not demonstrate the adequacy of the minimum subcritical margin. The condition that USL-1 < USL-2 is necessary, but not sufficient, to show that sufficient margin has been provided. Method 1 and Method 2 are two different statistical treatments of the data, and a comparison between them can only demonstrate whether the administrative margin is sufficient to bound statistical uncertainties not included in Method 1. There may also be non-statistical errors in the calculation of k_{eff} that are not handled in the statistical treatment.

Although the staff did not agree with the applicant's justifications, the staff believes that a minimum subcritical margin of 0.05 can be used for abnormal conditions at the facility, based on additional factors of safety present in the abnormal condition. These additional factors of safety to provide reasonable assurance of subcriticality in the abnormal case, with a minimum subcritical margin of 0.05. In accordance with the DCP, the abnormal condition should be at least "unlikely" to occur. This fact may be credited in accepting a smaller margin than would be acceptable in the normal case (NUREG-1718, Section 6.4.3.3.4). In addition, the applicant must evaluate the worst-case combination of parameter values in the abnormal condition, which typically results in additional conservatism. Also, a minimum subcritical margin of 0.05 has been accepted for abnormal conditions at similar facilities. Finally, there should be less uncertainty in the calculations since the abnormal condition is likely to result in a system with a well-moderated, highly thermal neutron spectrum, where the cross sections have been well-measured and the neutronic behavior well-understood.

Since these criteria are not met under normal conditions, the staff cannot conclude that a margin of 0.05 is sufficient for normal operations for all AOAs. However, for AOA(2), a margin of 0.05 is acceptable to the staff for both normal and credible abnormal conditions as explained below. 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 are well-known and have 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. The benchmarks are all well-moderated thermal systems and the neutron cross sections in the thermal energy range have been measured with a higher degree of accuracy than those in the intermediate energy range. Therefore, based on the inherent low risk posed by MOX fuel pellets, rods, and assemblies; the historical precedent for facilities with similar 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 an administrative margin of 0.05 acceptable, for both normal and credible abnormal conditions within AOA(2), for construction authorization.

These conditions are not present for AOAs (1), (3), (4), and (5), and therefore, the staff considers additional normal condition margin necessary to ensure subcriticality. These systems consist of plutonium (96wt% ^{239}Pu) or MOX (22wt% or 6.3wt% Pu), and display a much greater sensitivity of k_{eff} to changes in the controlled parameters. As stated above, they has similar neutronics to the high-enriched facilities which have historically been licensed with additional normal condition k_{eff} margins.

The staff considers the maximum k_{eff} for both normal and credible abnormal conditions to be part of the design basis of the facility. The maximum k_{eff}, plus all biases and uncertainties, shall not exceed 0.95, is a design basis value. Since the system sensitivity depends on the choice of controlled parameters, geometry, materials, and other aspects of the design, it cannot be determined *a priori* and is therefore, not part of the design basis. The staff, therefore, will review the determination of the normal condition margin as part of reviewing the applicant's request for an SNM possession and use license. The staff also considers the methodology for determining the amount of normal condition margin to be part of the design basis. The staff is continuing to review the applicant's proposed methodology (in revised DSER Reference 6.3.11) for setting the normal condition margin. Because the review of the validation of specific AOAs and the methodology for determining the subcritical margin is in progress, the staff considers this an open issue.

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 an SNM possession and use license. Since the specific controls relied on as NCS IROFS are not design information, the identification of such controls is not needed for review of the revised CAR, 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 facility, 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. Revised 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." Revised CAR Table 5.6-1, "MFFF Principal SSCs," identifies the PSSCs for criticality hazards as "criticality control", with specific controls to be determined during the design of the facility.

Because the applicant can choose any of the criticality control methods described in this revised CAR Chapter to prevent criticality, the staff did not evaluate any specific controls during review of the revised CAR. However, based on the commitments in this chapter, the staff has reasonable assurance that the design bases of the PSSCs (with the exception of the open item in this revised DSER) will provide adequate protection against the consequences of a criticality accident (as required by 10 CFR 70.23(b)), even though the specific IROFS have not yet been identified.

6.1.4.2 Commitment to DCP

The commitment to the DCP, including the corresponding definition of "unlikely", is 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. Additionally, 10 CFR 70.64(a)(9) requires, in part, that the design of the proposed facility adhere to the DCP. As discussed below, the staff considers the commitment to the DCP

acceptable and, in this regard, finds that the applicant has met the double contingency part of the 10 CFR 70.64(a)(9) requirement for construction authorization. However, as indicated in sections 6.1.3.4.2 and 6.1.3.5.1 above, the design basis of the proposed facility has not yet been found to adequately provide for criticality control, and 10 CFR 70.64(a)(9) requires that the design provide for criticality control. Accordingly, the staff finds that only the DCP portion of the criticality BDC has been met for construction authorization.

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. As discussed in revised DSER Section 6.1.4.3, “unlikely” for meeting the DCP is defined as “not expected...during the facility lifetime”. This is qualitatively consistent with a failure probability on the order of once in 100 years.

The applicant has stated that criticality will be made “highly unlikely”. An accidental criticality has the potential to exceed the 10 CFR 70.61 threshold for doses to facility workers. Dose mitigation in the form of shielding will not be credited in reducing the consequence level, as discussed in revised DSER Section 6.1.4.3. In revised CAR Section 5.4.3, the use of deterministic methods for demonstrating that criticality hazards are “highly unlikely” is described. This consisted of application of double contingency, 10 CFR 50 Appendix B, industry codes and standards, and management measures for failure detection. The applicant, however, did not explain in the revised CAR how it will determine when an acceptable level of likelihood has been reached, or how these generic criteria will be applied to specific controls. Therefore, in revised DSER Reference 6.3.11 (as amended by Reference 6.3.13), the applicant provided additional details on its method for ensuring that criticality hazards would be made highly unlikely. The description of the accident sequences leading to criticality and the controls relied on the prevent criticality will be described in NCSEs. The description of accident sequences would include the initiating event, methods of prevention, including at least two independent controls and their safety function, redundancy and diversity, the safety margin, failure modes, and failure detection and surveillance requirements. The description of controls would include the safety function, quality classification (Ql-1a or Ql-1b), operating range and limits, testing and maintenance requirements, environmental design factors, natural phenomena response, required instrumentation, and applicable codes and standards.

In accordance with the DCP, a least two independent controls whose failure is unlikely must be specified. For passively controlled units whose failure is not credible, it is considered sufficient to place them under the plant’s configuration management program and assign the highest quality level (QL-1a), as done for “sole IROFS”. In order to show incredibility, it must be demonstrated that the system will remain subcritical under all credible process conditions. As discussed in revised DSER Section 6.1.3.4, accident sequences in which there are no credible means to achieve criticality do not require dual independent controls to comply with the DCP. Therefore, this approach is acceptable to the staff.

For systems with credible failure paths leading to criticality, active or passive engineered controls must be classified at least QL-1b, and administrative controls must be simple and unambiguous. The likelihood determination will be based on consideration of all the applicable availability and reliability qualities, consistent with NUREG-1718 (described in NUREG-1718, Section 5.4.3.2), which include those specified above. The staff considers this sufficient to ensure that all factors that affect accident sequence and control likelihood will be taken into consideration. In addition, for each independent and unlikely-to-fail control, there must be either a means to detect the failure of the control (on a period to be justified in the NCSEs), or safety margin (sufficient to

ensure that repeated failure of the controls do not lead to criticality). Based on this, the staff has reasonable assurance that adherence to the DCP (in which each control failure is sufficiently reliable and available to be made "unlikely"), combined with failure detection or additional safety margin as described above, provides reasonable assurance that accident sequences leading to criticality will be highly unlikely.

For the first case, the requirement that a means to detect the control failure be provided can result in a significant reduction in the likelihood that both independent controls will be in a failed state at the same time, provided that this time period is sufficiently short compared to the expected time between failures of the control. This period will be specified and justified in the NCSEs. The staff has reasonable assurance that the combination of two independent and unlikely-to-fail controls (each of which is not expected during the facility lifetime) with adequately justified failure detection will ensure that criticality is highly unlikely.

For the second case, the safety margin must be sufficiently robust that multiple failures of the control will not lead to a loss of subcriticality. Specifically, there must be sufficient margin to accommodate at least three independent failures of the controls. In evaluating abnormal conditions, the most reactive credible change in the controlled parameter must be evaluated. This typically leads to abnormal conditions that have considerable margin, which would ensure considerable safety margin for the sequence. The requirement that the system must accommodate multiple independent failures provides a significant reduction in the likelihood of criticality. The staff considers the occurrence of multiple independent failures, each of which is not expected during the facility lifetime, to be highly unlikely.

In lieu of monthly failure detection or additional safety margin, the applicant stated that it may use other means to ensure that criticality is highly unlikely. In this case, the applicant will show that these alternate means provide comparable assurance to the two methods described above. A list of any systems utilizing methods other than failure detection or additional margin will be provided in an ISA Summary. Regardless of the means of demonstrating that criticality is highly unlikely, the rationale will be provided in NCSEs supporting a request for an SNM possession and use license and ISA Summary.

NUREG-1718, Section 5.4.3.2, contains the acceptance criterion: "A purely qualitative method of defining "unlikely" and "highly unlikely" is acceptable if it incorporates all of the applicable availability and reliability qualities to an appropriate degree...one acceptable definition of "highly unlikely" is a system of IROFS that possesses double contingency protection with each of the applicable qualities to an appropriate degree." In addition, the Commission has adopted a strategic goal of no inadvertent criticality. Pursuant to 10 CFR 70.64(a)(9), the staff, therefore, finds that the design basis of the facility adheres to the double contingency principle for construction authorization. However, as stated above, the design basis of the proposed facility has not yet been found to adequately provide for criticality control. The staff therefore finds that the criticality BDC set forth in 10 CFR 70.64(a)(9) has not been completely met.

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 revised CAR Sections 6.4 and 6.5. In revised 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 facility. Revised CAR Section 6.4 commits, in general, to "comply with the guidance (shall statements) and implement the recommendations (should statements)"

of the applicable standard, but identified several clarifications to specific commitments within certain of the standards. Thus, in the context of this section, "guidance" is equivalent to the "requirements" of the ANSI 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. Revised CAR Section 6.5 contains a discussion of commitments to ANSI/ANS-8.23-1997, which the applicant does not consider part of the design basis of the facility. 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 guidance (shall statements) of ANSI/ANS-8.1-1983 (R1988) and implement the recommendations (should statements), with clarification of three provisions. The clarifications are: (1) for 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 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 revised 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 facility lifetime" is, therefore, acceptable provided the lifetime of the facility is assumed to be greater than approximately 100 years.[4] In revised DSER Reference 6.3.6 (and revised DSER Reference 6.3.14), 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) For 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) For Section 4.3.2 of the ANSI standard, the applicant committed that where an 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. This is consistent with the endorsed standard, and is therefore acceptable to the staff.

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 1988 version, the staff considers the use of the 1988 version acceptable.

- **ANSI/ANS-8.3-1997, "Criticality Accident Alarm System":** The applicant commits to comply with the guidance (shall statements) of ANSI/ANS-8.3-1997 and implement the recommendations (should statements), as modified by RG-3.71, with clarification of one

[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.

provision. The clarification is: For 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. A request for exemption from the requirements of 10 CFR 70.24(a) will be handled as described in revised CAR Section 6.3.2, and may include risk arguments. This is consistent with the rule and regulatory practices and therefore is acceptable to the staff.

- **ANSI/ANS-8.7-1975, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials":** The applicant does not consider ANSI/ANS-8.7-1975 to be part of the design basis of the facility. The general commitment to ANSI/ANS-8.1-1983 (R1988) and technical practices as described in revised CAR Sections 6.3.3.2.8 and 6.3.4.3.2.8 should be sufficient to ensure that criticality safety is appropriately provided for fissile material storage areas.

- **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 facility. 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 does not consider ANSI/ANS-8.10-1983 to be part of the design basis of the facility, because the approach used for the facility is to prevent criticality in accordance with the DCP, rather than rely on shielding and confinement for dose mitigation. In addition, revised DSER Reference 6.3.11 states that events with the potential to exceed 10 CFR 70.61 requirements will be made highly unlikely, "including criticality events, without regard to actual dose consequences." Because shielding will not be credited in this way (it must still be considered for detector coverage), the staff considers it appropriate to exclude ANSI/ANS-8.10-1983 as a design basis for the facility.

- **ANSI/ANS-8.12-1987, "Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors":** The applicant does not consider ANSI/ANS-8.12-1987 to be part of the design basis of the facility. The staff notes that this standard does not contain any administrative requirements to which the applicant should commit. This standard only contains subcritical limits for certain plutonium-uranium mixtures. In the absence of a commitment to this standard, the commitment to ANSI/ANS-8.1-1983 (R1988) ensures the use of validated methods in computer calculations to demonstrate subcriticality. Therefore, the staff considers it appropriate to exclude ANSI/ANS-8.12-1987 as a design basis for the facility.

- **ANSI/ANS-8.15-1981, "Nuclear Criticality Control of Special Actinide Elements":** The applicant does not consider ANSI/ANS-8.15-1981 to be part of the design basis of the facility. Criticality control of special actinide nuclides will be explicitly 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-8.17-1984, "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors":** The applicant commits to comply with the guidance (shall statements) of ANSI/ANS-8.17-1984 and implement the recommendations (should statements), with clarification of two provisions. The clarifications are: (1) For Section 4.11 of the standard, 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 facility by 10 CFR 70.64(a)(9) and thus this commitment is acceptable to the staff. (2) For Section 5.1 of the standard, the applicant committed that where an 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. This is consistent with the endorsed standard, and is therefore, acceptable to the staff.

- **ANSI/ANS-8.19-1996, "Administrative Practices for Nuclear Criticality Safety":** The applicant commits to comply with the guidance (shall statements) of ANSI/ANS-8.19-1996 and implement the recommendations (should statements), with the exception that no commitments are made to Section 10 of the standard 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 revised CAR. These procedures will be evaluated by the staff if Duke, Cogema, Stone & Webster (DCS) submits a license a license application.

- **ANSI/ANS-8.20-1991, "Nuclear Criticality Safety Training":** The applicant commits to comply with the guidance (shall statements) of ANSI/ANS-8.20-1991 and implement the recommendation (should statements) without exception or clarification. This is, therefore, 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 guidance (shall statements) of ANSI/ANS-8.21-1995 (this standard contains no recommendations) without exception or clarification. This is, therefore, 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 guidance (shall statements) of ANSI/ANS-8.22-1997 and implement the recommendations (should statements), with clarification of one provision. The clarification is: For Section 4.1.7 of the standard, 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 in NCSEs supporting a license application.

- **ANSI/ANS-8.23-1997, "Nuclear Criticality Accident Emergency Planning and Response":** As stated in revised CAR Section 6.5, the applicant commits to comply with the guidance (shall statements) and implement the recommendations (should statements) of ANSI/ANS-8.23-1997 without exception or clarification. While the applicant does not consider

this part of the design basis of the PSSCs, this commitment provides reasonable assurance that consideration will be given to emergency planning during design. This is, therefore, acceptable to the staff.

- **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."

Revised CAR Section 6.3.4.3.2.9 states that wherever neutron absorber control is used, it will be part of the geometry and fixed by design. In addition, revised DSER Reference 6.3.2, RAI 77, (and revised DSER Reference 6.3.14) the applicant stated that it does not envision using raschig rings for criticality control in facility 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 facility and, therefore, are not part of the design basis of the facility.

Therefore, the following standards have been identified, in whole or in part, as part of the design basis of the facility: ANSI/ANS-8.1-1983 (R1988), ANSI/ANS-8.3-1997, ANSI/ANS-8.17-1984 , ANSI/ANS-8.19-1996, ANSI/ANS-8.20-1991, ANSI/ANS-8.21-1995, and ANSI/ANS-8.22-1997; ANSI/ANS-8.23-1997 has also been committed to. The other standards were either not applicable to the design of the facility or adequately covered by other commitments. This therefore represents an acceptable set of design bases of the facility. However, nothing precludes the use of any other standards endorsed in RG-3.71.

6.2 EVALUATION FINDINGS

In Section 6.4 of the revised CAR, DCS provided design basis information for nuclear criticality safety PSSCs that it identified for the facility. Based on the staff's review of the revised CAR and supporting information provided by the applicant relevant to nuclear criticality safety, the staff finds that, due to the open item discussed above and listed below, DCS has not met the 10 CFR 70.61(d) performance requirement or the BDC set forth in 10 CFR 70.64(a)(9). Further, until the open item is 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 item:

- Determination of design basis USLs for each process type, and determination of normal condition subcritical margin. Clarification of DCS' commitment to the preferred use of dual parameter control. (Revised DSER Section 6.1.3.4.2 and 6.1.3.5.1) (NCS-4)

The following open items in the April 30, 2002, draft Safety Evaluation Report, have been closed: NCS-1, NCS-2, NCS-3, NCS-5, NCS-6, NCS-7, NCS-8. See Appendix B.

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.

6.3.9 Ihde, R.H., Duke Cogema Stone & Webster, letter to U.S. Nuclear Regulatory Commission, RE Mixed Oxide Fuel Fabrication Facility Construction Authorization Request Revision, October 31, 2002.

6.3.10 Hastings, P.S., Duke Cogema Stone & Webster, letter to U.S. Nuclear Regulatory Commission, RE Mixed Oxide Fuel Fabrication Facility Construction Authorization Request Change Pages, December 20, 2002.

6.3.11 Hastings, P.S., Duke Cogema Stone & Webster, letter to U.S. Nuclear Regulatory Commission, February 11, 2003.

6.3.12 Hastings, P.S., Duke Cogema Stone & Webster, letter to U.S. Nuclear Regulatory Commission, RE Mixed Oxide (MOX) Fuel Fabrication Facility Construction Authorization Request Change Pages, February 18, 2003.

6.3.13 Hastings, P.S., Duke Cogema Stone & Webster, letter to U.S. Nuclear Regulatory Commission, RE Mixed Oxide (MOX) Fuel Fabrication Facility Construction Authorization Request Change Pages, April 10, 2003.

6.3.14 Hastings, P.S., Duke Cogema Stone & Webster, letter to U.S. Nuclear Regulatory Commission, RE Requests for Additional Information, Clarifications, and Open Item Mapping into the Construction Authorization Request Revision, November 22, 2002

6.3.15 Hastings, P.S., Duke Cogema Stone & Webster, letter to U.S. Nuclear Regulatory Commission, RE Duke Cogema Stone & Webster Mixed Oxide Fuel Fabrication Facility Criticality Validation Report - Criticality Validation Report - Revision 1 of Part I and Original Issue of Part II, January 8, 2003

This page intentionally left blank