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Final Safety Evaluation Report for the License Application To Possess and Use Radioactive Material at the Mixed Oxide Fuel Fabrication Facility in Aiken, SC

Docket No. 70-3098
Shaw AREVA MOX Services

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ABSTRACT

This report documents the U.S. Nuclear Regulatory Commission (NRC) staff review and safety and safeguards evaluation of the Shaw AREVA MOX Services (MOX Services) application for a license to receive, acquire, possess, use, store, and transfer by-product material, source material, and special nuclear material (SNM) at the Mixed Oxide Fuel Fabrication Facility located in Aiken, South Carolina.

The objective of the review is to evaluate the potential adverse impacts of operation of the facility on worker and public health and safety under both normal operating and accident conditions. The review also considers physical protection of SNM and classified matter, material control and accounting of SNM, and the management organization and administrative programs to ensure the safe design and operation of the facility.

The NRC staff concludes, in this Safety Evaluation Report, that the applicant's descriptions, specifications, commitments, and analyses provide an adequate basis for safety and safeguards of facility operations and that operation of the facility does not pose an undue risk to worker and public health and safety.

TABLE OF CONTENTS

EXECUTIVE SUMMARY

The U.S. and the former Soviet Union (Russian Federation) began dismantling thousands of nuclear weapons when the Cold War ended in the late 1980s. The dismantlement resulted in large quantities of surplus weapons-grade highly enriched uranium and plutonium. This surplus material necessitates special safety and management measures because of many issues related to protection of the environment, protection of public health and safety, and control of fissile material. One challenge is to dispose of this surplus material to significantly reduce both its accessibility and attractiveness for retrieval and future use in weapons.

Because of concerns over the vast stockpiles of nuclear weapons each country possessed, the U.S. and Russian Federation signed an agreement in September 2000, committing each country to dispose of 34 metric tons (approximately 75,000 pounds) of surplus plutonium. U. S. Department of Energy (DOE) evaluated different strategies to dispose of this material, and ultimately developed the Surplus Plutonium Disposition Program. Under this program, DOE plans to convert the surplus weapons-grade plutonium into Mixed Oxide (MOX) fuel to be irradiated in commercial nuclear power reactors.

In May 1998, DOE issued a request for proposals to design, construct, and operate a Mixed Oxide Fuel Fabrication Facility (MFFF), and eventually supply commercial fuel to an affiliated nuclear utility to be irradiated in its reactor. In March 1999, DOE selected a consortium consisting of Duke, Cogema, and Stone & Webster (DCS) (subsequently renamed Shaw AREVA MOX Services (MOX Services)). This partnership is to: (a) design the commercial MOX fuel; (b) design, construct, operate, and deactivate an MFFF; (c) design and execute the reactor modifications necessary for use of MOX fuel; and (d) provide the architect/engineering and construction management services associated with these activities.

The MFFF would be U.S. Government-owned and would be used to dispose of surplus plutonium and some waste from DOE's nuclear processes (alternate feedstock). Under the Strom Thurmond National Defense Authorization Act for Fiscal Year 1999, the U.S. Nuclear Regulatory Commission (NRC) was granted regulatory and licensing authority over the MFFF. The licensing review of the MFFF used existing regulations. Also, NRC developed a facility-specific Standard Review Plan (NUREG-1718), dated August 2000 (NRC, 2000), to help review the License Application (LA) (MOX, 2010a). It should be noted that NRC is also the regulatory authority for the commercial nuclear power reactor(s) that would irradiate the MOX fuel.

NRC regulatory review of the MFFF is being performed in two stages. The first stage, which has already been completed, consisted of the review and evaluation of the Construction Authorization Request (CAR) (DCS, 2004). This stage required the applicant to submit a description and safety assessment that detailed the design bases of the principal structures, systems, and components (PSSCs) of the plant, including provisions for protection against natural phenomena and the consequences of potential accidents [refer to Title 10 of the *Code of Federal Regulations* (10 CFR) 70.22(f) and 70.23(b)]. The second stage consists of a review of an LA for authority to possess and use the licensed material at the MFFF.

In February 2001, DCS submitted a CAR for an MFFF at DOE's Savannah River Site (SRS), near Aiken, South Carolina (SC). NRC conducted environmental and safety reviews of the MFFF CAR and supporting documentation. The results of the staff's environmental review are discussed in NUREG-1767, "Final Environmental Impact Statement on the Construction and Operation of a Proposed Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina," issued in January 2005 (NRC, 2005a). In March 2005, NRC issued a Construction Authorization (CA) (NRC, 2005b) to DCS for the MFFF. NRC staff's technical basis for issuing the CA is set forth in NUREG-1821, "Final Safety Evaluation Report on the Construction Authorization Request for the Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina," issued in March 2005 (NRC, 2005c).

MOX Services has submitted an LA (MOX, 2010a) and Integrated Safety Analysis (ISA) Summary (MOX, 2010b) for the MFFF at the SRS in Aiken, SC.

This safety evaluation report (SER) documents the results of the review of the application for a license to receive, acquire, possess, use store, and transfer by-product material, source material, and special nuclear material (SNM) at the MFFF.

The regulatory requirements of 10 CFR 70.23(a)(8), states that an application for a license will be approved if the Commission determines that, where the proposed activity is the operation of a plutonium processing and fuel fabrication plant, construction of the PSSCs approved pursuant to 10 CFR 70.23 (b) has been completed in accordance with the application. Thus in accordance with 10 CFR 70.23(a)(8), the staff has determined that the license to possess and use SNM will not be issued, before a determination that construction of the PSSCs is approved pursuant to 70.23(b) is in accordance with the application".

A summary of the NRC's review and findings in each of the review areas is provided below.

<u>General Information</u>

The staff concluded that (1) the level of detail in the facility and process overview provided an adequate understanding of the facility and processes and conveyed the purpose of the facility; (2) the facility and the process overview appropriately cross-referenced material presented in later sections of the LA; and (3) the facility and process overview is consistent with, yet less detailed than, material in later sections of the application.

The staff concluded that the applicant provided the geographic, demographic and land use, meteorologic, hydrologic, geologic, and seismic information relevant to the MFFF site. This information is generally current, appropriately referenced, and consistent with information in the safety assessments used to support the design bases of the item relied on for safety (IROFS) structures. The staff finds that the applicant has accurately described the site so as to properly define potential accident conditions. Based on its review of the LA and the relevant supplementary information provided by the applicant, the staff further finds that the applicant has met the baseline design criteria (BDC) in 10 CFR 70.64(a)(2) for natural phenomena hazards.

<u>Financial Qualifications</u>

As stated in §1.2.4.2 of the Construction Authorization Request (CAR) (DCS, 2004), the DOE has agreed to indemnify MOX Services in accordance with the provisions of the Price-Anderson Act set forth in §170(d) of the Atomic Energy Act of 1954, as amended, 42 U.S.C. 2210(d).

Based upon the DOE Indemnity Agreement and for the reasons discussed below, MOX Services requested an exemption from the NRC's requirements concerning agreements of indemnification and related financial protection requirements set forth in 10 CFR § 140.20 and §140.13a. 10 CFR §140.8. Based on its review of the exemption request, the staff finds that the requested exemption from the indemnity agreement and financial protection requirements of 10 CFR §140.20 and §140.13a is authorized by law and in the public interest.

Protection of Classified Matter

The specific risk of a loss or compromise of project-related classified information is the theft/diversion or radiological sabotage to the SNM at the MFFF. There is also a general risk to the classified technical information associated with the project. Classified matter will be in the form of information related to classified components. The applicant's submittals provided sufficient information, in accordance with 10 CFR Part 25 and 10 CFR Part 95, for the staff to determine that classified information will be adequately protected. Factors in the staff's decision also include the licensee's oversight by DOE and the MFFF location within the DOE SRS.

Organization and Administration

The staff evaluated the proposed organization for operation; the administration of the project; and the responsibilities, qualifications, and authorities of key management positions. The proposed organization, administration, and key management position descriptions and qualifications are consistent with guidance in NUREG-1718 and meet the regulatory requirements for organization and administration in 10 CFR 70.22 and 70.23 and are, therefore, acceptable.

The staff concludes that the applicant's organization and administration provide reasonable assurance that the applicant has an acceptable organization, appropriate administrative policies, and qualified key management positions to satisfy the regulatory requirements for a license to possess and use radioactive material.

Integrated Safety Analysis

The staff's review confirmed that the applicant's license application contains appropriate commitments, including commitments to: (1) perform and maintain an ISA; (2) compile and maintain process safety information; (3) engage personnel with appropriate training to conduct the ISA; (4) use appropriate methods to conduct the ISA; and (5) implement appropriate measures and procedures to ensure that the ISA stays accurate and up-to-date.

The staff has also verified that the applicant performed an ISA to identify and evaluate the hazards and potential accidents associated with the facility, and to establish engineered and administrative controls to ensure facility operation will be within the bounds of the 70.61 performance requirements. The staff confirmed that the applicant's ISA Summary (1) identified the hazards at the facility; (2) analyzed for accident sequences through the use of process hazards analysis; (3) evaluated and assigned consequences to the accident sequences; and (4) evaluated the likelihood of each accident consistent with the guidance in NUREG-1718. Moreover, the applicant identified all items relied on for safety, including administrative and engineered controls and has included a listing of these controls, including sole IROFS, in their ISA Summary. As a result, the NRC staff has concluded that there is reasonable assurance that the applicant's postulated accidents resulting from the facility hazards that may be anticipated to occur should be in compliance with the performance requirements of 10 CFR 70.61.

Also, the applicant uses features to reduce the challenge to IROFS where it is practical. For example, to minimize the ignition sources, the applicant will ground pipes, vessels, and gloveboxes. The NRC staff finds that these features comply with the defense-in-depth requirements of 10 CFR 70.64(b).

The staff concludes that the identification and evaluation of the hazards and accidents as part of the ISA and establishment of controls to maintain safe facility operation from their consequences meet the requirements for a license to possess and use SNM under 10 CFR Part 70, and provide reasonable assurance that the health and safety of the public, the workers, and the environment will be adequately protected.

Nuclear Criticality Safety

The staff reviewed the description of the applicant's Nuclear Criticality Safety (NCS) Program contained in Chapter 6 of the license application. Based on the review, the staff determined that there is reasonable assurance that the applicant will establish and maintain a program sufficient to ensure health and safety and compliance with all criticality safety regulatory requirements. In particular, the staff has reasonable assurance that the applicant will: (1) have in place a staff qualified to develop, implement, and maintain an NCS Program in accordance with the application's description of facility organization, administration, and management measures; (2) conduct its operations based on technical practices sufficient to ensure that licensed material will be possessed, stored, and used safely according to the requirements of 10 CFR Part 70; (3) develop, implement, and maintain a criticality accident alarm system in accordance with the requirements of 10 CFR 70.24; and (4) establish safety limits and controls sufficient to ensure subcriticality, including an appropriate margin of subcriticality for safety, and the BDC of 10 CFR 70.64. Based on this review, the staff has reasonable assurance that the applicant's NCS Program, will meet the requirements for a license to possess and use SNM under 10 CFR Part 70, and will ensure protection of public health and safety, including workers and the environment.

The staff also reviewed selected portions of the applicant's ISA Summary and supporting on-site ISA documents. Based on the review, the staff determined that there is reasonable assurance that the applicant will implement and maintain safety limits and controls sufficient to ensure health and safety and compliance with all criticality safety regulatory requirements. In particular, the staff has reasonable assurance that the applicant will establish controls on all credible accident sequences leading to criticality sufficient to ensure that: (1) credible accident sequences will be highly unlikely, (2) that all processes will be subcritical under normal and credible abnormal conditions; and (3) that all processes will adhere to the double contingency principle. Based on this review, the staff has reasonable assurance that the applicant's implementation of its ISA, will meet the applicable requirements of 10 CFR 70.66(a), and will ensure protection of public health and safety, including workers and the environment.

Fire Protection

The staff evaluated the organization and conduct of operations, facility fire protection features and systems, manual firefighting capability, and the fire hazards analysis and concludes that the applicant's proposed equipment and facilities are adequately described and will protect health and minimize danger to life or property.

The staff reviewed the applicant's fire accident analyses including the reliability and applicability of selected IROFS to the postulated initiators and fire area hazards. The staff concludes that the applicant's proposed equipment, facilities, and procedures provide a reasonable level of assurance that adequate fire protection will be provided and maintained for those IROFS to meet the safety performance requirements and the BDC of 10 CFR Part 70.61.

The staff reviewed the design bases for fire protection systems, fire related administrative controls, and buildings as described in the LA and ISA Summary. The staff concludes that fire protection related IROFS and defense in depth controls will be designed, constructed, and/or utilized consistent with good engineering practice which dictates that certain minimum requirements be applied as design and safety considerations for any new nuclear process or facility. These minimum requirements are met through the applicant's use of applicable nationally accepted fire protection codes and standards. The staff concludes that the facility meets 10 CFR 70.64(a)(3) with respect to fire protection.

Chemical Safety

The staff evaluated information provided by the applicant in the license application, ISA Summary, Nuclear Safety Evaluations, various technical reports, and responses to requests for additional information. Staff found applicant's facility and system design and facility layout pertaining to chemical safety are based upon defense-in-depth practices; and the applicant's facility design and items relied on for safety provide reasonable assurance of chemical safety at the facility for routine operations, off-normal conditions, and potential accidents. Based on the review of the license application, the staff concluded that the applicant adequately described and assessed accident sequences having potentially significant chemical consequences and effects that could result from the handling, storage, or processing of licensed materials. The license application and ISA Summary identified those chemical process hazards and potential accidents, and established safety controls to ensure safe facility operation. To ensure that the performance requirements in 10 CFR Part 70 are met, the applicant will ensure that controls are maintained available and reliable. The staff reviewed these safety controls and the applicant's plan for managing chemical process safety and its potential effects upon licensed radioactive materials and finds them acceptable.

The staff concludes that the applicant's plan for managing chemical process safety and the chemical process safety controls meet the requirements to possess and use SNM according to 10 CFR Part 70.

Radiation Safety

The applicant's radiation protection (RP) program includes the following:

- an effective, documented program to ensure that occupational radiological exposures are as low as reasonably achievable (ALARA)
- an organization with adequate qualification requirements for the radiation safety personnel
- approved written RP procedures for RP activities
- radiation safety training for all personnel who have access to restricted areas
- requirements for an air sampling program

- control of radiological contamination within the facility
- a respiratory protection program
- requirements for radiological measurement instrumentation
- a program for monitoring the external and internal radiation exposure of personnel

Conformance to this program should ensure safe operation and provide early detection of unfavorable trends to allow prompt corrective action.

The NRC staff concludes, with reasonable assurance, that the applicant's RP program is adequate and that the applicant has the necessary technical staff to administer an effective RP program that meets the requirements of 10 CFR Parts 19, 20, and 70 for a license to possess and use SNM.

Environmental Protection

The applicant has developed a program to implement adequate environmental protection measures during operation. These measures include (1) environmental and effluent monitoring and (2) effluent controls to maintain doses to the public ALARA as part of the RP program. The NRC staff concludes that the applicant's program, as described in its application and environmental report, is adequate to protect the environment and the health and safety of the public and complies with regulatory requirements imposed by the Commission in 10 CFR Part 20; 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material"; 10 CFR Part 40, "Domestic Licensing of Source Material"; 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions"; and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material."

Plant Systems

Mixed Oxide Process Description

In Section 11.1 of the LA and Section 4.1 of the ISA Summary, the applicant provided design basis information for the Mixed Oxide Process (MP) process and identified IROFS for the facility. Based on the staff's review of the LA and ISA Summary and supporting information provided by the applicant relevant to the MP process, the staff finds that the applicant has met the BDC set forth in 10 CFR 70.64(a)(3) for explosions and 10 CFR 70.64(a)(5) for chemical safety.

Aqueous Polishing Process and Chemistry

In Section 11.2 of the LA and Section 4.2 of the ISA Summary, the applicant provided design basis information for chemical process safety IROFS indentified for the MFFF. Based on the staff's review of the chapters and supporting information, provided by the applicant, that is relevant to Aqueous Polishing and chemical process safety, the staff finds that, for the reasons discussed above, MOX Services has met the BDC set forth in 10 CFR 70.64(a)(3) for explosions, and 10 CFR 70.64(a)(5) for chemical safety. Further, the staff concludes, pursuant to 10 CFR 70.23(b), that the design bases of the IROFS identified by the applicant will provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents.

Ventilation and Confinement Systems

In Section 11.3 of the revised LA and Section 4.3 of the ISA Summary, the applicant provided information for the ventilation and confinement systems that it identified as IROFS for the proposed MFFF. The staff evaluated the above information and based on the review of this information and relevant supporting information provided by the applicant, the staff concluded that the applicant's ventilation and confinement system designs and operations satisfy the staff's acceptance criteria and the systems are adequately available and reliable to perform their intended functions when needed. The applicant has satisfactorily addressed the applicable regulatory requirements, including the performance requirements, the BDC, and the defense-in-depth practices contained in 10 CFR Part 70.

Electrical Systems

The NRC staff has evaluated the information provided by the applicant for Electrical Systems in the Section 11.4 LA and the Section 4.4 of the ISA Summary. Based on the review of the chapters and supporting information, provided by the applicant, the NRC staff concludes that the BDC of 10 CFR 70.64 has been achieved, and that the concept of defense-in-depth has been applied to the design of the electrical power systems. In addition, there is reasonable assurance that the electrical systems design and operation will fulfill the functional requirements of providing reliable power to enable the MFFF IROFS to perform their required safety actions, and the electrical systems will be available and reliable to perform their intended safety functions when needed.

Instrumentation and Control Systems

The NRC staff has evaluated the information provided by the applicant for Instrumentation and Control (I&C) Systems in Section 11.5 of the LA and Section 4.5 of ISA Summary. Based on the review of the chapters and supporting information, provided by the applicant, the NRC staff has determined that the design guidance and recommendations contained in the regulatory guidance, industry codes and standards, and licensing review guidance documents to which the applicant has committed for use in completing the design of the MFFF will provide a reasonable assurance that the design criteria identified in the regulations will be adequately addressed. The NRC staff also concludes that the BDC of 10 CFR 70.64 has been achieved, and that the concept of defense-in-depth has been applied to the design of these systems. In addition, there is reasonable assurance that the I&C systems design and operation will be available and reliable to enable the MFFF IROFS to perform their required safety actions when needed.

Material Handling Systems

The staff evaluated the information provided by the applicant for material-handling equipment and controls in Section 11.6 of the LA and Section 4.6 of the ISA summary. The review of the design and operation of the material-handling systems was also closely coordinated with the review of other applicable portions of Chapters 4 and 5 of the ISA Summary, which discusses the material-handling operations and potential load-handling events.

The staff concluded that the applicant's proposed equipment, facilities, and procedures provide a reasonable level of assurance that load-handling events that cause a release of radioactive material or radiation exposures in excess of the performance requirements of 10 CFR 70.61 are highly unlikely, given the use of the designated IROFS, codes and standards, and management measures, as well as the quality assurance (QA) program. The staff further concludes that the baseline design requirements of 10 CFR 70.64 are satisfied

Fluid Transport Systems

The staff evaluated the information provided by the applicant for fluid transport equipment and controls in Chapter 11.7 and Section 4.7 of the ISA Summary. The staff concluded that the applicant's proposed equipment, controls, and procedures provide a reasonable level of assurance that events related to the fluid transport systems that could cause a release of radioactive material or radiation exposures in excess of the performance requirements of in 10 CFR 70.61 are highly unlikely with the use of the designated IROFS, codes and standards, management measures and the QA program and that the baseline design requirements of 10 CFR 70.64 are satisfied

Fluid Systems

The applicant provided design information for the fluid systems that are identified as IROFS. Based on the staff's review of Section 11.8 of the LA and Section 4.8 of the ISA Summary and supporting information provided by the applicant relevant to the fluid systems, the staff concludes, a) pursuant to 10 CFR 70.61(e) that each engineered or administrative control or control system that is needed to meet the performance requirements be designated as an IROFS and relating to the safety program that ensures each IROFS will be available and reliable to perform its intended function when needed and b), that the design bases of the IROFS evaluated in this section will provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents.

Heavy Lift Cranes

In Section 11.9 of the LA, the applicant provided design-basis information for the heavy lift cranes in the proposed facility. The applicant stated that "no MFFF heavy lift cranes have been identified as an item relied on for safety." Based on the staff's review of the Chapter 11.9 of the LA and Section 4.9 of the ISA Summary, the staff agrees with this finding and concludes, pursuant to 10 CFR 70.61(e), that regarding the heavy lift cranes of the MFFF, the applicant's proposed equipment and facilities are adequate to protect health and minimize danger to life or property

Laboratory Description

The applicant provided information related to the laboratory in Section 11.10 of the LA and Section 4.10 of the ISA summary. Based on the staff's review of these Sections and supporting information provided by the applicant relevant to the laboratory, the NRC staff finds the descriptions of the MFFF laboratory to be adequate to facilitate an understanding of the operations and possible hazards. The safety strategy for the explosion scenarios satisfies the applicant's ISA strategy. The NRC staff reasonable assurance that the identified high-consequence scenarios are highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61, the BDC in 10 CFR 70.64(a)(3) and 10 CFR 70.64(a)(5), and the defense-in-depth practices in 10 CFR 70.64(b).

Civil Structural Systems

Section 1.1.2 of the LA and Sections 2 and 3 of the ISA summary provided design-basis and structural design information for civil structural systems for the MFFF. Based on the staff review of the LA, ISA summary and supporting information that the applicant provided relevant to civil structural systems, the staff finds that the applicant has met the BDC set forth in 10 CFR 70.64(a)(2). In addition, the staff concludes, pursuant to 10 CFR 70.23(b), that the design bases of the civil structural systems identified by the applicant will provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents.

Human Factors Engineering

The staff reviewed the application of Human Factors Engineering (HFE) to personnel activities described in Chapter 12 of the LA. The staff concludes that the applicant included commitments that applied HFE to personnel activities identified as IROFS, consistent with the results of the ISA, and that its personnel activities meet the requirements associated with human factors given in 10 CFR Part 70.

Safety and Security

Material Control and Accounting

The staff reviewed the MFFF license application to possess and use strategic special nuclear materials and special nuclear materials according to Section 13.2 of NUREG-1718 (NRC, 2000) and 10 CFR Part 74. Based on the review of the license application, the staff concludes that the applicant provided an acceptably robust FNMCP for the facility operations that will meet the applicable 10 CFR Part 74 requirements. The FNMCP describes acceptable methods for achieving the performance objectives in 10 CFR 74.51(a) and the system capabilities of 10 CFR 74.51(b). As a result, the staff has determined that the applicant meets the requirements in the area of MC&A to operate the facility under 10 CFR Part 74.

Physical Protection

The NRC staff reviewed the applicant's Physical Protection Plan (PPP) for fixed site physical protection of Strategic Special Nuclear Materials (SSNM). The methods, alternate methods, and procedures outlined in the PPP satisfy or sometimes exceed the performance objectives, systems capabilities, and reporting requirements specified in 10 CFR 73.45 and 73.46. The PPP for the facility is acceptable and provides reasonable assurance that the requirements for the physical protection of SSNM will be met.

The NRC staff's reviewed the applicant's Safeguards Contingency Response Plan (SCRP). The methods, alternate methods, and procedures outlined in the SCRP satisfy or sometimes exceed the performance objectives, systems capabilities, and reporting requirements specified in 10 CFR 73.45, 73.46, and Part 73 Appendix C. The SCRP for the facility is acceptable and provides reasonable assurance that the requirements for the physical protection of SSNM will be met.

The NRC staff's reviewed the applicant's Training and Qualification Plan (T&QP). The methods, alternate methods, and procedures outlined in the T&QP satisfy or sometimes exceed the performance objectives, systems capabilities, and reporting requirements specified in 10 CFR

73.45, 73.46, and part 73 Appendix B. The T&QP for the facility is acceptable and provides reasonable assurance that the requirements for the physical protection of SSNM will be met.

Emergency Management

The staff has reviewed MOX Services dose calculations and determined that (1) the calculated dose to the offsite public is reasonable and conservative, (2) the dose to the offsite public is less than 0.01 Sieverts (1 rem) effective dose equivalent or an intake of 2 mg of soluble uranium, and (3) no formal emergency plan is required.

The staff notes that although an NRC-approved emergency plan is not required by 10 CFR 70.22, "Contents of Applications," the MFFF has committed to maintaining an emergency plan, implementing procedures, and emergency response organization for internal use. The staff considers this commitment prudent and acceptable. Based on this evaluation, the staff concludes that the MFFF has demonstrated reasonable assurance of compliance with 10 CFR 70.22(i)(1)(i), 10 CFR 70.65(a)(6), and the performance criteria in NUREG-1718, Sections 14.4.1 through 14.4.3.1.4 (NRC, 2000).

Management Measures

The staff reviewed the applicant's management measures in Chapter 15 of the LA including a) QA, b) configuration management, c) maintenance, d) training and qualification, e) procedures, f) audits and assessments, g) incident investigations, and h) records management.

Based on the evaluation of the applicant's management measures program, the staff concludes that the applicant has demonstrated reasonable assurance of compliance with 10 CFR 70.62(a)(3), 10 CFR 70,62(d), 10 CFR 70.63(a)(1), 10 CFR 70.64(a)(8), and 10 CFR 70.74(a).

Authorizations and Exemptions

The staff finds that the criteria provided by the applicant for determining whether prior NRC approval is needed are consistent with the type of changes that would be made to the LA. The staff finds that the timeliness required for prompt updating of the onsite documentation and the timeframe for reporting changes not requiring NRC prior approval are reasonable and consistent with the process for making changes to the safety program as described in 10 CFR 70.72. The staff also finds that the commitment to performing and documenting the evaluation of NRC prior approval and maintaining records is acceptable. The staff therefore finds that the authorization for making changes to the LA is acceptable.

The staff evaluated the requests for exemptions related to radiation labeling, decommissioning, and financial assurance, and found them acceptable.

REFERENCES

(DCS, 2004) Mixed Oxide Fuel Fabrication Facility Construction Authorization Request, Charlotte, NC, 2004.

(MOX, 2010a) Shaw AREVA MOX Services, "MFFF—License Application," Aiken, SC, March 2010.

(MOX, 2010b) Shaw AREVA MOX Services, “MFFF—Integrated Safety Analysis Summary,” Aiken, SC, March 2010.

(NRC, 2005a) NUREG-1767, Final Environmental Impact Statement on the Construction and Operation of a Proposed Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina, Washington, DC, January 2005).

(NRC, 2005b) Construction Authorization for the Mixed Oxide Fuel Fabrication Facility, Washington, DC, March 2005 (CA) to DCS for the MFFF.

(NRC, 2005c) U.S. Nuclear Regulatory Commission, NUREG-1821, “Final Safety Evaluation Report on the Construction Authorization Request for the Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina,” March 2005.

(NRC, 2000) U.S. Nuclear Regulatory Commission, NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility,” August 2000.

10 CFR Part 20, “Standards for Protection Against Radiation”,

10 CFR Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material”;

10 CFR Part 40, “Domestic Licensing of Source Material”;

10 CFR Part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions”;

10 CFR 70, Domestic Licensing of Special Nuclear Material

Atomic Energy Act, as amended 1954

Strom Thurmond National Defense Authorization Act for Fiscal Year 1999

Price-Anderson Nuclear Industries Indemnity Act, Washington, DC, 1957

1.0 GENERAL INFORMATION

1.1 Facility and Process Overview

1.1.1 Conduct of Review

This chapter of the Safety Evaluation Report (SER) discusses general information contained in Chapter 1 of the revised Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF or the facility) license application (LA) to possess and use radioactive material (MOX 2009). Chapter 1 of the MFFF LA provides general information about the facility processes and the site. It consists of a general facility description, material flow, and process overview. The objective of SER Chapter 1 is to familiarize the reader with the pertinent features of the proposed facility and the site.

1.1.1.1 General Facility Description

The facility will be a "plutonium processing and fuel fabrication plant," as defined in Title 10 *Code of Federal Regulations* (10 CFR), Section 70.4, "Definitions," of the 10 CFR 70.4. The facility is designed to convert surplus weapons-grade plutonium into MOX fuel that can be used to generate electricity at commercial nuclear power stations. The assemblies are composed of fuel rods which contain fuel pellets consisting of a blend of uranium and plutonium dioxides (PuO_2) (i.e., mixed oxides). The PuO_2 to be used would be obtained from weapons-grade plutonium inventories held by the U.S. Department of Energy (DOE), which are declared surplus to national security needs.

The MFFF is located in the F-Area of the DOE Savannah River Site (SRS) near Aiken, South Carolina (SC). The site encompasses approximately 0.17 km^2 (41 a), of which approximately 6.9×10^{-2} km^2 (17 acres (a)) will be developed with roads, facilities, or buildings. No roads, railroads, or waterways traverse the MFFF.

1.1.1.1.1 Controlled Area Boundary

With respect to the controlled area boundary (CAB), 10 CFR 70.61(f) requires that the applicant establish a controlled area, as defined in 10 CFR 20.1003, "Definitions." Section 20.1003 of 10 CFR defines the controlled area as "means an area, outside of a restricted area but inside the site boundary, access to which can be limited by the licensee for any reason."

The controlled area established for the MFFF includes those areas and buildings that are under Shaw AREVA MOX Services (MOX Services or the applicant) control and that are a direct part of the MFFF. The CAB is coincident with the MFFF site boundary. The CAB is depicted in Figure 1.1.2-2 of the LA.

1.1.1.1.2 Facility Buildings and Structures

Facility buildings consist of the MOX fuel fabrication building, the emergency diesel generator building, the emergency fuel storage vault, safe haven buildings, the reagent processing building, the standby diesel generator building, the secured warehouse building, the administration building, and the technical support building. Miscellaneous site structures consist of a bulk gas storage pad; heating, ventilation, air-conditioning, and process chiller pads; diesel fuel filling stations; and other minor structures.

The main building is the MOX fuel fabrication building. This building will contain all of the PuO_2 handling, fuel processing, and fuel fabrication operations of the facility. It is a reinforced concrete building having a footprint of approximately 91.5 m (300 ft) by 137 m (450 ft) by approximately 22.3 m (73 ft) above grade. The building is composed of three major functional areas, (1) the MOX processing (MP) area, (2) the aqueous polishing (AP) area, and (3) the shipping and receiving area. In the AP area, PUO_2 feedstock received from either the pit disassembly and conversion facility (PDCF), or from alternate feedstock, would be processed to remove impurities, such as gallium and americium. The purified PUO_2 would then be blended with depleted uranium dioxide (UO_2) powder and processed into MOX fuel, and ultimately fuel assemblies, in the MP area. In the shipping and receiving area, plutonium and UO_2 would be received along with other materials necessary to produce fuel assemblies. Completed fuel assemblies would be shipped to commercial nuclear power plants from this area.

1.1.1.2 Material Flow

The facility would receive PuO_2 from the PDCF, to be located on the SRS near the facility, as well as other DOE sources (i.e., alternate feedstock). The material would be transported to the shipping and receiving area of the facility in approved shipping containers. The material would be unloaded and inspected according to the MFFF Material Control and Accounting (MC&A) and Radiation Protection Programs. The material would then be moved to the AP Area. The facility also would receive depleted UO_2 at the material receipt area of the secured warehouse building, where it would be inspected according to the MC&A and Radiation Protection Programs. The depleted UO_2 would be trucked to the shipping and receiving area of the facility. Fresh MOX fuel assemblies would be stored in the assembly storage vault in the facility before shipping offsite. For shipping to commercial power plants, the assemblies would be moved to the shipping and receiving area of the facility where they would be loaded into a MOX fresh fuel transportation package that had been approved by the U. S. Nuclear Regulatory Commission (NRC) in accordance with 10 CFR Part 71, and then loaded onto a secure transport vehicle for transport to commercial power plants for irradiation.

1.1.1.3 Process Overview

The facility would have two main process operations, (1) an AP process that serves to remove impurities, such as americium and gallium (i.e., polishing), and (2) the MP which converts the plutonium and depleted UO_2 into fuel pellets, fuel rods, and fuel assemblies. A summary of the major processes in the facility is provided below. A more detailed discussion of process chemistry and chemical safety is provided in Chapter 8 and Sections 11.1 and 11.2 of this SER.

1.1.1.3.1 Aqueous Polishing Process Overview

All feedstock, both from the PDCF and from other DOE sources, will be received as PuO_2. The PuO_2 received at the MFFF will contain small amounts of impurities that must be removed for use of the MOX fuel in reactors. Feedstock from the PDCF will contain impurities such as gallium, americium, and highly enriched uranium. The diversity of impurities and the level of impurities will be higher in alternate feedstock. Some of this alternate feedstock may have higher than normal salt contaminants (other than chlorides), some will contain chloride contaminants, and some will contain small amounts of uranium. The AP process is used to remove these impurities. The AP process consists of three major steps, (1) dissolution, (2) purification, and (3) conversion.

In the dissolution step, the PuO_2 powder received from the PDCF and other DOE sources would be placed into solution by electrolytic dissolution with silver in nitric acid. For AFS material containing chlorides, a dechlorinization process will be performed in the electrolyser prior to dissolution with silver in nitric acid.

The purification step involves purification of the plutonium solution in pulsed columns by solvent extraction. The solvent mixture would be tibutyl phosphate dissolved in hydrogenated polypropylene tertrame solvent. The plutonium and uranium are extracted into the organic phase and the impurities (americium, gallium, silver, etc.) remain in the aqueous phase as raffinates. The plutonium is then separated from the uranium in the solvent by reducing the valence state of the plutonium from +4 to +3 with the addition of hydroxylamine nitrate and acid stripping, during which the plutonium is removed from the organic stream into the aqueous stream. In the aqueous purified nitrate stream, the plutonium valence state is oxidized back to the +4 valence state by passing nitrous oxide fumes through the plutonium solution in a packed column. Downstream of the plutonium separation process, the solvent solution with the plutonium removed is stripped of uranium with a nitric acid solution. The unloaded solvent solution in sent to the solvent recovery unit, while the uranium stream is sent to the aqueous liquid waste system.

The organic waste streams are collected and sent to the solvent recovery unit where they are scrubbed in a multistage mixer-settler unit to remove the degradation products. The composition of the solvent mixture is adjusted to 30% tributylphosphate in the multistage mixer-settler before being recycled to the purification step.

Various aqueous waste streams are collected and sent to the acid recovery unit where the raffinates are concentrated and the nitric acid is recovered in a two-step concentration process that is followed by rectification. The recovered acid is then reused in the process while excess acid and concentrated raffinates are sent to the aqueous waste stream.

The conversion step converts the purified plutonium nitrate stream to PuO_2 powder by the processes of precipitation and calcination. The plutonium nitrate stream is reacted with oxalic acid to form a plutonium oxalate slurry that is collected by a filter and dried in a rotary calciner where the oxalate is converted into oxide at high temperature. The PuO_2 powder is then homogenized, sampled, and stored in cans for use in the fuel fabrication process. The filtered oxalic liquor stream is treated with manganese to facilitate the decomposition of the oxalates, concentrated, and then recycled to the beginning of the extraction cycle to maximize plutonium recovery. Off gas from the rotary calciner is routed through High Efficiency Particulate Air filters prior to discharge to the atmosphere through the plant vent stack.

1.1.1.3.2 MOX Processing Overview

The purified PuO_2 powder would be used in the MP where it would be blended with depleted UO_2 powder to make MOX fuel. The MP process consists of four major steps, (1) powder blending, (2) pellet production, (3) rod production, and (4) fuel assembly production.

The first operation is the production of the powder master blend. Polished PuO_2 is mixed with Depleted UO_2 and recycled powder/pellet material to produce an initial mixture that is approximately 20% plutonium. This mixture is subjected to micronization in a ball mill and mixed with additional Depleted UO_2 and recycled material to produce a final blend with the required plutonium content (typically from 2% to 6%). This final blend is further homogenized to meet plutonium distribution requirements. During the final homogenizing steps, a lubricant and pore-former are added to control density.

The final homogenized powder blend is pressed to form “green” pellets, which are then sintered to obtain the required ceramic qualities. The sintering step removes organic products dispersed in the pellets and removes the previously introduced pore-former. The sintered pellets are ground to a specified diameter in center-less grinding machines and sorted. Powder recovered from grinding and discarded pellets are recycled through a ball mill and reused in the powder processing.

Fuel rods are loaded to an adjusted pellet column length, pressurized with helium, welded, and then decontaminated. The decontaminated rods are removed from the gloveboxes and placed on racks for inspection and assembly. Fuel rods are inserted into the fuel assembly skeleton, and the fuel assembly construction is completed. Each MOX fuel assembly is subjected to a final inspection prior to shipment in a fresh fuel shipping cask.

1.1.2 Evaluation Findings

The staff reviewed the facility and process overview from the applicant’s LA to possess and use radioactive material at the MFFF in accordance to Section 1.1 of NUREG-1718. The staff evaluated the facility and process overview descriptions provided by the applicant in Section 1.1 of the LA focusing on new or changed material when compared to the safety evaluation for the construction authorization (NRC, 2005).

The staff concluded that (1) the level of detail in the facility and process overview provided an adequate understanding of the facility and processes and conveyed the purpose of the facility; (2) the facility and the process overview appropriately cross-referenced material presented in later sections of the LA; and (3) the facility and process overview is consistent with, yet less detailed than, material in later sections of the application. As a result, the staff finds that the application meets the regulatory requirements of 10 CFR Part 70.22 and 70.65 for providing a facility and process overview for a license to possess and use radioactive material. More detailed facility and process descriptions are provided in other sections of the LA and are discussed in other chapters of this SER.

REFERENCES

(NRC, 2000) U.S. Nuclear Regulatory Commission, NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide Fuel Fabrication Facility,” Washington, DC, August 2000.

(NRC, 2005) U.S. Nuclear Regulatory Commission, NUREG-1821, “Final Safety Evaluation Report on the Construction Authorization Request for the Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina.” Washington DC, March 2005

(MOX, 2009) Shaw AREVA MOX Services, “License Application,” Aiken, SC, October 2009.

10 CFR Part 20, “Standards for Protection Against Radiation”

10 CFR Part 70, “Domestic Licensing of Special Nuclear Material”

10 CFR Part 71, “Packaging and Transportation of Radioactive Material”

SECY-07-0047, “Staff Approach to Verifying the Closure of Inspections, Tests, Analyses, and Acceptance Criteria Through a Sample-Based Inspection Program,” dated May 16, 2007

1.2 <u>Institutional Information</u>

1.2.1 Conduct of Review

This chapter of the SER contains the staff’s review of institutional information described by the applicant in Chapter 1 of the LA. The staff used Chapter 1 in NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility,” (NRC, 2000) as guidance in performing the review.

1.2.1.1 Corporate Identity

MOX Services is the applicant for the license to possess and use by-product material, source material, and special nuclear material (SNM). MOX Services is incorporated in the State of South Carolina as an LLC owned by Shaw Project Services Group, Inc. (SPSG), Shaw Environmental & Infrastructure, Inc. (SE&I), and AREVA, Inc. These three companies are the equity owners of the LLC (SPSG 40%, SE&I 30%, and AREVA 30%). MOX Services was formed to provide MOX fuel fabrication and other services to support the mission of DOE for the disposition of U.S.-owned surplus weapons-usable plutonium.

1.2.1.2 Foreign Ownership, Control, or Influence

DOE is the owner of the MFFF, which is located at SRS in Aiken, SC. MOX Services is a South Carolina LLC, whose direct owners are all U.S. corporations. AREVA, Inc. (formerly COGEMA, Inc.), which owns a minority share of MOX Services (30%), is itself a wholly owned subsidiary of AREVA NC, a French company. SPSG and SE&I together hold a 70% majority interest in MOX Services. As a result, there is no direct foreign ownership, no foreign control, and no significant foreign interest in MOX Services. Furthermore, in awarding the contract to MOX Services to design, construct, and operate the MFFF, DOE engaged in a foreign ownership, control, or influence (FOCI) review in accordance with DOE Order 470.1, “Safeguards and Security Program”. Based upon that review, DOE rendered a favorable FOCI determination on July 9, 1999, based on a Security Control Agreement between Shaw AREVA MOX Services, LLC and DOE, mitigating FOCI. Additionally, a favorable FOCI determination has been made for SE&I (30 September 2006, through reciprocity with the Department of Defense).

The NRC accepts DOE FOCI determinations based on a memorandum of understanding between the NRC and DOE dated October 9, 1996.

1.2.1.3 Proposed License Information

The applicant requested a license to receive, acquire, possess, use store, and transfer by-product material, source material, and SNM pursuant to 10 CFR Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material”, 10 CFR Part 40, “Domestic Licensing of Source Material”, and 10 CFR Part 70 for the materials identified in Table 1.2-1.

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The term of the license that was requested is 20 years.

1.2.1.3.1 Regulatory Requirements to be Met Prior to Issuing a License to MOX Services for the MFFF

The regulatory requirements of 10 CFR 70.23(a)(8) states that an application for a license will be approved if the Commission determines that, where the proposed activity is the operation of a plutonium processing and fuel fabrication plant, construction of the principal systems, structures, and components (PSSCs) approved pursuant to 10 CFR 70.23(b) has been completed in accordance with the application. PSSCs are safety controls that are identified in the design bases as providing protection against the consequences of accidents or natural phenomena. A safety control is a system, device, or procedure intended to regulate a device, process, or human activity to maintain a safe state.

Thus in accordance with 10 CFR § 70.23(a)(8), the staff has determined that any license to possess and use SNM will not be issued, before a determination that construction of the PSSCs approved pursuant to section 70.23(b), is in accordance with the application.

The MFFF Construction Authorization Requests (CAR) lists the 53 PSSCs in Table 5.6-1 and their associated safety functions. The PSSCs are identified as administrative controls, active engineered controls or passive engineered controls. Since the NRC approval of the MFFF CAR on in March 2005 (NRC, 2005), the applicant has identified in the Integrated Safety Analysis (ISA) Summary associated with the LA, approximately 15,000 Items Relied On for Safety (IROFS) designated to perform the design basis safety functions of the PSSCs.

The staff's findings as documented in the MFFF construction authorization, stated "in accordance with 10 CFR 70.23(b), on the basis of information described in the CAR, as revised, and the additional statements and commitments heretofore made by DCS (now called Shaw Areva MOX Services), the design bases of the PSSCs for the proposed MFFF and the quality assurance program, provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents", Additionally, the ISA Summary provides the IROFS for the facility that support the performance of the MFFF's safety functions. These IROFS perform the safety functions needed to satisfy the design bases defined in the CAR.

The verification of the construction of a PSSC will vary depending on the type and nature of the system, structure, or component. In some cases, a PSSC may be an administrative control (e.g., combustion loading controls), an active or passive engineering control, use of an approved item (e.g., 3013 transport cask approved under 10 CFR Part 71), or some combination of the above. Verification of PSSCs includes evaluations of procedures and of administrative and engineering IROFS. To support the verification, PSSCs that include items such as safety related inaccessible tanks in process cells will need to be verified prior to their entrapment.

As applicable to the specific type of PSSC, NRC construction inspection and/or the technical review programs will verify that the construction of each PSSC listed in Table 5.6-1 of the MFFF CAR has been completed and the design basis safety function can be met. It is expected that the IROFS related to a specific PSSC will range from one to several thousand.

Inspection of each IROFS is not required for providing reasonable assurance that construction of the PSSCs has been completed in accordance with the application. For this reason, the NRC has historically relied on a sample-based inspection program. One approach is for the staff to rely on a randomly selected set of samples for inspection. An alternative approach is to select a sample of inspection targets IROFS to determine if there is a reasonable basis for concluding that construction of the PSSCs has been successfully completed. The staff's chosen approach for a particular PSSC will be that which best fits the nature of the PSSC and can be practically performed.

The staff will prioritize potential IROFS for inspection considering the following attributes: (1) safety significance; (2) propensity for errors; (3) construction and testing experience; and (4) opportunity to verify by other means. A more detailed discussion of these attributes can be found in SECY-07-0047 (NRC, 2007), dated May 16, 2007. The NRC will focus its inspections on activities contributing to IROFS determined to have higher inspection value[1]. Similar to the definition in SECY-07-0047, it is not the IROFS that are prioritized, but rather the value of inspecting the IROFS to maximize the agency's ability to detect significant construction flaws. This inspection sample will include both observation of IROFS-related work at the MFFF construction site, vendor facilities, and review of calculations and analyses by the Office of Nuclear Material Safety and Safeguards (NMSS) technical staff. These inspection targets will define the minimum sample set the NRC will inspect. This will provide the staff with a comprehensive sample based on inspection and technical review for IROFS-related work.

It should be noted that some of the PSSCs described in the MFFF CAR have only one safety function and have only a few IROFS associated with that safety function. In that scenario, the inspection target sample size would be equal to the number of IROFS. The verification process will incorporate one or more of the following methods: (1) PSSC field inspection results; (2) technical staff reviews and evaluations; and (3) staff review of MOX Services PSSC completion bases. The inspections will include reviews of procedures, design verification and engineering reviews, vendor and procurement inspections, receipt inspections, installation inspections, reviews of testing and surveillance and maintenance inspections.

For each PSSC, the NRC will develop a verification plan to outline the inspection and technical review activities that will be performed in order to support the staff findings regarding 10 CFR 70.23 a(8). The NRC will certify that the verification of construction completion for all PSSCs will be subject to verification through inspection and technical review. The certification process is similar to that described in Inspection Procedure 94300 and will include the issuance of a PSSC construction completion memorandum following staff verification. This certification would indicate that there is reasonable assurance that the construction of each PSSC has been completed based on a comprehensive inspection verification process and include references to the relevant inspection reports. The NRC will file a notice advising the Atomic Safety and Licensing Board (if the record is still open) or the Commission (if the record is closed) once all information relevant to the verification of construction completion is before the agency.

1.2.2 Evaluation Findings

The staff evaluated the institutional information for approval to construct an MFFF at the SRS according to Section 1.2 of NUREG-1718 (NRC, 2000). The staff evaluated institutional information identifying the applicant's corporate structure, FOCI determinations, and proposed license possession limits in the license application focusing on new or changed material when compared to the safety evaluation of the construction authorization (NRC, 2005). The staff finds that the information is complete and accurate, is consistent with the recommendations in NUREG-1718 (NRC, 2000), and is, therefore, acceptable.

Based on the review, the staff concluded that the applicant meets the regulatory requirements in 10 CFR Part 70 for ownership, location, planned activities, and nuclear material to be handled in connection with the LA to possess and use radioactive material for the MFFF.

[1] The term *higher inspection value* is defined in SECY-07-0047, "Staff Approach to Verifying the Closure of Inspections, Tests, Analyses, and Acceptance Criteria Through a Sample-Based Inspection Program," dated May 16, 2007.

REFERENCES

(NRC, 2000) NRC. NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide Fuel Fabrication Facility". Washington, DC, 2000.

(NRC, 2005) U.S. Nuclear Regulatory Commission, NUREG-1821, "Final Safety Evaluation Report on the Construction Authorization Request for the Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina." Washington DC, March 2005.
(MOX, 2008) Shaw AREVA MOX Services. "License Application," October 2009.

(MOX, 2005) Duke Cogema Stone and Webster, "Construction Authorization Request", March 2005.

(NRC, 2007) SECY-07-0047, "Staff Approach to Verifying the Closure of Inspections, Tests, Analyses, and Acceptance Criteria Through a Sample-Based Inspection Program," dated May 16, 2007.

1.3 Site Description

1.3.1 Conduct of Review

This section of the SER contains the staff's review of the site description provided by the applicant in the Shaw AREVA MOX Services LA (MOX, 2010a) and ISA Summary (MOX, 2010b). This review (1) ensures that site conditions, including site geography, demographics, meteorology, hydrology, and geology, are accurately described to properly define potential accident conditions and (2) determines whether Items Relied Upon for Safety (IROFS) and their design bases, identified by the applicant, provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents. The staff based its review of the site description on the natural phenomena accident sequences described in the ISA Summary. The staff's review was performed in accordance with the review guidance in NUREG-1718 (NRC, 2000.)

The staff reviewed how the information in the LA addresses the following regulations:

- 10 CFR 70.61(e). "Performance requirements," which requires that each engineered or administrative control or control system that is needed to meet the performance requirements be designated as an IROFS. It also requires that the safety program will ensure that each IROFS will be available and reliable to perform its intended function when needed.

- 10 CFR 70.64, "Requirements for new facilities or new processes at existing facilities," which requires that baseline design criteria and defense-in-depth practices be incorporated into the design of new facilities. With respect to natural phenomena hazards, 10 CFR 70.64(a)(2) requires that the design of new facilities must adequately protect against such hazards, considering the most severe documented historical events for the site.

Section 1.3 of the LA discusses the geographic location of the MFFF and its environment, including demographic, meteorological, hydrological, geological, seismological, and geotechnical characteristics of the site and the surrounding area. It describes population distribution near the site, land and water uses, transportation routes, and nearby industrial facilities that potentially affect the site. It also describes and evaluates site characteristics that influence the magnitude of natural phenomena (e.g., rain, snow, wind, and earthquakes) that may affect the site. This section also evaluates site characteristics with respect to safety and identifies assumptions and input needed to evaluate safety and the design bases.

The staff evaluated site characteristics and made findings of regulatory compliance by reviewing Section 1.3 of the LA, documents cited in the LA, and other relevant literature.

1.3.1.1 Site Geography

The application provided information on the site location, including State, county, municipality, and topographic information; public and SRS roads, railroads, and waterways; nearby bodies of water; and significant geographic features.

The MFFF site is located in the F-Area of the SRS in southwestern South Carolina near Aiken. The site has restricted access. There are no unrestricted public roads near the F-Area. DOE operates a rail system at the SRS, which connects to commercial rail lines outside SRS boundaries.

Nearby, the principal bodies of water are Thurmond Lake and the Savannah River. The Savannah River forms the SRS southwestern boundary. The only river navigation that takes place is infrequent construction-related barge traffic. The significant physiographic features at the SRS are the Pleistocene Coastal Terraces and the Aiken Plateau.

Only two airports that provide scheduled air passenger services are within 97 kilometers (km) (60 miles (mi)) of the SRS. The applicant identified six general aviation airports. An Internet search confirmed that the list of airports identified in the LA is complete. Wackenhut Services, Inc., operates a heliport at the SRS in the B-Area, approximately 4.8 km (3 mi) from the MFFF. Section 1.1.3.2.3 of this SER discusses aircraft hazards.

The geographic information provided in the application is current, accurate, and is consistent with information used in the LA to support the design bases of IROFS and meets the guidance in Section 1.3.3(A) of NUREG-1718 (NRC, 2000).

1.3.1.2 Demographics and Land Use

In the LA, the applicant provided information on demographics and land use. This information includes data for the area and for minority and low-income populations; a description, and distance and direction to, nearby population centers, public facilities, hospitals, and industrial facilities that could present potential hazards; residential, industrial, commercial, and agricultural land use data in the vicinity of the proposed site; and uses of nearby bodies of water. Data was derived by the applicant from a Westinghouse Savannah River Site document that was prepared in 2000. The applicant did not update its demographic data since the submittal of the Construction Authorization Request (CAR)(DCS, 2004).

Based on the 1990 Census data, 621,527 people live within 80 km (50 mi) of the proposed facility site. The population is expected to grow to slightly more than 1 million in 2030. This population includes those living in the two metropolitan areas of Augusta, GA, and Aiken, SC. Because the site is on the SRS, there are no residents within 8 km (5 mi). Between 8 km (5 mi) and 16 km (10 mi) from the site, 6,528 people reside, the majority in the towns of New Ellenton and Jackson, SC. Note that the applicant stated that the population within 50 miles of the MFFF increased by 16% based on the 2000 census, which is 2% greater than the project data from 1990. However, the applicant stated that since the ISA Summary does not use any of this population data in any calculations of event consequences and the difference in population data from the 1990 census to the 2000 census is small, that population data did not need to be updated.

Nearby industrial areas include the following:

- other DOE SRS operations
- several other Federal- and State-sponsored activities
- Chem-Nuclear Systems, Inc., commercial low-level waste disposal and waste transportation activities in Barnwell County
- Transnuclear, Inc., waste transportation activities in Aiken County
- Carolina Metals, Inc., depleted uranium processing operations in Barnwell County
- Vogtle Electric Generating Plant nuclear power generating operations across the Savannah River in Georgia
- Urquhart Station fossil fuel-fired electric generating operations 32 km (20 mi) north of the SRS
- Fort Gordon Army post operations southwest of Augusta, GA

Within the SRS, land use is controlled for the purposes of DOE operations and timber management. The U.S. Forest Service manages forested areas within the SRS.

The Savannah River water is classified as Class B, suitable for domestic supply after treatment, fish propagation, and commercial and agricultural uses. Domestic uses of water from the Savannah River occur approximately 161 km (100 mi) downstream at treatment plants near Hardeeville, SC, and Savannah, GA. Except for limited transportation of construction equipment, no commercial shipping occurs on the river.

Ground water extracted near the SRS is used for domestic, industrial, and agricultural activities. Small communities, schools, and small commercial businesses also use local ground water. Nearly 133 million liters per day (35 million gallons per day) of ground water were pumped and used in 1985, by 56 communities and industries near the SRS.

The demographic and land use information provided in the application is appropriately referenced, and is consistent with information used in the LA to support the design bases of IROFS and meets the guidance in Section 1.3.3 (B) of NUREG-1718 (NRC, 2000). This information was previously evaluated by the staff in NUREG-1821 (NRC, 2005)

1.3.1.2 Meteorology

In the LA, the applicant provided meteorological information on temperatures; windspeeds and average and prevailing wind directions; amounts and forms of precipitation; design-basis values for maximum snow and ice loads and probable maximum precipitation; and types, magnitudes, and frequency of severe weather events, such as tornadoes, hurricanes, and lightning.

Temperature data for the SRS are presented in the LA based on 35 years of measurements at the site. The annual average temperature is 18.2 degrees Celsius (C) (64.7 degrees Fahrenheit (F)). Observed temperature extremes ranged from 41.7 to −19.4 degrees C (107 to −3 degrees F). Data in the LA for Augusta, Georgia, indicate that daytime high temperatures rarely fall below 0 degrees C (32 degrees F) during the winter. Temperatures are above 32.2 degrees C (90 degrees F) on more than half the days in the summer months.
Winds near the SRS are generally light to moderate, with the highest windspeeds occurring in the spring. The lightest winds occur in the summer and fall. The prevailing wind direction varies throughout the year, coming from the northwest in the winter, from the southeast in the late spring and early autumn, and from the southwest in the summer. The peak wind gust at Bush Field (airport) in Augusta, GA, was 96.5 kilometers per hour (km/h) (60 miles per hour (mph)) based on 10 years of data.

The average annual precipitation for the SRS from 1967 to 1996 was 126 centimeters (cm) (49.6 inches (in.)). The most rainfall during a 24-hour period was 19 cm (7.5 in.) in October 1990. During summer thunderstorms, rainfall rates of up to 5.1 cm/h (2 in./h) can occur. An average of 54 thunderstorm days per year has been observed. Hail storms occur infrequently—an average of once every 2 years.

Snowfalls of 2.5 cm (1 in.) or greater occur an average of once every 3 years. The greatest single snowfall recorded from 1951 to 1995 occurred in Augusta, GA, in 1973, when 35.6 cm (14.0 in.) of snow fell. The maximum ground snow load for a 100-year recurrence period is 0.29 kilopascals (kPa) (6 pounds per square foot (psf)). Ice accumulates once every 2 years. The maximum accumulation for a 100-year recurrence period is 1.7 cm (0.67 in.) or an ice load of 0.14 kPa (3 psf).

During a 30-year period (1967–1997), 165 tornadoes occurred near the SRS. Five Fujita scale 2 and four Fujita scale 1 tornadoes have occurred on site or nearby since site operations began. Damage was primarily to trees. One of these tornadoes produced windspeeds up to 241 km/h (150 mph). Design-basis windspeeds for the DOE moderate-hazard performance category (PC-3) facilities and high-hazard performance category (PC-4) facilities are 290 km/h (180 mph) and 386 km/h (240 mph), respectively. The “IROFS safety” structures are evaluated for a tornado recurrence interval of 2×10^{-6} per year and a design-basis tornado with a 3-second tornado speed of 386 km/h (240 mph).

For other extreme winds from hurricanes, tropical weather systems, thunderstorms, and winter storms, IROFS structures are evaluated based on a recurrence period of 1×10^{-4} per year for a 3-second windspeed of 209 km/h (130 mph). These extreme windspeeds are based on SRS meteorological data and data from National Weather Service stations in Columbia, SC, and Augusta, Macon, and Athens, GA.

During the period 1700–1992, 36 hurricanes caused damage in South Carolina. However, no hurricane-force winds of greater than 120 km/h (75 mph) have been measured at the SRS.

Extreme rainfalls generally occur during spring and summer thunderstorms and tropical storms. The IROFS structures are evaluated for a design-basis rainfall representing a recurrence interval of 1×10^{-5} for various rainfall durations (e.g., 9.9 cm (3.9 in.) for a 15-minute rainfall and 58 cm (22.7 in.) for a 24-hour rainfall).

The number of lightning strikes is estimated at 10 strikes/km^2/year (yr) (26 strikes/mi^2/yr). From 1989 to 1993, SRS data show an average of 4 strikes/km^2/yr (10.4 strikes/km^2/yr).

The applicant used meteorological data in support of the natural phenomena (NPH) design bases. The NPH design bases was established in the CAR and reviewed by the staff (as documented in NUREG-1821) and is consistent with the safety assessment of the design bases presented in the CAR and the ISAS. The information reflects observations of meteorology over a period of years and was sufficient to establish the design basis. Additionally, dispersion factors which were used to calculate offsite dosed were bases on SRS data conservatively calculated based on the 95% meteorological conditions. This dispersion data was used by the applicant to demonstrate compliance with 10 CFR Part 70.61 for normal and accident conditions (MOX, 2010c).

Meteorological information provided in the application is appropriately referenced, and is consistent with information used in the LA to support the design bases of IROFS and meets the guidance in Section 1.3.3 (C) of NUREG-1718 (NRC, 2000).

1.3.1.3 Hydrology

In the LA, the applicant provided information on surface hydrology, including descriptions of nearby rivers, streams, and other water bodies; subsurface water hydrology, including water table depths, flow characteristics, potentiometric surfaces, and aquifer characteristics; and design-basis floods.

The Savannah River forms the southwestern boundary of the SRS and is the dominant body of surface water in the nearby area. The Savannah River Basin drains an area of 27,394 km^2 (10,577 mi^2) and extends 465 km (289 mi) from the Atlantic Ocean to the Blue Ridge Mountains. The principal streams that enter the Savannah River from the SRS are Upper Three Runs, Fourmile Branch, Pen Branch (PB), Steel Creek, and Lower Three Runs. These streams discharge water from rainfall, subsurface waters, and various effluent streams from SRS operations. Surface water bodies include Par Pond and L Lake, which were created as cooling water reservoirs for production reactors, marshes, and natural basins, including Carolina bays.

The record historical Savannah River flood at Augusta, GA, in 1796 had a discharge of 10,000 cubic meters per second (m^3/s) (360,000 cubic feet per second (ft^3/s)). The peak Savannah River flow recorded by the U.S. Geological Survey was 9,900 m^3/s (350,000 ft^3/s) in 1929. No major floods have occurred in the Augusta area since dams were constructed upstream of Augusta beginning in the 1950s. The estimated 50-year maximum flow is 2,100 m^3/s (74,600 ft^3/s). The probable maximum flood at the SRS is a water level of 68.4 meters (m) (224.5 feet (ft)) above mean sea level. The normal Savannah River flow elevation at the SRS boat dock is 25.9 m (85 ft). The design-basis flood for the MFFF is 63.4 m (207.9 ft) above mean sea level with an annual recurrence interval of 1×10^{-5}. Because the facility is located at an elevation of 83 m (272 ft), the probabilities of flooding the site are calculated to be less than 1×10^{-5} per year. A cascading failure of the Savannah River dams upstream of Augusta, GA, was estimated to produce a peak flow in the Savannah River of 27,751 m^3/s (980,000 ft^3/s) and a flood elevation of 43 m (141 ft) at the Vogtle station, which is directly across from the SRS on the Georgia side. Because the MFFF is at an elevation of 83 m (272 ft), this cascading failure and other events, such as ice flooding, wave surges, and seiches, will not affect the facility based on review of the information in the LA.

Three aquifer systems that overlie the bedrock formations of the Southeastern Coastal Plain characterize the ground water setting at the SRS. The Southeastern Coastal Plain consists of sediments deposited from erosional processes of the Appalachian Mountains that lie to the west of the SRS. These sediments consist of water-bearing sandy materials and limestone and clayey confining units. In the F-Area, the confining units of the three aquifer systems become disjointed and have poor separation that allows flow between aquifer systems. In the uppermost Floridan Aquifer System (FAS), the Three Runs Aquifer overlies the deeper Gordon Aquifer. These aquifers are separated by a Gordon confining unit. Recharge of these aquifers is primarily through local precipitation, and discharge is primarily through local streams. Because Upper Three Runs Creek and the Savannah River incise the FAS, there is a head reversal between the Floridan Aquifer and the Crouch Branch Aquifer in the Dublin Aquifer System, which lies just below the FAS. This means that ground water from the lower system is under a greater head and flows up into the Floridan. This phenomenon tends to limit migration of contamination into the lower aquifer systems. The Midville Aquifer System is the deepest system and lies just above the bedrock formations.

At the MFFF site, the ground water table is nearly 15 m (50 ft) below the existing ground level. Potentiometric surface maps show that ground water in the uppermost Upper Three Runs Aquifer flows principally toward Upper Three Runs Creek and toward the unnamed creek located northeast of the proposed site. The underlying Gordon Aquifer flows horizontally toward the Savannah River. The deeper Dublin and Midville Aquifer Systems flow to the southeast toward the Savannah River and the coast. The hydraulic conductivity of the Upper Three Runs Aquifer varies from less than 0.3 meters per day (m/d) (1.0 foot per day (ft/d)) to almost 10 m/d (33 ft/d), with an average of nearly 3 m/d (10 ft/d). At the MFFF site, ground water is abundant, usually soft, slightly acidic, and low in dissolved solids. Ground water used in site operations from the F-Area is treated to raise the pH and remove iron.

The F-Area seepage basin, located west of the MFFF site, was remediated in 2000 according to a hazardous waste Part B postclosure permit issued by the State of South Carolina. After remediating the site, boundary wells hydrologically downstream of the seepage basin were installed and samples were analyzed. The first set of analyses indicated that a contamination plume, exceeding the U.S. Environmental Protection Agency drinking water standards, exists. This contaminant plume extends beneath the MFFF site and is most pronounced under the western edge of the site.

The applicant indicated that there is radioactive contamination in the Upper Three Runs aquifer from upgradient contamination sources in the F-Area, as well as the F-Area seepage basin. This ground water contamination consists of concentrations of gross alpha and beta activity, uranium, tritium, and trichloroethylene exceeding the maximum contamination limits for drinking water. The applicant also indicated that ground water contamination occurs at least 9.1 m (30 ft) below the deepest level of expected construction.

During site characterization activities, the applicant measured radioactivity levels of soils using Geiger-Mueller detector scans and gross alpha and beta measurements of soil samples. The applicant indicated that the sensitivity of the gross alpha and beta measurements was 7,407,407 becquerels per kilogram (Bq/kg) (200,000,000 picocuries per kilogram (pCi/kg)) and 3,704 Bq/kg (100,000 pCi/kg), respectively. In responses (Hastings, 2003), the applicant stated that the sensitivity of soil radioactivity measurement in the preconstruction environmental monitoring report (Fledderman, 2002) was much better than that described in the calendar year 2000 geotechnical investigations.

The 2002 preconstruction environmental monitoring report measured actinide concentrations in soil and reported a mean value of 0.46 Bq/kg (12.5 pCi/kg) ^{239}Pu and a maximum of 162 Bq/kg (4,380 pCi/kg) ^{239}Pu, for example. The SRS radiological soil guideline for SRS worker protection is 9,185 Bq/kg (248,000 pCi/kg) (Jannik, 1995). Across the depth profile, the values for ^{239}Pu follow in Table 1.3-1.

Table 1.3-1 ^{239}PU Content at Various Depths

Depth, cm (in.)	Mean, Bq/kg (pCi/kg)	Maximum, Bq/kg (pCi/kg)
0–7.6 (0–3)	5.0 (137.0)	25.6 (690)
7.6–15.2 (3–6)	3.2 (87.1)	58.9 (1,590)
15.2–22.9 (6–9)	5.70 (154.0)	162.2 (4,380)
22.9–30.1 (9–12)	4.5 (121.0)	158.5 (4,280)

These values correspond to a calculated potential maximum dose of 0.003 millisieverts (mSv) (0.3 millirem (mrem)) to an exposed worker using the mean values, and a maximum exposure of 0.033 mSv (3.3 mrem) using the maximum values (Fledderman, 2002). The 0.033-mSv (3.3-mrem) annual projected dose is acceptable because the NRC's annual limit for members of the public, including construction workers in the controlled area, is 1 mSv (100 mrem).

The hydrologic information provided in the application is appropriately referenced, and is consistent with information used in the LA to support the design bases of IROFS and meets the guidance in Section 1.3.3 (D) of NUREG-1718 (NRC, 2000).

Construction of the facility will not penetrate into the upper ground water table that exists 15 m (50 ft) below grade level. Therefore, the ground water contamination in the Upper Three Runs Aquifer is not expected to result in hazardous conditions that could affect the facility.

1.3.1.5 Seismic Hazards

To assess the potential seismic hazard at the site, the applicant established two sets of ground motion spectra in the LA: one for the design of the surface facilities and one for soil stability analyses (liquefaction and dynamic settlements). Although the details of these spectra differ, analyses presented in the LA show that they are comparable. The design spectra (both vertical and horizontal) for the facility use a spectrum included in Regulatory Guide (RG) 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants" (NRC, 1973), anchored at 0.20g peak ground acceleration (PGA). The spectra are also used for the design of the nearby Vogtle Nuclear Power Plant (licensed under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"). For soil stability, a spectrum was developed based on the existing DOE uniform hazard spectra developed for the SRS. Because the seismic design and analyses rely on these established spectra, much of the site-specific seismic hazard information presented in the LA was developed to establish that these two proposed design and soil stability analysis spectra are adequate to meet the regulatory requirements of 10 CFR Part 70 and the performance guidelines in NUREG-1718 (NRC, 2000).

The following areas concerning the seismic hazards applicable to the safety analysis and design of the proposed facility were reviewed:

- seismic source characterization
- ground motion attenuation
- seismic hazard calculations
- SRS-wide rock and surface response spectra

- site response and design ground motion
- surface faulting

1.3.1.5.1 Seismic Source Characterization

1.3.1.5.1.1 Geological and Tectonic Setting

The LA details the local and regional geologic and tectonic characteristics. The LA notes that the SRS is located on sediments of the Upper Atlantic Coastal Plain in South Carolina. These sediments consist of stratified, but generally unconsolidated, sands, silts, clays, and carbonaceous muds deposited in fluvial, deltaic, near-shore, and marine shelf environments. They range in age between Late Cretaceous (about 100 million years) and the present and reach a maximum thickness of approximately 1,200 m (4,000 ft). Similar to Coastal Plain sedimentary sequences along the entire Atlantic seaboard, the South Carolina Coast sediments rest without conformity on Precambrian to Paleozoic (about 1,100–245 million years) metamorphic, metasedimentary, and igneous rocks of the Appalachian Orogen and on Triassic to Early Jurassic (about 245–180 million years) siliciclastic rocks associated with early rifting along the North American continental margin. Age and distribution of the rocks and strata provide an adequate geologic record to assess faulting and earthquake hazards.

Earthquakes that could affect safe operation of the proposed facility are associated with two seismic sources: repeat of the Charleston 1886 earthquake within the Middle Place-Summerville Seismic Zone and small shallow earthquakes of the South Carolina Piedmont. Earthquake source characteristics associated with these seismic zones are consistent with information used in two seismic hazard studies for the eastern United States: the Electric Power Research Institute report “Probabilistic Seismic Hazard Evaluations at Nuclear Power Plants in the Central and Eastern United States: Resolution of the Charleston Earthquake Issue” (EPRI, 1989) and NUREG-1488, “Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains” (NRC, 1994). As discussed in Section 1.3.1.5.2 of this report, the bedrock uniform hazard spectra for both of these sources form the basis for the sitewide DOE PC-3 and PC-4 hazard spectra (Westinghouse Savannah River Company (SRC), 1997).

1.3.1.5.1.2 Historical Seismicity

The LA includes a summary of the records of historical seismicity, including those from the cultural historical record (historical accounts date back to about 1698), as well as more recent instrumented earthquake records (the South Carolina seismic network and the SRS network, both in operation since the mid-1970s). As noted in the LA, the most significant earthquake source would be s a repeat of the 1886 Charleston, estimated to have a modified Mercalli intensity (MMI) at the SRS of VI–VII. The magnitude estimate of the 1886 Charleston earthquake is *M* 7.3 ± 0.3 (Johnston, 1996; Ambraseys, 1988). Other significant historical earthquakes felt at the SRS include the 1913 Union County earthquake (an MMI of VII at the epicenter and an MMI of II–III at Aiken, SC); the 1811–12 New Madrid earthquakes (*M* greater than 8.0 at New Madrid, MO); and the 1897, Giles County, VA, earthquake (MMI of VII, *M* 5.6 at Pearisburg, VA).

Paleoliquefaction features indicate that the Charleston-type earthquake has recurred at least seven times in the last 6,000 years (Talwani and Schaeffer, 2001). These prehistoric earthquakes appear to be restricted to the Carolina Coastal Plain (Talwani and Schaeffer, 2001). Two scenarios have been proposed to explain the distribution of the paleoliquefaction features. In the first scenario, the earthquakes occurred at Charleston, Georgetown, and Bluffton, SC. In the second scenario, all the prehistoric earthquakes occurred at Charleston. Hu, et al. (2002) concluded that the paleoliquefaction features were produced by earthquakes with magnitudes between 5.3 and 7.8.

No definitive geologic evidence has yet been discovered to tie the 1886 Charleston earthquake to a causative seismogenic fault. Tarr, et al. (1981) defined the Middleton Place—Summerville Seismic Zone to include the known distribution of seismicity and paleoseismicity associated with the Charleston-type earthquake. The Middleton Place—Summerville Seismic Zone is located 20 km (12 mi) northwest of Charleston, SC. Based on geological and geophysical data, Marple (1994), Madabhushi and Talwani (1993), and Marple and Talwani (2000) all inferred that complex and interactive strike slip and reverse faulting associated with the northwest trending Ashley River fault and the north-northeast trending Woodstock fault were the most likely causes of the Charleston earthquake. Recently, Weems and Lewis (2002) concluded that the region around Charleston, SC, is an active tectonic zone that accommodates differential movement between the Cape Fear arch and the Southeast Georgia embayment. All these models are consistent with the source characterization of the Charleston-type earthquake presented in the EPRI (1989) and Lawrence Livermore National Laboratory (LLNL) (NRC, 1994) probabilistic seismic hazard assessment (PSHA) studies.

Near the SRS, instrumented historical seismic records indicate that seismicity associated with the SRS and surrounding region is closely related to the earthquake activity within the South Carolina Piedmont (Bollinger, 1992). This activity is characterized by shallow, small-magnitude, and infrequent earthquakes. A search of the National Earthquake Information Center and Council of the National Seismic System showed that the vast majority of these earthquakes are M 3 or less. The largest magnitude earthquakes in the record are the 1974 M 4.9 and M 4.7 events. All instrumented earthquakes on the SRS itself were M 2.7 or less.

1.3.1.5.1.3 Earthquake Recurrence

The long repeat times (more than 500 years) and relatively brief historical record (less than 350 years), coupled with the absence of active surficial deformation, limit estimates of earthquake recurrence for a Charleston-type earthquake. The most complete record of the temporal and spatial distribution of large prehistoric earthquakes comes from identification of earthquake-induced liquefaction features called sand blows. Numerous sand blows have been identified throughout the South Carolina coastal area, but few if any outside this region (Westinghouse SRC, 2000a). Recent reanalysis of the paleoliquefaction investigations in South Carolina and the recalibrated ages using the radiometric ^{14}C dating technique suggest that as many as seven large-magnitude earthquakes occurred in the Charleston region within the last 6,000 years (Talwani and Schaeffer, 2001). These results translate to a recurrence interval for the Charleston-type earthquake of 500 to 600 years. This estimated recurrence interval is conservative because it assumes the maximum number of possible paleoearthquakes using the age constraints derived from the ^{14}C age data. Talwani and Schaeffer (2001) used 1σ error ranges to develop their list of age-distinct paleoearthquakes. Overlap of the ^{14}C ages using 2σ error ranges, as advised by Tuttle (2001), would result in a smaller number of age-distinct paleoearthquakes during this same 6,000-year interval and thereby increase the recurrence interval. Nevertheless, the recurrence interval of 500 to 600 years for the Charleston-type earthquake is consistent with the LLNL and EPRI PSHA studies.

Staff Review of Seismic Source Characterization

The staff reviewed the seismic source information presented in the LA and finds it sufficient because the applicant identified and assessed all of the potentially significant seismic sources related to the SRS (including, but not limited to, the Charleston seismic zone). The characterization of the tectonic setting and identification of capable seismic sources are based on extensive review of the published geological literature, regional and site geological and geophysical data, historical and instrumental seismicity data, regional stress field analysis, and geological investigations of prehistoric earthquakes. The information follows guidelines presented in RG 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion" (NRC, 1997) and NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 2.5.2.2 (NRC, 1987). Criteria used to assess capable fault and areal source zones include those outlined in Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100, "Reactor Site Criteria," as well as those in "Natural Phenomena Hazards Characterization Criteria" (DOE, 1994).

Information provided by the applicant to determine the tectonic setting of the facility is presented in a coherent, well-documented discussion that provides an adequate technical basis to evaluate the seismic potential of the site. Specifically, documentation in the LA is sufficient to determine the earthquake potential of geological structures and potential tectonic zones (i.e., regions of uniform earthquake potential). The information provided in the LA is also sufficient to evaluate uncertainties associated with seismic source geometry (e.g., fault dip, width, segmentation, and depth of seismogenic crust) and recurrence models. Thus, the staff reviewed the information in the LA and finds it acceptable because the basic geologic and seismic characteristics of the site and vicinity are adequately detailed to allow investigation of seismic characteristics at the facility.

1.3.1.5.2 Ground Motion Attenuation

Seismic hazards used to define bedrock uniform hazard spectra at the SRS are based on the LLNL and EPRI PSHAs. The LLNL and EPRI bedrock uniform hazard spectra were averaged and then broadened using the site-specific spectral shapes to develop bedrock response spectra.

Ground motion attenuation models contained in the LLNL and EPRI hazard studies incorporated several models developed for the southeastern United States. These models represent state-of-the-art studies of ground motion attenuation characteristics in the region and have captured the diverse opinions in the scientific community.

The ground motion attenuation model used to develop site-specific spectral shapes was the band limited white noise/random vibration theory ground motion model (Hanks and McGuire, 1981; Boore, 1983). In applying this stochastic approach, the applicant used the layered crustal velocity model developed by Herrmann (1986), with some modifications; the EPRI median site attenuation model (Q-model); and the range of the EPRI site-dependent parameter kappa values (EPRI, 1989).

<u>Staff Review of Ground Motion Attenuation</u>

Ground motion attenuation models used in the LLNL and EPRI studies represent the current scientific understanding of ground motion attenuation in the southeastern United States. These attenuation models adequately capture uncertainty in ground motion estimates, including the potential for Moho bounce effects. For example, a recent ground motion attenuation model for the eastern United States (Campbell, 2003), which accounts for Moho bounce effects, yields ground motion estimates consistent with those derived using the LLNL and EPRI studies. Application of the LLNL and EPRI models to the SRS and, consequently, to the facility is considered acceptable. The NRC staff has previously accepted the LLNL and EPRI ground motion modeling for sites in the central and eastern United States (NRC, 1997).

The use of the stochastic model or numerically simulated ground motions in the central and eastern United States, instead of recorded ground motions, is consistent with common practice and the state of knowledge because sufficient strong motion data are lacking in this tectonic regime as a result of low seismicity rates. The staff accepted the approach in its review of the PSHA for the Paducah Gaseous Diffusion Plant (Center for Nuclear Waste Regulatory Analyses, 1999). In addition, the random vibration theory model has been shown to yield conservative results for crustal conditions in the eastern United States (Silva, 1989). Thus, the staff has determined that the applicant's ground motion attenuation modeling provides reasonable assurance of the accuracy of ground motion attenuation modeling.

1.3.1.5.3 Seismic Hazard Calculations

The applicant used the seismic hazard results from the LLNL and EPRI PSHAs to define bedrock uniform hazard spectra at the SRS. No other probabilistic seismic hazard calculations were conducted specifically for the SRS or the facility. The LLNL and EPRI hazard studies include site-specific hazard calculations for the SRS.

<u>Staff Review of Seismic Hazard Calculations</u>

The LLNL and EPRI studies represent the state-of-the-art probabilistic hazard studies in the southeastern United States. Application of these results to the SRS and, consequently, to the facility is acceptable. The NRC staff previously accepted the LLNL and EPRI data, seismic sources, seismic hazard methods, and results for sites in the central and eastern United States (NRC, 1997). Thus, the staff finds that the use of LLNL and EPRI hazard results is technically sound.

1.3.1.5.4 Savannah River Sitewide Rock and Surface Response Spectra

Westinghouse SRC developed the SRS-wide rock response spectra for the entire site (Westinghouse SRC, 1997). These are site-specific uniform hazard spectra for bedrock from the LLNL and EPRI seismic probabilistic hazard studies, broadened by using site-specific spectral shapes. The rock response spectra were used as the bases for developing bedrock time histories as input into site response analyses for the facility and the SRS-wide surface response spectra.

The SRS-wide surface response spectra are not directly used in the design of structures or in soil stability analyses for the facility. However, the applicant used them to justify the sufficiency of the selected design spectra for the facility.

The SRS-specific rock uniform hazard spectra for bedrock were developed following the guidance and methodologies outlined in "Natural Phenomena Hazards Assessment Criteria," Change Notice No. 1 (DOE, 1995). Probabilistic hazards were developed according to DOE PC-3 and PC-4 spectra. The DOE PC-3 and PC-4 spectra were developed following seismic design and evaluation criteria in "Natural Phenomena Hazards Design and Evaluation Criteria for DOE Facilities" (DOE, 1996) and Change Order No. 1 to that document (DOE, 2002). In

DOE's 1996 report, PC-3 and PC-4 categories have mean annual probabilities of exceedance for design ground motions at 5×10^{-4} and 1×10^{-4}, respectively. In terms of the annual return period ground motions, mean annual probabilities of exceedance of 5×10^{-4} and 1×10^{-4} correspond to mean 2,000-year and 10,000-year return period ground motions, respectively.

The development of the rock response spectra included the following procedures:

- The mean bedrock uniform hazard spectra were computed for two mean annual probabilities of exceedance, 5×10^{-4} and 1×10^{-4} (corresponding to performance categories of PC-3 and PC-4, respectively), by averaging the LLNL and EPRI mean uniform hazard spectra for the SRS.
- Site-specific spectral shapes were generated using EPRI mean magnitude and mean distance values based on the magnitude and distance deaggregation results at each probability of exceedance.
- The spectral shapes were then scaled to the corresponding mean bedrock uniform hazard spectrum at frequencies of 1 to 2.5 and 5 to 10 hertz (Hz).
- The resulting three spectra (the averaged LLNL and EPRI uniform hazard spectrum and the 1 to 2.5-Hz and the 5 to 10-Hz-scaled site-specific spectra) were then enveloped and smoothed to obtain the broadened bedrock response spectra for the PC-3 and PC-4 hazards.

<u>Sitewide Surface Response Spectra</u>

Sitewide surface response spectra were obtained by multiplying the broadened bedrock uniform hazard spectra by frequency-dependent site amplification factors to account for soil effects. In deriving site amplification factors, hypothetical bedrock spectra were vertically propagated through soil columns representative of the site soil conditions using the one-dimensional equivalent linear analysis procedure developed by Silva (1989). The procedure was considered equivalent to SHAKE analyses summarized in Idriss and Sun (1992).

The hypothetical bedrock spectra were power spectral density functions and spectral accelerations for a suite of PGAs at the soil/bedrock interface (bedrock motions described previously) and were developed using the random vibration theory model (Boore, 1983). Three magnitude- and distance-dependent spectra were developed for each control motion acceleration representing the 5th, 50th, and 95th percentile contributions to the probability of exceedance. Again, the magnitude and distance pairs were obtained from the EPRI deaggregated hazard results.

The calculation of the site amplification factors considered SRS-wide variability in velocity profile, soil column thickness, bedrock velocity, and dynamic properties (Westinghouse SRC, 1997). Soil conditions characterized in the most recent study of the site's geotechnical properties (Duke Cogema Stone & Webster (DCS), 2003) are consistent with subsurface conditions reported in the previous geotechnical reports for the site. The sitewide, uniform-hazard-based response spectrum was taken as the envelope of all soil response spectra obtained by multiplying the broadened mean bedrock uniform hazard spectra by the site amplification factors for different soil/bedrock categories, scaling frequencies, and magnitude levels. As with the design ground motions, the site-specific soil spectra were shown to envelop the Charleston earthquake spectra.

Staff Review of Savannah River Site Rock and Surface Response Spectra

The LLNL and EPRI studies represent the state-of-the-art probabilistic hazard studies for the southeastern United States. Applying these probabilistic hazard results to the SRS and, consequently, to the facility is acceptable. The NRC staff previously accepted the LLNL and EPRI ground motion modeling for sites in the central and eastern United States (NRC, 1997). In addition, broadening the LLNL and EPRI bedrock uniform hazard spectral shapes and the development of surface response spectra is consistent with the methodologies of DOE (1995). These methodologies and procedures are well established within the ongoing seismic program at the SRS. Westinghouse SRC and DOE have extensively reviewed these site-specific adjustments. Thus, the staff has determined that SRS-wide rock and surface response spectra provides reasonable assurance that potential seismic hazards are sufficiently estimated.

1.3.1.5.5 Design Spectra and Site Response Analyses

1.3.1.5.5.1 Design Spectra

The design-basis ground motions proposed by the applicant for the surface facilities are an RG 1.60 spectra (NRC, 1973) anchored at 0.20g PGA, which is the same spectra used for the design of the nearby Vogtle Nuclear Power Plant (licensed under 10 CFR Part 50). More recently, regulations at 10 CFR 100.23, “Geologic and Seismic Siting Criteria,” for nuclear power plants have been updated to include the application of probabilistic methods to the assessment of seismic hazards. RG 1.60 provides general guidance for determining the safe-shutdown earthquake for new nuclear reactors based on a PSHA, consistent with the regulatory requirements of 10 CFR 100.23. RG 1.60 recommends a reference median annual probability of exceedance of 1×10^{-5}. As shown by a similar DOE analysis (DOE, 2002, Appendix C), a median annual probability of exceedance of 1×10^{-5} corresponds approximately to a mean annual probability of exceedance of 1×10^{-4}.

The applicant performed evaluations in the safety analysis report that show that the 0.20g RG 1.60 spectra have mean annual exceedance probabilities that range between 1.6×10^{-4} and 4.5×10^{-5} (or equivalent return periods that range between 6,300 and 22,000 years (see DCS, 2001b, Enclosure B, Table 1). For frequencies between 2 and 10 Hz, the mean annual probabilities of exceedance are equal to or less than 1×10^{-4} (or equivalent return periods greater than 10,000 years). For higher frequencies up to the PGAs, the mean annual probabilities of exceedance are equal to or slightly greater than 1×10^{-4}. These mean annual exceedance probabilities are based on ground motions from the averaged EPRI (EPRI, 1989) and LLNL (NRC, 1994) seismic hazard results for the eastern United States.

The applicant selected RG 1.60 spectra (NRC, 1973) anchored at 0.20g because these were deemed to be conservative with respect to currently available site-specific information. The applicant has shown that this seismic design spectrum lies between the SRS-wide DOE PC-3 and PC-4 spectra. The current PC-3 and PC-4 sitewide spectra are based on ground motion spectra developed by Westinghouse SRC (1997) for the entire SRS. The PC-3 and PC-4 spectra were developed following seismic design and evaluation criteria in DOE (1996) and DOE (2002), as discussed in more detail in Section 1.3.1.5.4 of this SER.

To ensure safe operation of the structures, systems, and components (SSCs) beyond the design ground motions, DOE (1996) and DOE (2002) developed performance goals associated with each performance category. The performance goals are defined in terms of the ability of the SSCs to perform essential safety functions during and after the natural hazard phenomena (in this case, an earthquake). The acceptable behavior limit for normal-use SSCs, such as buildings, is major damage, but the damage is limited in extent such that the occupants can safely exit the building. For more critical SSCs, such as nuclear containment structures, damage at the performance goal should be limited such that the containment is not compromised. In DOE (1996) and DOE (2002), the seismic ground motion performance goals for PC-3 and PC-4 SSCs were established with a mean annual probability of exceedance of 1×10^{-4} and 1×10^{-5}, respectively.

In DCS (2001a), the applicant indicated that the desired performance goal probability is based on the approach recommended in DOE (1996) and DOE (2002). That assertion is supported by performance calculations (DCS, 2002, Enclosure B) showing that many of the SSCs performed their safety functions to ground motion levels with a mean annual probability of exceedance of 1×10^{-5} or less. These calculations support the conclusion that the design criteria—RG 1.60 (NRC, 1973) spectra anchored to the 0.20g PGA, which is significantly greater than the site-wide PC-3 spectra—are adequate to ensure the safe design of the facility.

The applicant also showed that the design spectra envelop the deterministic spectra for a repeat of the Charleston-type earthquake. This deterministic check analysis follows the requirements of DOE (1995) using the largest historic earthquakes within 121 km (75 mi) having a moment magnitude greater than 6. In this analysis, the deterministic median bedrock and soil spectra were generated for the 1886 Charleston earthquake using median source parameters, a source-to-site distance of 200 km (124 mi), and other parameters used in generating response spectra based on uniform hazards. The applicant evaluated the vertical-to-horizontal seismic spectral ratios for the facility (MOX, 2006a). The results show that the vertical-to-horizontal spectral ratios could exceed the standard generally used at the SRS (normally the vertical is assumed to be two-thirds of the horizontal), particularly for frequencies greater than approximately 3 Hz. Thus, the applicant has agreed to use both the horizontal and vertical spectra in RG-1.60 anchored at 0.20g PGA.

1.3.1.5.5.2 Site Response Analyses

The applicant indicated that the sitewide response spectra are intended for simple response analysis and are not appropriate for soil-structure interaction and soil stability analyses. It further indicated that the sitewide response spectra represent a surface response—not an embedded response. For soil stability and soil-structure interaction analyses, the applicant established a one-dimensional, free-field site response analysis procedure (DCS, 2001a). The control ground motions for site response analyses include the modified PC-3 motion and the 1886 Charleston motion. The modified PC-3 motion is the SRS-wide PC-3 rock response spectrum increased by a factor of 1.25 (PC-3+ rock spectrum) to yield a bedrock PGA of 0.14g. This modified motion would result in a design surface PGA of 0.20g at the facility through site response analyses. The 1886 Charleston motion is the 50th percentile attenuated rock motion at the actinide packaging and storage facility site. The applicant used it to evaluate the liquefaction potential associated with large, distant earthquakes. Westinghouse SRC developed the spectrum-compatible acceleration time histories for both of these design motions.

Site response analyses were conducted using PROSHAKE, a Windows version of SHAKE91 (Idriss and Sun, 1992). The design motion time histories were applied at the base of the soil column. Properties for the soil column were developed from geotechnical studies specific to the facility (Westinghouse SRC, 2000b; DCS, 2003 and 2001b). Soil conditions characterized in the most recent study of the site geotechnical properties (DCS, 2003) are consistent with subsurface conditions reported in the previous geotechnical reports of the site. The cyclic stress ratios computed from the site-response analyses were input into the dynamic soil-structure interaction analyses of the critical structures and into the liquefaction analyses.

Results from site response analyses show that the PC-3+ bedrock time history produces a surface PGA of 0.20g and a surface spectrum that correlates well with the RG 1.60 (NRC, 1973) surface spectrum anchored at 0.20g PGA. Thus, the applicant concluded that the PC-3+ bedrock spectrum satisfies the requirement for a bedrock time history that can be used for dynamic analysis at the facility.

Staff Review of Site Response and Design Ground Motion

The staff finds it acceptable to use the RG 1.60 spectrum (NRC, 1973) anchored at 0.20g PGA. The applicant analyzed several SSCs using the RG 1.60 design spectrum to demonstrate that the performance objectives necessary for highly unlikely events with potentially high consequences set forth in NUREG-1718 (NRC, 2000) are met. Similarly, the staff finds the analysis procedures, input bedrock time histories, and soil column properties for soil stability analyses and soil-structure interaction analyses acceptable. The resulting surface response spectrum exceeds the DOE PC-3 spectrum and is comparable to the design spectrum. The applicant further verified that the SRS-wide PC-3 spectrum is applicable to the design of the MFFF through examinations of the soil stratigraphy, soil column thickness, bedrock type, velocity profile, and geologic formations at the MFFF site. Use of response spectra that envelop the SRS-wide PC-3 spectrum to analyze soil and subsurface stability of the MFFF site is therefore conservative.

1.3.1.5.6 Surface Faulting Hazard

The LA summarizes the tectonic structures of interest in the SRS and surrounding region, including faults, folds, arches, basins, and paleoliquefaction features that resulted from past earthquakes. Many of these features are vestiges of the contractional tectonism that characterized the Appalachian Orogen from the Late Precambrian through the Late Paleozoic (about 1,100 to 245 million years) and rifting and extensional tectonism that characterized the break up of Pangea and the opening of the Atlantic Ocean in the Triassic and Early Jurassic periods (about 245 to 180 million years). Although reactivation of some of these features has been proposed to explain the origin of the Charleston-type earthquake, none of these features has an impact on direct-faulting hazards at the SRS.

Faulting of the Atlantic Coastal Plain sediments is evident from geologic and geophysical data (e.g., Prowell and Obermeier, 1991). Most of the faults are moderately to steeply dipping reverse faults, although some small normal faults were noted in the Late Cretaceous and Early Tertiary strata (100 to 37 million years). Maximum displacements are less than 80 m (250 ft), and displacements become progressively smaller in younger sediments, suggesting that faulting was coeval with deposition.

At the SRS, the PB fault has been identified as the primary structural feature of interest to a potential faulting hazard. This fault appears to be an upward propagation of the boundary fault on the northern side of buried Dunbarton Basin, a Triassic to Early Jurassic rift feature. This boundary fault was originally a down-to-the-southeast normal fault, but was reactivated as an up-to-the-southeast reverse fault in the Late Cretaceous and Early Tertiary (100 to 37 million years). Extensive geological and geophysical evidence summarized in the LA documents that the PB fault has not been active in the last 500,000 years and probably was not active in the

Quaternary (last approximately 2 million years). Thus, the PB fault is not deemed capable, according to criteria established in Appendix A to 10 CFR Part 100.

Staff Review of Surface Faulting Hazard

The staff reviewed the information in the LA and finds it acceptable because the potential for surface faulting of the site and vicinity has been adequately assessed using historical seismic data and analyses. The evidence is sufficient to conclude, with reasonable assurance, that surface faulting hazards do not exist at the SRS.

1.3.1.6 Stability of Subsurface Materials

The objective of the staff review in this section is to determine, with reasonable assurance, whether characterization of the stability of the subsurface materials for the facility is adequate for foundation design for the civil structural systems. The staff reviewed the following areas concerning subsurface material stability that were applicable to the safety analysis and design of the proposed facility:

- soil liquefaction potential assessment
- soft zone characterization
- slope stability assessment

1.3.1.6.1 Soil Liquefaction Potential Assessment

Section 1.3.5.3.4.3, “Post-Rift and Cenozoic Structures,” of the LA provides information regarding paleoliquefaction at the SRS where the facility is located. The LA indicates that no systematic reconnaissance surveys in search of paleoliquefaction evidence within the geomorphic and geologic environment of the SRS were performed in the past because of limited access, high water table conditions, dense vegetative cover, and few exposures.

For seismically induced liquefaction to occur and be identified, the following conditions must be met (MOX, 2006a):

- presence of Quaternary-age unconsolidated deposits
- presence of a shallow ground water table
- proximity to potential seismogenic features
- quality and extent of exposure

According to these conditions, young fluvial terraces at or slightly above the level of the modern flood plain and Carolina bays may have the highest potential for generating and recording Holocene (last 10,000 years) and Quaternary (last approximately 2 million years) seismically induced liquefaction.

Limited investigation of the exposed young fluvial terraces along the Savannah River adjacent to the SRS suggests that most of the exposed deposits were clay and silt and thus have a low liquefaction potential. Although local clean sand deposits with a high liquefaction potential exist, evidence of seismically induced liquefaction is not observed (MOX, 2006a). In general, these young fluvial deposits are historical in age. In historical times, no strong ground motions occurred in the SRS area. Consequently, evidence for seismically induced liquefaction in the young fluvial deposits may not exist.

According to the LA, potential paleoliquefaction for the flood plain deposits at depth is likely. Evaluation of postdepositional features associated with the upland areas at the SRS, however, suggests that they are not related to seismically induced liquefaction (MOX, 2006a).

Section 1.3.7.1 of the LA discusses liquefaction susceptibility at the facility. Detailed soil geotechnical testing data, as documented in three facility site geotechnical reports (DCS, 2003 and 2001a and b), support this discussion. The site geotechnical reports present properties of soils, including soil classifications, particle-size distributions, water contents, plasticity indices, liquid limits, blow counts from standard penetration tests, tip shear resistances from cone penetration tests, and shear wave velocities.

The liquefaction potential of the facility site within the proximity of the MFFF and emergency generator buildings was evaluated using the cyclic stress approach described by the National Center for Earthquake Engineering Research (1997). This approach is acceptable to the staff for a liquefaction potential investigation because it represents the state-of-the-art procedure. This procedure can evaluate liquefaction resistance of soils under level to gently sloping ground; the surface gradient at the proposed facility is gently sloping (as shown in Figures 1.3.1-1, 1.3.4-5, and 1.3.7-1 of the LA).

In the most recent of the geotechnical reports (DCS, 2003), liquefaction potential was evaluated for the 95 soil columns from cone penetration tests and 14 soil columns from standard penetration tests. Cyclic stress ratio and cyclic resistance ratio are two important parameters for assessing liquefaction. The cyclic stress ratios for the aforementioned penetration tests were estimated directly using the PROSHAKE computer program with a generalized soil profile. The liquefaction potential for 18 of the cone penetration tests was also estimated using the test-hole-specific soil profiles to compare with the results using the generalized soil profile. The LA assumed that full liquefaction was triggered if the factor of safety (cyclic stress ratio/cyclic resistance ratio) was equal to or smaller than 1.1. For factors of safety between 1.1 and 1.4, soil settlement may result because of the excessive water pressure buildup that reduces soil strength and stiffness.

The analysis results indicate that the liquefaction potential at the facility site is low. Only a few localized areas have been identified to be liquefiable or to have soil settlement potential because of excessive pore water pressure. The potentially liquefiable soils identified at the site are located in the lower Tertiary (about 65 to 33 million years) Tobacco Road, Dry Branch, and Santee formations. The LA indicates that the analysis results are conservative because the analysis did not consider the effects of soil aging and the cohesiveness of the soils for the cone-penetration-based results.

Staff Review of Soil Liquefaction Potential Assessment

The staff reviewed the information presented in the LA and finds reasonable assurance that paleoliquefaction at the SRS was sufficiently discussed to support the design of the IROFS structures of the facility. The staff concurs that the analysis of liquefaction potential in the proximity of the MFFF demonstrated a conservative approach and is acceptable. Section 11.1 of this SER evaluates the review of the effect of the seismically induced settlements caused by either liquefaction or excessive water pressure buildup in developing design criteria for the facility IROFS structures.

1.3.1.6.2 Soft Zone Characterization

Soft zones in the soils are unique features of the SRS. Section 1.3.5.1.5.5, "Carolina Bays," of the LA discusses the origin of the soft zones. Section 1.3.7.2, "Evaluation of Soft Zones," of the LA discusses the characterization of the soft zones at the facility and is supported by site geotechnical data (DCS, 2003 and 2001a).

The soft zones are often found in the Tinker/Santee Formation, particularly in the upper third of this section. These soft zones consist of weak material zones interspersed in stronger carbonate-rich matrix materials. Soft zones may pose a concern for foundation design by developing undesirable soil settlement not accounted for in the design. In engineering terms, a soft zone is defined as a zone with a cone penetration test corrected tip resistance less than 1.44 megapascals (209 psi) or blow counts from a standard penetration test less than 5 over a continuous interval of at least 0.6 m (2 ft) (DCS, 2001a). In characterizing the soft zones, the applicant used these criteria to identify soft material zones not located in the Tinker/Santee Formation. The staff considered this approach prudent and acceptable.

Three site geotechnical reports (DCS, 2003 and 2001a and b) document the results of the site exploration program related to identifying soft material zones. The spacing of exploration holes in the vicinity of an identified soft zone was generally 27 m (90 ft) or less. The lateral extent of soft zones was conveniently estimated to be half of the exploration spacing. The exploration program identified soft zones in the vicinity and beneath the MOX fuel fabrication building with limited lateral extent. The thickness of these soft zones ranges from 0.91 to 2.13 m (3 to 7 ft).

Staff Review of Soft Zone Characterization

Based on the review of the soft zone information, the staff concludes that the applicant's exploration program sufficiently characterized the soft zones at the facility to support design of the IROFS. Section 11.11 of this SER evaluates the consideration of soft zone effects in developing design criteria for the IROFS of the facility.

1.3.1.6.3 Slope Stability

Section 1.3.7.3, "Slope Instability Hazard Evaluation," of the LA does not specifically discuss slope stability. In the evaluation of the natural phenomena that may occur at the site, however, debris avalanching and landslides were determined not to be applicable, because the site is relatively flat and no significant quantities of soil or rock are available in the surrounding area. An examination of topographic contours provided in Figures 1.3.4-5 and 1.3.7-1 of the LA confirms that the slopes at the facility site are relatively gentle and therefore pose no threat for instability or landslide. The staff site visit further confirmed that slope stability is not a safety concern at the site.

1.3.2 Evaluation Findings

The staff reviewed Chapter 1.3 of the LA and its supplementary information in accordance with Section 1.3 of NUREG-1718 (NRC, 2000). The applicant provided the geographic, demographic and land use, meteorologic, hydrologic, geologic, and seismic information relevant to the MFFF site. This information is current, appropriately referenced, and consistent with information in the safety assessments used to support the design bases of the IROFS structures. The staff finds that the applicant has accurately described the site so as to properly define potential accident conditions. Based on its review of the LA and the relevant supplementary information provided by the applicant, the staff further finds that the applicant has met the baseline design criteria in 10 CFR 70.64(a)(2) for natural phenomena hazards.

Based on the review, the staff concludes that the applicant's site description meets the regulatory requirements in 10 CFR 70.22, "Contents of Application," for a license to possess and use radioactive material.

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2.0 FINANCIAL QUALIFICATIONS

2.1 Conduct of Review

This chapter of the safety evaluation report contains the staff's review of the financial qualifications presented by Shaw Areva MOX Services (MOX Services) in Chapter 2 of the revised license application (LA) (MOX 2009) and the exemption request entitled "Request for Exemption from Indemnity Agreement and Financial Protection Requirements" dated September 27, 2006 (DCS 2006).

MOX Services is the applicant for the license to possess and use byproduct material, source material, and special nuclear material (SNM). MOX Services is incorporated in the State of South Carolina as a limited liability company (LLC) owned by Shaw Project Services Group, Inc. (SPSG), Shaw Environmental & Infrastructure, Inc. (SE&I), and AREVA, Inc. These three companies are the equity owners of the LLC (SPSG 40 percent, SE&I 30 percent, and AREVA 30 percent). MOX Services was formed to provide MOX fuel fabrication and other services to support the mission of the U.S. Department of Energy (DOE) for the disposition of U.S.-owned surplus weapons-usable plutonium.

2.1.1 Project Costs

In September 2000, the United States and the Russian Federation concluded a bilateral agreement on plutonium disposition, "Agreement Between the Government of the United States of America and the Government of the Russian Federation Concerning the Management and Disposition of Plutonium Designated as No Longer Required for Defense Purposes and Related Cooperation." Under the agreement, the United States will dispose of surplus weapons-grade plutonium. The mixed oxide fuel fabrication facility (MFFF) is intended to fulfill the U.S. obligation for disposition of that plutonium.

MOX Services operates the MFFF under a contract with DOE. During operations, DOE reimburses MOX Services for the full cost of operating the MFFF, minus fuel payments that MOX Services receives from the mission reactor utilities, plus possible incentive fees for performance. MOX Services does not intend to: (1) finance or rely on the proceeds from debt, (2) rely on equity securities, (3) rely on any other source of external financing other than DOE funding, and (4) does not rely on any revenue stream to cover such costs (with the exception of the revenue stream from the mission reactor utilities as described above). In light of the MFFF's importance to the U.S. obligation and congressional support for this program, the Federal Government has a significant continuing incentive to adequately fund the MFFF and to continue providing the necessary annual appropriations to support operation of the MFFF.

2.1.2 Financial Qualifications

Because the MFFF is a project funded by the U.S. Government, the specific financial resources and capabilities of MOX Services and its equity owners are not relevant to the determination of adequate financial resources to operate the facility. MOX Services does not intend to rely on its financial resources, or those of an equity partner or parent company, to provide financing.

MOX Services is not a publicly held entity, and thus, its financial statements are not publicly available. MOX Services previously submitted, under separate cover, proprietary financial statements providing information concerning its financial condition.

The structure of MOX Services reimbursement for the MFFF operation is designed to support the MFFF project as a viable business enterprise. Thus, MOX Services is financially qualified to safely operate the MFFF, and that financial qualification is supported by the Federal Government's obligation through the DOE-MOX Services contract for the MOX Project.

2.1.3 Exemption Request

As stated in Section 1.2.4.2 of the Construction Authorization Request (DCS 2004), DOE has agreed to indemnify MOX Services in accordance with the provisions of the Price-Anderson Act set forth in Section 170(d) of the Atomic Energy Act of 1954, as amended, 42 U.S.C. 2210(d).

Based upon the DOE Indemnity Agreement and for the reasons discussed below, MOX Services requested an exemption from the NRC's requirements concerning agreements of indemnification and related financial protection requirements set forth in 10 CFR §§ 140.20 and 140.13a. 10 CFR § 140.8 provides that: "The Commission may, upon application of any interested person or upon its own initiative, grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and are otherwise in the public interest."

MOX Services is a DOE contractor and is thus fully covered by DOE's nuclear liability protection under the Price-Anderson Act, as amended. Section 170(d)(1)(A) of the Atomic Energy Act provides that the DOE Secretary shall enter into agreements of indemnification with certain persons "who may conduct activities under a contract with the Department of Energy that involve the risk of public liability and that are not subject to financial protection requirements under subsection b. or agreements of indemnification under subsection c. or k." In accordance with this statutory authority, the contract between MOX Services and DOE contains the following "Nuclear Hazards Indemnity Agreement" excerpt from DOE Acquisition Regulations (DEAR 952.250-70), which fully indemnifies MOX Services and its subcontractors up to the statutory limit of liability:

> (d)(1) *Indemnification*. To the extent that the contractor and other persons indemnified are not compensated by any financial protection permitted or required by DOE, DOE will indemnify the contractor and other persons indemnified against (i) claims for public liability as described in subparagraph (d)(2) of this clause; and (ii) such legal costs of the contractor and other persons indemnified as are approved by DOE, provided that DOE's liability, including such legal costs, shall not exceed the amount set forth in section 170e (1)(B) of the Act in the aggregate for each nuclear incident or precautionary evacuation occurring within the U.S. or $100 millions in the aggregate for each nuclear incident occurring outside the U.S., irrespective of the number of persons indemnified in connection with this contract.

The public liability referred to in subparagraph (d)(1) of this clause is public liability as defined in the Price-Anderson Act, as amended, which "(i) arises out of or in connection with the activities under this contract, including transportation; and (ii) arises out of or results from a nuclear incident or precautionary evacuation, as those terms are defined in the Act." The DOE indemnity agreement with MOX Services provides full protection and coverage for public liability arising from operation of the MFFF.

The requested exemption from the requirements to enter into an indemnity agreement with the NRC and to maintain financial protection is authorized by law because:

(a) The Price-Anderson Act does not require the NRC to enter into indemnity agreements or to impose financial protection requirements in connection with a plutonium fuel fabrication facility, and;

(b) There is no statutory prohibition on granting the exemption.

Nothing in the Price-Anderson Act, the Atomic Energy Act, or any other statute precludes the NRC from granting the requested relief. In SECY-99-177 (Issue 7), the NRC Staff specifically stated that "[n]o additional legislation is needed" to support the requested exemption.

The requested exemption is in the public interest. DOE has entered into an indemnity agreement with MOX Services that provides for effective coverage in the event of a nuclear incident related to the MOX Facility. If MOX Services were required to enter into an indemnity agreement with the NRC and to obtain $200 million in financial protection (presumably via private insurance coverage), the cost of the financial protection would be passed through to the U.S. Government under the MOX Services-DOE contract. Thus, there would be no benefit, and additional costs would be incurred, by imposing the NRC's indemnity agreement and financial protection requirements.

2.2 Evaluation Findings

Based on its review of the exemption request, the staff finds that the requested exemption from the indemnity agreement and financial protection requirements of 10 CFR §§ 140.20 and 140.13a is authorized by law and in the public interest as discussed in Section 2.1.3 of this SER. This exemption will be included in the license to possess and use radioactive material that may be granted to the applicant after completion of other regulatory requirements in 10 CFR Part 70.

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3.0 PROTECTION OF CLASSIFIED MATTER

This chapter of the safety evaluation contains the staff's review to determine, with reasonable assurance, whether the applicant has established policies and procedures that meet the regulatory requirements for the protection of classified matter against loss or compromise. Classified matter includes secret and confidential National Security Information (NSI) and Restricted Data (RD) received or developed in conjunction with activities licensed, certified or regulated by the NRC.

3.1 Regulatory Requirements

The regulatory basis for the NRC's review of an applicant's protection of classified matter are the requirements described in the (10 CFR 70.22) contents of an application, in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 25, "Access Authorization," and in 10 CFR Part 95, "Facility Security Clearance and Safeguarding of National Security Information and Restricted Data." These regulations implement the requirements of the Atomic Energy Act of 1954, as amended, and Executive Order 12958, "Classified National Security Information," as amended, for the protection of Restricted Data and classified National Security Information. Chapter 3 of NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility" (NRC, 2000), provides the references for this information and the guidance for staff review.

3.2 Regulatory Acceptance Criteria

The acceptance criteria that the U.S. Nuclear Regulatory Commission (NRC) uses for reviews of practices and procedures for protecting classified matter are outlined in the NRC's "Standard Practice Procedures Plan Standard Format and Content for the Protection of Classified Matter at NRC Licensees, Certificate Holder, or Other Activities as the Commission May Determine" (referred to hereafter as the SPPP Standard Format) (NRC, 2006.) The SPPP Standard Format provides guidance to the licensee in meeting the regulatory requirements of 10 CFR Part 25 and 10 CFR Part 95. Where NRC criteria are not specified, the staff may use relevant guidance from the U.S. Department of Energy (DOE), Department of Defense, National Security Agency, or National Industrial Security Program Operating Manual as a basis for acceptance. The criteria for acceptance are also discussed in the Standard Review Plan for the Review of an Application for a Mixed Oxide Fuel Fabrication Facility (NRC 2000)

3.3 Staff Review and Analysis

The licensee prepared a detailed SPPP for the mixed oxide fuel fabrication facility (MFFF). The purpose of the MFFF facility security plan is to define and document the security measures related to the protection of classified matter.The NRC staff review evaluates the MFFF plan to assure compliance with NRC regulations. Federal laws, specified by 10 CFR Part 25 and 10 CFR Part 95, and applicable orders require that all appropriate access authorization and job-related need-to-know personnel assigned to the MFFF protect classified information from unauthorized access and disclosure. The MFFF plan establishes the criteria for granting, reinstating, extending, transferring, and terminating access authorization to employees, contractors, and other agents who may require access to classified information at the MFFF site. The MFFF plan also provides details regarding the facility's physical location and classified and unclassified mailing addresses. It also discusses the buildings and security receptacles where classified matter will be stored.

The MFFF Security Department has the responsibility and authority for enforcing security requirements governing the safeguarding of classified information and matter. Responsibility for plant security is vested in an MFFF Facility Security Officer (FSO). Plans, procedures, post orders, and similar documents designate the security responsibilities of all members of the MFFF security organization and authorized individuals performing security functions. Security matters that cannot be handled through normal procedure will be reported to the FSO or designee.

The FSO and designees are cleared to the level commensurate with the facility clearance. Based on need-to-know, other key management officials are granted specific access authorizations or will be excluded from classified access. The MFFF plan also addresses the types of classified matter that will be handled at the facility. In addition, the plan outlines the personnel access authorizations and security clearance required for individuals with the requirement for a need-to-know who may have access to classified information and may visit classified portions of the facility.

Classified matter, while unattended, will be stored in security containers or safes approved by the General Services Administration (GSA). These containers and safes have changeable three-position combination GSA-approved combination locks. Such containers (repositories) will be located within assigned and approved locations in the protected area or in other restricted areas on the MFFF site. Access to all areas that contain classified information will be restricted.

Classified matter, while in use, will be constantly attended by, or under the direct control of, an authorized individual (appropriately cleared and with a need to know) to preclude visual, physical, and audio access by persons who do not have the prescribed access authorization. Classified matter will be protected in accordance with 10 CFR 95.27, “Protection While in Use,” by the markings placed on it, by access control, by the requirement of a need to know, by the availability and use of prescribed storage containers (repositories), and by the means of prescribed transmission and destruction methods. Classified information may be established and controlled anywhere within the confines of restricted area(s) at the MFFF.

The security education program at the MFFF is developed and maintained by the MFFF Training Manager with oversight by the FSO or FSO designee. The security education program includes consideration and coverage of personnel access authorization requirements, the physical security features of the facility, and the classified nature of the work.

Designated derivative classifiers at MFFF are authorized to make classification determinations to classify NRC documents. The classifiers make such determinations based on issued guidance pertinent to the security plan. Information generated or possessed at the MFFF site that is believed to contain classified information will be protected and marked appropriately, in accordance with the guidelines of 10 CFR 95.37, “Classification and Preparation of Documents,” pending the review and signature of a derivative classifier.

Documents and matter containing classified information received or originated in connection with an NRC license or certificate will be transmitted only to NRC-approved security facilities.

Destruction of classified documents will be accomplished by use of National Security Agency approved shredders, which are located in restricted areas. Person(s) having the appropriate clearance, and need-to-know, will perform the destruction.

Any alleged or suspected violation of the Atomic Energy Act, Espionage Act, or other Federal statutes related to classified National Security Information (i.e., deliberate disclosure of classified information to persons not authorized to receive it and theft of classified information) will also be reported in accordance with 10 CFR 95.57, "Reports," to the NRC Region II Regional Administrator.

Standard Practice Procedure Plan

The applicant plans to use classified matter in support of operations at the MFFF. The MOX project is not considered inherently classified, but because of its work with strategic special nuclear material (SSNM), classified information related to the protection of the SSNM and limited technical information related to the project will be used.

3.4. Evaluation Findings

The "specific risk" of a loss or compromise of project-related classified information is the theft/diversion or radiological sabotage to the SSNM at the MFFF. There is also a "general risk" to the classified technical information associated with the project in that the classified matter will be in the form of information related to classified components. As described above, the applicant's submittals provided sufficient information, in accordance with 10 CFR Part 25 and 10 CFR Part 95, for the staff to determine that classified information will be adequately protected. Supporting factors in the staff's decision also include the licensee's oversight by DOE and the MFFF location within the DOE Savannah River Site.

REFERENCES

(DOD, 2006) U.S. Department of Defense, DOD 5229.22M, "National Industrial Security Program Operating Manual," Washington, DC, February 28, 2006.

(MOX, 2008) Shaw AREVA MOX Services, "Revised MFFF Classified Matter Protection Plan," Aiken, SC, May 21, 2008.

(NRC, 2006) U.S. Nuclear Regulatory Commission, "Standard Practice Procedures Plan Standard Format and Content for the Protection of Classified Matter at NRC Licensees, Certificate Holder, or Other Activities as the Commission May Determine," Washington, DC, June 2006.

(NRC, 2000) U.S. Nuclear Regulatory Commission, NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility," Washington, DC, August 2000.

10 CFR Part 25, Access Authorization.

10 CFR Part 95, Facility Security Clearance and Safeguarding of National Security Information and Restricted Data.

Atomic Energy Act of 1954, as amended.

Executive Order 12958, "Classified National Security Information," as amended, 1995
ended, 1995.

4.0 ORGANIZATION AND ADMINISTRATION

4.1 Conduct of Review

This chapter of the safety evaluation report reviews the organization and administration information presented in Chapter 4 of the revised license application (LA) to possess and use radioactive material at the mixed oxide (MOX) fuel fabrication facility (MFFF) (MOX, 2009). The staff used Chapter 4 in NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility” (NRC, 2000), as guidance in performing the review. The objective of the review is to determine whether organizational and administrative functions have been identified that will enable the applicant to plan, implement, and control site activities in a manner that adequately ensures the safety of workers and individuals outside the controlled area, protect the environment, and meet the requirements of 10 CFR Part 70. This review ensures that the qualifications for key management positions are adequate. The applicant’s MOX Project quality assurance plan, also presents organizational information.

4.1.1 Organization

The applicant proposed a functional organization for facility management, quality assurance, production (operation), engineering, environmental safety and health, licensing, and support services. Operations, engineering, and environmental safety and health licensing are independent functions, allowing each organization to provide objective audits, assessments, and reviews. Independence means that none of these organizations reports administratively to another. The applicant provided proposed organization charts showing lines of responsibility and communications.

The president of Shaw Areva MOX Services (MOX Services or the Applicant) has overall responsibility for health, safety, and environmental matters for the MFFF. Reporting to the president are managers responsible for environmental safety and health, licensing, the plant manager, the engineering manager, the support services manager, and the quality assurance manager. The quality assurance manager has a direct line of communication to the president of MOX Services and is independent of responsibilities for costs or schedules.

The organizational information provided by the applicant describes clear and unambiguous controls and communications between organizational groups responsible for designing, constructing and operating the facility. Lines of communication, responsibility, and authority are clearly delineated in the organization chart. The president has overall responsibility for safety and nuclear fuel manufacturing activities at the facility.

4.1.2 Administration

The managers responsible for the key management functions are appropriately available to perform their duties. When they are absent, their duties may be delegated to other qualified personnel, as determined by the responsible manager. While these managers have the authority to delegate tasks to other individuals, the responsible manager retains the ultimate responsibility and accountability for compliance with applicable requirements.

MOX Services will utilize procedures to implement health, safety, and environmental functions associated with the MFFF and management measures that supplement items relied on for safety (IROFS). Plant procedures are formally controlled and approved. If a procedure cannot be adhered to, work is stopped and not resumed until the procedure has been corrected or changed.

4.1.3 Key Management Positions

The management positions described in Section 4.1.1 above have responsibilities for activities involving the proposed facility. The key management functions are (1) facility management, (2) quality assurance, (3) production, (4) engineering, (5) environmental safety and health, (6) licensing, and (7) support services. The managers of the key functions are responsible for IROFS and related activities. The applicant also identified the responsibilities and minimum qualifications for each of these positions.

The scope and number of key management positions are described appropriately for every management function involving the proposed facility. The qualification requirements for key management positions provide an adequate breadth and level of experience for their respective responsibilities and authorities. The staff filling key management positions will be available during the operational phases of the project.

4.2 Evaluation Findings

In accordance with Chapter 4 of NUREG-1718 (NRC, 2000), the staff reviewed the organization and administration described in the LA to possess and use radioactive material for the MFFF. The staff evaluated the proposed organization for operation; the administration of the project; and the responsibilities, qualifications, and authorities of key management positions. The proposed organization, administration, and key management position descriptions and qualifications are consistent with guidance in NUREG-1718 and meet the regulatory requirements for organization and administration in 10 CFR 70.22 and 70.23 and are, therefore, acceptable.

The staff concludes that the applicant's organization and administration provide reasonable assurance that the applicant has an acceptable organization, appropriate administrative policies, and qualified key management positions to satisfy the regulatory requirements for a license to possess and use radioactive material.

REFERENCES

(MOX, 2009) Shaw AREVA MOX Services, "Mixed Oxide Fuel Fabrication Facility—License Application," Aiken, SC, October 2009.

(NRC, 2000) U.S. Nuclear Regulatory Commission, NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide Fuel Fabrication Facility," Washington, DC, August 2000.

10 CFR Part 70, Domestic Licensing of Special Nuclear Material.

5.0 INTEGRATED SAFETY ANALYSIS

This chapter of the safety evaluation report (SER) contains the U.S. Nuclear Regulatory Commission (NRC) staff's review of the integrated safety analysis (ISA) provided by Shaw AREVA MOX Services (MOX Services or the applicant) in Chapter 5 of the license application (LA) (MOX, 2010a) and whose results are summarized in the ISA Summary for the Mixed Oxide Fuel Fabrication Facility (MFFF) (MOX, 2010b). The staff performed the review using guidance from NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility," issued August 2000 (NRC, 2000) (SRP). The objective of this review is to verify whether the applicant has established and committed to an organization and procedures related to performing and maintaining an ISA in accordance with the regulatory requirements provided in Subpart H, "Additional Requirements for Certain Licensees Authorized To Possess a Critical Mass of Special Nuclear Material," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 70, "Domestic Licensing of Special Nuclear Material." The review is necessary to verify that (1) commitments for performing and maintaining an ISA, as required by the regulations in 10 CFR Part 70, Subpart H, are provided, (2) the MFFF is adequately protected against internal and external events, and (3) the items relied on for safety (IROFS) identified by the applicant adequately protect against natural phenomena and the consequences of potential accidents. The staff evaluated the information provided by the applicant for the ISA by reviewing Chapter 5 of the LA, other related sections of the LA as needed, the ISA Summary, and supplementary information provided by the applicant.

5.1 Regulatory Requirements

The staff reviewed how the applicant's ISA-related information in the LA and ISA Summary addressed the following regulations:

- 10 CFR 70.61, "Performance Requirements"
- 10 CFR 70.62, "Safety Program and Integrated Safety Analysis"
- 10 CFR 70.64, "Requirements for New Facilities"
- 10 CFR 70.65, "Additional Content of Applications"
- 10 CFR 70.72, "Facility Changes and Change Process"

More specifically, 10 CFR 70.62(a) requires the applicant to establish and maintain a safety program that demonstrates compliance with 10 CFR 70.61. The regulations in 10 CFR 70.62(a)(2) require that the applicant establish and maintain records that demonstrate compliance with the safety program. The requirement to maintain records of IROFS failures is in 10 CFR 70.62(a)(3). The regulation in 10 CFR 70.62(b) requires an applicant to maintain process safety information (PSI) that supports the performance and maintenance of an ISA. The requirement to conduct and maintain an ISA is specified in 10 CFR 70.62(c). In addition, 10 CFR 70.62(c) specifies the requirements for the tasks comprising the ISA process (i.e., identification of radiological, chemical, and facility hazards) and the qualifications of ISA team personnel used to ensure the adequacy of the ISA. This regulation further requires the ISA to evaluate whether the applicant's facility, with its listed IROFS, meets the safety performance requirements of 10 CFR 70.61. The requirement to establish management measures to ensure compliance with the performance requirements of 10 CFR 70.61 is in 10 CFR 70.62(d). The regulations in 10 CFR 70.65(a) specify the requirements for having a description of the safety program specified by 10 CFR 70.62, while the requirements for the content of an ISA Summary can be found in 10 CFR 70.65(b)(1)–(9). The regulations of 10 CFR 70.64 specify design criteria requirements for new facilities. Additionally,10 CFR 70.72 provides requirements for keeping the ISA and its supporting documentation current and for determining whether NRC preapproval is needed when facility changes are made. SRP Section 5.4.3.2 outlines the acceptance criteria for NRC's review of the applicant's ISA.

5.2 Safety Program

Section 5.1 of the LA describes the elements of the safety program for the MFFF. In this section, the applicant commits to a safety program that consists of PSI, an ISA that analyzes MFFF hazards and potential accident sequences and identifies IROFS, and management measures to ensure that those IROFS identified in the ISA Summary are available and reliable to perform their safety function when needed.

The staff review finds that, consistent with the requirements of 10 CFR 70.62(a) to establish and maintain a safety program, the requirements of 10 CFR 70.65(a) to include a description of the applicant's safety program, and the guidance in SRP Section 5.4.3.2.A, the applicant's description of the safety program and the content of the program provided in Section 5.1 of the LA are acceptable.

The applicant also committed in Section 5.1.5 of the LA to maintain records of IROFS failures, as required by 10 CFR 70.62(a)(3). The applicant provided commitments to ensure that deficiencies in IROFS or failures of management measures are addressed in accordance with the corrective action program described in the MOX Project Quality Assurance Plan (MPQAP). The applicant committed to maintain records of failures so that they are readily retrievable and available for inspection by the NRC and to document each discovery that an IROFS or management measure has failed to perform its function upon demand or has degraded such that the performance requirements of 10 CFR 70.61 are not satisfied. The applicant will maintain records that identify the IROFS or management measure that has failed and the safety function affected, date of discovery, date of failure, and duration of time that the item was unable to perform its function. The records will also contain other affected IROFS or management measures and their safety function, affected processes, cause of failure, whether the failure was in the context of the performance requirements or upon demand or both, and corrective or compensatory action taken. The applicant has committed to record failures at the time of discovery and to update the record promptly upon the conclusion of the failure investigation. Given the commitments provided by the applicant regarding maintaining records of failure and the information included in these records, the staff finds that the requirement in 10 CFR 70.62(a)(3) has been met.

5.2.1 Process Safety Information

In Section 5.1.1 of the LA, the applicant committed to compile and maintain current written PSI for the MFFF to identify and understand the hazards associated with the processes and to update the ISA as required. For the MFFF, the applicant defined PSI to include descriptions of the hazards, equipment used in the process, and the technology of the process.

Consistent with the requirement in 10 CFR 70.62(b) and the guidance in SRP Section 5.4.3.2 to maintain PSI to enable the performance and maintenance of an ISA, the staff finds the applicant's commitments for compiling and maintaining PSI, as well as the information included in these records, as described in Section 5.1.1 of the LA, to be acceptable.

5.2.2 Integrated Safety Analysis

The applicant performed an ISA to demonstrate compliance with the performance requirements described in 10 CFR 70.61. The ISA identified the plant's internal and external hazards and their potential for initiating events; potential accident sequences and their likelihood and consequences; and the structures, systems, and components (SSCs) and activities of personnel that are relied on for safety. The ISA supported preparation of the ISA Summary, which summarizes the results and conclusions of the ISA process. In Section 5.1.2 of the LA, the applicant committed to conduct an ISA with a level of detail commensurate with the complexity of the processes and to maintain the ISA during all phases of the facility life cycle. The staff finds that, consistent with the requirement in 10 CFR 70.62(c)(1) and the guidance in SRP Section 5.4.3.2.A to conduct and maintain an ISA, the applicant's commitments in the LA to conduct and maintain an ISA are acceptable.

Table 5.1-1 in the LA provides the consequence severity categories for meeting 10 CFR 70.61 performance requirements; Table 5.1-2 presents the event risk matrix showing when IROFS need to be applied to meet 10 CFR 70.61 performance requirements. Table 5.1-3 outlines the limits for hazardous chemicals used in the chemical processes at the MFFF. The staff review finds that the consequence severity categories provided in Table 5.1-1 and the risk matrix provided in Table 5.1-2 of the LA are consistent with the requirements in 10 CFR 70.61(b) and 10 CFR 70.61(c) and Section 5.4.3.2.B of the SRP; therefore, they are acceptable. Chapter 8 of this SER addresses the evaluation of the chemical limits found in Table 5.1-3.

5.2.3 Management Measures

The applicant provided the management measure program commitment to implement and maintain management measures in Section 5.1.3 of the LA. Management measures are applied to IROFS by providing the administrative and programmatic framework for configuration management, maintenance, training and qualification, procedures, audits and assessments, incident investigation, and records management. The management measure commitments are consistent with the requirement in 10 CFR 70.62(d) to establish management measures to ensure compliance with the performance requirements of 10 CFR 70.61. The staff review determined that these commitments were acceptable for meeting the 10 CFR 70.62(d) requirements. The applicant implements and maintains these management measures, as described in Chapter 15 of this SER, to ensure the required reliability and availability of the IROFS. Section 5.2.5.2.4 of the LA describes the application of management measures to IROFS, which the staff has addressed in the review of Chapter 15 of this SER.

5.3 Areas of Review

5.3.1 Safety Assessment of the Design Bases

The review performed by the staff for the construction authorization (CA) (see NUREG-1821, "Final Safety Evaluation Report on the Construction Authorization Request for the Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina," issued March 2005 (NRC, 2005)), included a review of the description of the plant site and a safety assessment of the design bases that demonstrated that the applicant's principle structures, systems, and components (PSSCs) would protect against natural phenomena and the consequences of other accidents, in accordance with the performance requirements of 10 CFR 70.61. Following the staff's review, and in accordance with 10 CFR 70.23(b), the Commission granted a construction approval allowing the applicant to begin constructing the facility. At that time, the staff also reviewed the methodology used to determine hazards to the facility, the criteria for meeting the performance requirements of 10 CFR 70.61, and the definitions of highly unlikely, unlikely, and credible provided by the applicant.

The safety assessment of the design bases used a method that was similar to that used in the ISA for the process hazards analyses and demonstration of compliance with the performance requirements of 10 CFR 70.61. This assessment also evaluated the acceptability, in terms of the consequences, likelihoods, and application of controls, of the external and internal hazards of the facility. Although the analysis performed to support the CA was a precursor to the ISA, it was not a substitute for the ISA that was submitted with the application for a license to possess and use special nuclear material. The safety assessment of the design bases allowed the staff to determine that (1) the applicant's design bases were sufficient to meet the requirements of 10 CFR 70.23(b) and (2) the applicant, by using the safety assessment of the design bases, was building a foundation for the ISA to support the LA. The processes the applicant used to develop the safety assessment for the design bases, and which were approved by the staff, were analogous to the processes that the applicant used to develop the ISA. However, the previous approval of the methods used for the safety assessment of the design basis does not ensure that the methods for the ISA have been implemented acceptably, the performance requirements can be met, or the PSSCs described in the construction authorization request (CAR) represent the complete set of IROFS in the ISA Summary. The staff has reviewed and evaluated the ISA-related processes and implementation of the methods used to perform the ISA and compliance with the performance requirements of 10 CFR 70.61 in this and other technical safety sections of the SER.

5.3.2 The Integrated Safety Assessment in the License Application

The staff reviewed the application for a license to possess and use special nuclear material to determine the adequacy of the applicant's commitments associated with performing and maintaining an ISA. The areas of review relating to the performance of an ISA are discussed below:

- The applicant described its ISA commitments, which included procedures for the following:

  - performing and updating the ISA
  - review responsibility
  - ISA documentation
  - reporting ISA Summary changes
  - maintenance of ISA records

  The configuration management program described in Chapter 15 of this SER will control the applicant's ISA management procedures.

- The applicant described its commitment to compile and maintain a current and accurate set of PSI, including information on the hazardous materials, equipment, and technology used in each process. The applicant provided this commitment in Section 5.1.1 of the LA, consistent with the requirement in 10 CFR 70.62(b) and Section 5.4.3.2.A of the SRP. The staff found that commitment, if implemented as stated, is acceptable.

- The applicant described its commitments for meeting the requirements of 10 CFR 70.62(c)(2) and discussed the ISA team makeup and qualifications in

Section 5.3 of the LA. The applicant provided the required makeup of the ISA team and the technical areas that could be selected depending on the process and associated hazards evaluated. The applicant also provided detailed descriptions of the responsibilities and qualifications for the ISA team leader, scribe, engineer, and discipline experts. The staff finds that the applicant provided sufficient content and detail to meet the 10 CFR 70.62(c)(2) requirement and the provisions of SRP Section 5.4.3.2.A and therefore is acceptable. Section 5.3.3 of this SER provides additional discussion regarding the ISA team in its review of the ISA Summary contents.

- The applicant described its commitment to ISA methods, the ISA method selection criteria, and the specific methods that were and would be used for implementing the ISA process for particular classes of process nodes. The review performed by the staff evaluated the applicant's steps for conducting the ISA process as described in Section 5.2 of the LA. These steps include the following:

  – Identify internal facility hazards, natural phenomena hazards (NPHs), and external manmade hazards (EMMHs) that could affect the safety of licensed material.

  – Identify radiological hazards related to possessing or processing licensed material at the facility.

  – Identify chemical hazards of licensed material and hazardous chemicals produced from licensed material.

  – Develop potential events involving the identified hazards.

  – Determine the consequence and the likelihood of potential events and the methods used to determine the consequences and likelihoods.

  – Determine IROFS and the characteristics of their preventive, mitigative, or other safety function, as well as the assumptions and conditions under which the item is relied upon to support compliance with the performance requirements of 10 CFR 70.61.

Hazard identification is performed to identify the hazardous materials and hazardous energy sources associated with the operations of the MFFF process and auxiliary units. The ISA team utilized a checklist of hazardous materials and hazardous energy sources in the hazard identification process. The checklist was developed and used in accordance with the Checklist Analysis and What-If/Checklist methods found in the American Institute of Chemical Engineers, "Guidelines for Hazard Evaluation Procedures—Second Edition—With Worked Examples," (AIChE, 1992). The checklist was tailored for the MFFF and includes hazardous material, energy sources, confinement types, and auxiliary systems. A chemical interaction matrix is used to identify chemical hazards introduced by the mixing of incompatible chemicals and reagents. The matrix is facility specific and includes the chemicals and reagents used at the MFFF.

The ISA process identified NPHs and external manmade hazards. The applicant also used a checklist analysis to identify NPHs and external manmade hazards that may affect the MFFF. The list was developed through an extensive documentation review of Savannah River Site

(SRS) information, including site maps, site visits, and the SRS generic safety analysis report. The applicant also used information provided in NRC regulatory requirements, U.S. Department of Energy (DOE) guidance documents, DOE orders, and NRC NUREG reports to identify potential external events.

The process hazards analysis (PrHA) is used to develop and evaluate potential events involving the identified hazards. The applicant performed PrHAs for each process unit to identify specific event scenarios in detail, including causes of the events, and associated prevention and mitigation features (IROFS). All modes of operation are considered, including startup, normal operation, shutdown, and maintenance. PrHAs are performed in accordance with the guidance provided in AICHE (1992) and NUREG-1513, “Integrated Safety Analysis Guidance Document,” issued in 1999 (NRC, 1999).

AICHE (1992) and NUREG-1513 provided guidance on selecting the specific PrHA methodologies utilized for each process unit.

Hazard of Operability Studies (HAZOPs) and What-If/Checklists were the main techniques the applicant used to evaluate MFFF events. Supplemental hazard evaluations may have been performed in specific instances to support the ISA. These supplemental analyses are performed to gain insight into event likelihoods, event sequences, single failure vulnerability, and other safety aspects of hazards evaluation and may include such techniques as preliminary hazards analysis, failure modes and effects analysis, fault tree analysis, and event tree analysis. Selection of techniques is based upon the specific application and the guidance in AICHE (1992) and NUREG-1513.

For each credible accident event sequence determined to potentially result in unacceptable consequences, the PrHA identifies the IROFS necessary to demonstrate that the performance requirements of 10 CFR 70.61 are satisfied. The PrHAs utilize dose threshold calculations to screen event sequences whose consequences are acceptable to all potential receptors. For facility workers, dose threshold calculations identified the quantity of material, known as the material at risk, which would result in dose consequences from radiation inhalation greater than the low consequence category defined in Table 5.1-1 of the LA. Sections 5.2.4 and 5.2.5 of the LA discuss the assessment of consequence.

The staff review evaluated the details of each of the individual steps provided in Sections 5.2.1 through 5.2.5 of the LA that describe the ISA methodology. Based on a systematic analysis of each plant process, the ISA performed identified a set of individual accident sequences or process upsets that could result from the hazards. The applicant's ISA methods addressed the following:

- hazard identification
- Preliminary Hazard analyses (PHA) (accident identification)
- accident sequence construction and evaluation
- consequence determination and comparability to 10 CFR 70.61
- likelihood categorization for determining compliance with 10 CFR 70.61

The staff finds that the applicant’s ISA method, method selection, and description of the specific methods that may be used, as described in the LA, are acceptable and are consistent with the provisions of SRP Section 5.4.3.2 and the requirements of 10 CFR 70.62(c)(i–vi) to conduct and maintain an ISA for analyzing facility hazards.

5.3.2.1 Integrated Safety Analysis Change Control

The staff reviewed the LA to determine that the applicant established and committed to maintaining the organization and procedures for a formal system to manage changes to the ISA. The staff reviewed the applicant's commitments in the LA regarding change control for the ISA. Section 5.1.4 of the LA addresses control of facility changes that are associated with the ISA. The applicant will use the MFFF configuration management processes to maintain the ISA, ISA Summary, and LA to ensure that the information supporting the ISA is accurate and up to date. The applicant will evaluate changes to the facility and its processes for their impact on the ISA and LA and update the LA and ISA Summary, as needed, to ensure their continued accuracy and timeliness. The evaluation of the facility and process changes includes identification and assessment of the impact of changes to parameters used in the postulated accident sequences of the ISA. As described in Chapter 4 of the LA, the manager of the support services function is responsible for maintaining and updating the ISA, ISA Summary, and the LA.

The applicant will address safety-significant vulnerabilities or unacceptable performance deficiencies in the evaluation of the proposed facility and process changes. The applicant will take prompt and appropriate actions to address identified vulnerabilities. The applicant will control facility and process changes in accordance with the following requirements:

- A change to the facility or its processes is evaluated, as described above, before the change is implemented. The evaluation of the change determines, before the change is implemented, whether an application for an amendment to the license is required to be submitted in accordance with 10 CFR 70.34, "Amendment of Licenses."

- Both the LA and the ISA Summary describe the sites, structures, processes, systems, equipment, components, computer programs, and activities of personnel. Under 10 CFR 70.72, the applicant may make changes to these items, as described in the ISA Summary, without prior NRC approval, if the following is true:

  – The change does not create new types of accident sequences that, unless mitigated or prevented, could exceed the performance requirements of 10 CFR 70.61, and that have not previously been described in the ISA Summary.

  – The change does not use new processes, technologies, or control systems for which the applicant has no prior experience.

  – The change does not remove, without at least an equivalent replacement of the safety function, an IROFS that is listed in the ISA Summary.

  – The change does not alter an IROFS listed in the ISA Summary that is the sole item preventing or mitigating an accident sequence that exceeds the performance requirements of 10 CFR 70.61.

  – The change is not otherwise prohibited by 10 CFR 70.72, license condition, or order.

If a change allowed under 10 CFR 70.72 is made, the applicant has committed to have the affected onsite documentation updated promptly, as required by its written procedures. The applicant committed to maintain records of changes to its facility consistent with the requirements of 10 CFR 70.72. These records include a written evaluation that provides the bases for the determination that the changes do not require prior NRC approval under 10 CFR 70.72(a) and 10 CFR 70.72(b). The applicant committed to ensure that these records are maintained until termination of the license.

The applicant committed to have changes communicated to the NRC as follows:

- For changes that require NRC preapproval under 10 CFR 70.72, MOX Services submits an amendment request to the NRC, in accordance with 10 CFR 70.34 and 10 CFR 70.65.

- For changes that do not require NRC preapproval under 10 CFR 70.72, MOX Services submits to the NRC annually, within 30 days after the end of the calendar year during which the changes occurred, a brief summary of the changes to the records required by to be maintained by 10 CFR 70.62(a)(2).

- For changes that affect the ISA Summary, MOX Services submits to the NRC annually, within 30 days after the end of the calendar year during which the changes occurred, revised ISA Summary pages.

The staff has reviewed the commitments and processes for facility change control associated with the ISA and ISA Summary. The applicant has provided the criteria for NRC preapproval of changes consistent with current regulations and has committed to provide a description of the changes on an annual basis, consistent with the requirements in 10 CFR 70.72(c) and 10 CFR 70.72(d). The applicant has committed to promptly update onsite documentation, as required by 10 CFR 70.72(e). The applicant has also committed to maintain records of changes and will maintain these records until termination of the license, as required by 10 CFR 70.72(f). The staff finds that the information provided by the applicant in the LA, including the description of the processes and commitments relating to ISA change control, is acceptable and is consistent with the requirements of 10 CFR 70.72 and the guidance in SRP Section 5.4.3.2.A.ii.

5.3.3 Integrated Safety Analysis and Integrated Safety Analysis Summary

The purpose of the staff's review of the ISA results, primarily as described in the ISA Summary, is to establish reasonable assurance that the applicant performed a comprehensive ISA of the MFFF and its processes using effective systematic methods and competent staff. The staff review verified the applicant's commitment to identify and evaluate all hazards and credible accident sequences in the ISA, which involve process deviations or other events internal to the facility, and credible external events, which could result in consequences to the public, workers, or the environment of the types specified in 10 CFR 70.61. The review also determined that the applicant designated engineered and administrative IROFS and evaluated the set of items for each accident sequence which provides reasonable assurance, through preventive or mitigative measures, that the safety performance requirements of 10 CFR 70.61 are met. This determination, as provided in the SER, reflects a composite of all of the staff's technical evaluations of the results included in the ISA Summary and discussed in this SER in the various technical chapters. This chapter will also discuss the overall conclusions regarding the contents of the ISA Summary and the associated regulatory requirements of 10 CFR 70.65(1–9).

5.3.3.1 Integrated Safety Analysis Results and Summary

The technical staff reviewed the ISA results to determine with reasonable assurance that the applicant had performed a systematic evaluation of the hazards and credible accident sequences and had identified IROFS and management measures that satisfy the performance requirements of 10 CFR 70.61. To be acceptable, the information in the ISA Summary must correspond to the applicant's ISA methods, consequence, and likelihood definitions, which are also described in the submittal. The applicant's information must also show the basis and the results of applying these methods to each process. In addition, the applicant's information must show that the methods have been properly applied in each case.

The technical review performed by the staff included evaluation of those accidents that resulted in a release of radioactive material, a nuclear criticality event, or any other exposure to radiation resulting from use of licensed material. In addition, the technical staff reviewed accidents involving hazardous chemicals produced from licensed materials; that is, chemicals that are licensed materials, or have licensed materials as precursor compounds, or substances that physically or chemically interact with licensed materials and that are toxic, explosive, flammable, corrosive, or reactive to the extent that they endanger life or health. These include substances that are commingled with licensed material or are produced by a reaction with licensed material.

The staff's review of the ISA results provided in the ISA Summary includes the evaluation of whether the contents of the applicant's ISA Summary meet the content requirements of 10 CFR 70.65(b) and are consistent with the discussion in SRP Sections 5.4.3.2.B(i–xii) for ISA results and summary. In particular, these contents are described below, with the corresponding SRP section noted in the heading.

<u>Site Description (SRP Section 5.4.3.2.B.i)</u>

As required by 10 CFR 70.65(b)(1), the site description should emphasize those factors that could affect safety, such as geography, meteorology (e.g., high winds and flood potential), seismology, demography, and nearby industrial facilities and transportation routes. The applicant provided a general site description in Chapter 2 of the ISA Summary. That section provided an overall description of the MFFF site and its environment, including regional and local geography, demography, meteorology, hydrology, geology, seismology, and stability of subsurface materials. The section also described public roads and transportation, nearby bodies of water, and other geographic features. Chapter 1.3 of this SER provides a detailed review of the site. Section 2.2 of the ISA Summary provides sufficient population information, based on recent census data, showing population distribution as a function of distance from the facility, to permit evaluation of compliance with regulatory requirements, including evaluation of the public consequences. Table 2.2-1 of the ISA Summary identifies the cities and towns within 50 miles of the site and their populations. Section 1.3.1 of this SER provides the staffs analyses of the site description.

The staff concludes that the requirements for a general site description, as required by 10 CFR 70.65(b)(1) and discussed in Section 5.4.3.2.B.i of the SRP, has been met because the applicant provided sufficient site detail in the ISA Summary to evaluate the factors affecting safety.

<u>Facility Description (SRP Section 5.4.3.2.B.ii)</u>

The facility description, as required by 10 CFR 70.65(b)(2), should emphasize the facility features that could affect potential accidents and their consequences. These features are facility location, facility design information, and the location and arrangement of buildings on the facility site. Chapter 3 of the ISA Summary describes the facility. That chapter provides the facility civil/structure design, including the structural systems' design description and requirements. The ISA Summary includes the seismic qualification of SSCs, as well as the qualification of process equipment structural components. Section 3.1.2 of the ISA Summary provides the location of the buildings, and Section 3.1.3 describes the individual buildings. Section 11.11 of this SER includes a detailed review of the facility description provided in the ISA Summary.

The staff therefore concludes that the requirement for a facility description, as detailed by 10 CFR 70.65(b)(2) and discussed in SRP Section 5.4.3.2.B.ii, has been met because the applicant provide sufficient detail in the ISA Summary to evaluate the factors that could affect safety regarding the facility. The detailed review of facility description can be found in Section 11.11 of this SER.

Processes (SRP Section 5.4.3.2.B.iii)

The description in the ISA Summary of each process analyzed as part of the ISA is required by 10 CFR 70.65(b)(3). Specific areas reviewed include basic process function and theory, functions of major components and their operation, process design and equipment, and process operating ranges and limits. The applicant provided the process and system descriptions in Chapter 4 of the ISA Summary. This chapter also included design, operational, and other PSI to support hazard and accident analyses. The function, description, control concepts, interfaces, and applicable codes and standards were provided for each system. Sections 11.1 and 11.2 of this SER include a detailed review of the processes at the MFFF.

The staff therefore concludes that the applicant has met the requirement to provide a description in the ISA Summary of each process sufficient to understand the theory of the process and the hazards identified for each process, as required by 10 CFR 70.65(b)(3) and discussed in SRP Section 5.4.3.2.B.iii. The applicant has provided sufficient detail to understand the process and evaluate the factors affecting process safety. The detailed review of facility processes can be found in Section 11.1 and 11.2 of this SER.

Compliance Information (SRP Section 5.4.3.2.B.viii)

As required by 10 CFR 70.65(b)(4), the applicant must provide information that demonstrates compliance with the performance requirements of 10 CFR 70.61. In addition, 10 CFR 70.65(b) requires the applicant to describe in the ISA Summary how it will demonstrate that the 10 CFR 70.61 performance requirements are met. Since the requirements of 10 CFR 70.61 are expressed in terms of consequences and likelihoods of events, the information should show that all events meet the applicant's definitions of consequences and likelihood. The information provided is acceptable if it includes consequence and likelihood information for each accident showing that high consequence events are highly unlikely and intermediate consequence events are unlikely. The performance review criteria for 10 CFR 70.61 have three elements: (1) completeness, (2) consequences, and (3) likelihood. Completeness, as discussed in the SRP, refers to the fact that the applicant must address each credible event. The staff evaluates the completeness of the ISA Summary contents in the individual technical sections of this SER. Consequences refers to the magnitude of the chemical and radiological doses used by the applicant to categorize accidents as being of high or intermediate consequence. Likelihood, as discussed in the SRP, refers to the fact that 10 CFR 70.61 requires the applicant to demonstrate that intermediate consequence events will be unlikely and high consequence events will be highly unlikely. The staff also evaluates the consequence and likelihood of each accident sequence in the individual technical sections based on the information provided by the applicant in Chapter 5 of the ISA Summary. Section 5.3.2 of the LA provides the accident analysis results. These results were separated by event type to include the following:

- loss of confinement
- fire
- load handling
- explosion
- criticality
- natural phenomenon
- external manmade
- external radiation exposure
- chemical release

The ISA Summary discusses each of these individual event types, including event descriptions, the IROFS associated with the event, a risk discussion which demonstrates that the performance requirements of 10 CFR 70.61 are met, and a discussion of defense in depth. For each event type, the ISA Summary provides tables showing the consequences for each accident event; a list of both engineered and administrative IROFS, including the safety function of each IROFS; and a summary of the event. Table 5.3-1 identifies where in this SER a discussion of individual ISA Summary events can be found.

Table 5.3-1 Crosswalk to Discussions of ISA Summary Events

ISAS Event Number	Event description	SER discussion location
F-01	Fire in AP Process Cells	7.3.6.1
F-02, F-04	Fire Involving Gloveboxes in the AP/MP C3 Glovebox Areas	7.3.6.2
F-03	Fire in the AP Gloveboxes with Vessels Containing Solvents	7.3.6.3
F-05	Fire in Titanium/Electrolyzer in the AP/MP C3 Glovebox Area	11.2.1.3.4, 7.3.6.4
F-06	Fire in 3013 Canister in C1 and/or C2 Areas	7.3.6.5
F-07	Fire Event in the PuO_2 Truck Bay Receiving Area	7.3.6.6
F-08	Fire Involving Fuel Rods/Fuel Assembly	7.3.6.7
F-09	Fire in MOX Fuel Transport Cask in C1 and/or C2 Areas	7.3.6.8
F-10	Fires in Waste Container in C1, C2, or C3 Areas	7.3.6.9
F-11	Fire in Transfer Container in C2 Area	7.3.6.10
F-12	Fire Involving KWG, POE, and VHD Final HEPA Filters	7.3.6.11
F-13	Fire outside the BMF Propagating to the inside	7.3.6.12
F-14	Fire in Facilitywide Systems	7.3.6.13
F-15	Fire in Facility Propagating from One Fire Area to Another Fire Area	7.3.6.14
F-16	Fire in Secured Warehouse Building	7.3.6.15
F-17	Fire Involving Vessels Containing Solvents in C2 Confinement Areas	7.3.6.16
F-18	Fire in UO_2 Intermediate Storage Room in C2 Area	7.3.6.17
F-19	Fire Event Involving Zircaloy Swarf	7.3.6.18
F-20	Fire Event Potentially Producing Excessive Soot in a Fire Area Ventilated by HDE	7.3.6.19

F-21	Fire Event in the Hydraulic Pump Room Ventilated by HDE	7.3.6.20
LOC-1	Overtemperature	11.3.3.4.1
LOC-2	Small Breach in Glovebox Boundary/Backflow from Glovebox to Utility Line	11.3.3.4.2
LOC-3	Leaks from AP Vessels or Pipes within Process Cells	11.7.5.1
LOC-4	Leaks from AP Vessels or Pipes within Gloveboxes	11.7.5.2
LOC-5	Backflow from Process Vessels through Utility Lines	11.7.5.3
LOC-6	Rod-Handling Operations	11.6.5.15
LOC-7	Breaches in Containers outside a Glovebox (see LH-5, LH-10)	11.6.5.5, 11.6.5.10
LOC-8	Over- or under-pressurization of a Glovebox	11.3.3.4.3
LOC-9	Excessive Temperature due to Decay Heat from Radioactive Materials	11.3.3.4.4
LOC-10	Glovebox Dynamic Exhaust Failure	11.3.3.4.5
LOC-11	Process Fluid Line Leak in a C3 Area outside a Glovebox	11.7.5.4
LOC-12	Sintering Furnace Confinement Boundary Failure	11.3.3.4.6
LOC-13	Uncontrolled Release of Nitrogen Tetroxide	11.7.5.5
LH-01	Process Vessel Breaches as a Result of Maintenance Activities in AP Process Cell	11.6.5.1
LH-02	Load-Handling Events during Normal Operations within the Confinement Capabilities of the Gloveboxes	11.6.5.2
LH-03	Powder Jar Falls and Impacts onto the Glovebox	11.6.5.3
LH-04	Maintenance Operations Cause a Glovebox Breach	11.6.5.4
LH-05	3013 Container Handling Events outside of Glovebox	11.6.5.5
LH-06	Handling Shipping Package for the 3013 Container	11.6.5.6
LH-07	Handling of Fuel Assemblies	11.6.5.7
LH-08	Handling of MOX Fuel Transport Cask	11.6.5.8
LH-09	Handling of Waste Container	11.6.5.9
LH-10	Handling of Transfer Container	11.6.5.10
LH-11	Load Impacts to Final VHD HEPA Filter	11.6.5.11
LH-12	Consolidated with LOC-02	11.3.3.4.2
LH-13	Breaching of Waste Transfer Line outside MFFF Building	11.6.5.12
LH-14	Heavy Loads or Load-Handling Equipment Damaging Principal Structures or Primary Confinement Boundaries of the MFFF Building	11.6.5.13
LH-15	Load-Handling of Depleted UO_2 Container	11.6.5.14
LH-16	Rod-Handling Operations	11.6.5.15
EXP01	Hydrogen Explosion	8.1.2.4.1.1
EXP02	Steam Overpressurization Explosion	11.1.1.2.4
EXP03	Radiolysis Explosion	8.1.2.4.1.2
EXP04	HAN Explosion	8.1.2.4.3.1
EXP05	Hydrogen Peroxide Explosion	8.1.2.4.4
EXP06	Solvent Explosion	8.1.2.4.2
EXP07	TBP-Nitrate (Red Oils) Explosion	8.1.2.4.5
EXP08	AP Vessel Overpressurization Explosion	11.7.6.1
EXP09	Pressure Vessel Overpressurization Explosion	11.7.6.2
EXP10	Hydrazoic Acid Explosion	8.1.2.4.3.2
EXP11	Metal Azide Explosion	8.1.2.4.3.3

EXP12	Pu(VI) Oxalate Explosion	8.1.2.4.6
EXP13	Electrolysis Related Explosion	8.1.2.4.1.3
EXP14	Laboratory Explosion	11.10.2.1
EXP15	Outside Explosion (outside the BMF, but on the MFFF site)	11.11.8.1
EXP16	Miscellaneous Explosions	11.11.8.2
EXP17	Perchlorate Explosions	8.1.2.4.1.3
NPH-01	Earthquake Affecting the BMF, BEG/UEF, KWD, Fluid Transport System, and Hazardous Material Release	11.11.6.1
NPH-02	Tornado at the BMF, BEG/UEF, KWD, Tornado-Driven Missiles, and a Wind and Atmospheric Pressure Change of 150 psf At a Rate of 55 psf/s	11.11.6.2
NPH-03	Severe Winds Affecting the BMF, BEG/UEF, Waste Transfer Lines, Extreme Winds, and Wind-Driven Missiles	11.11.6.3
NPH-04	External Fire Starting from NPH (see F-13)	7.1.6.12
NPH-05	Rain, Snow, Ice Affecting the BMF, BEG/UEF, and Waste Transfer Lines	11.11.6.4
EMMH-2	External Explosions	11.11.7.1
EMMH-3	Loss of Offsite Power	11.4.5
EMMH-4	External Fire (see F-13)	7.1.6.12
CRE-1	Events Involving a Release of Hazardous Chemicals Not Subject to 10 CFR Part 70	8.1.2.3.2.1
CRE-2	Events Involving a Release of Licensed Material or Hazardous Material Produced from Licensed Material (see NPH-01, NPH-02, F-16, LH-15, LOC-3, LOC-4, LOC-11, LOC-13, and explosion events)	8.1.2.3.2.2
Criticality	Events Related to Criticality	6

The staff therefore finds that, from a content perspective, the applicant has met the requirement in 10 CFR 70.65(b)(4) because MOX Services has demonstrated in the ISA Summary that it has met the 10 CFR 70.61 performance requirements for all accident sequences listed in the ISA Summary. The staff’s technical evaluations of the individual types of events and its review to confirm that the applicant’s demonstrations comply with the performance requirements and that the ISA methodology has been acceptably implemented is provided in each of the individual technical sections of this SER.

<u>ISA Team Qualifications (SRP Section 5.4.3.2.B.iv)</u>

The ISA Summary describes the applicant’s ISA team qualifications required by 10 CFR 70.65(b)(5)). The discussion of the ISA team makeup and qualifications in Section 5.2 of the ISA Summary includes the following details:

- The ISA team has a team leader who is formally trained and knowledgeable in the ISA methodology chosen for the hazard and accident evaluations. In addition, the team leader should have an adequate understanding of all process operations and hazards under evaluation, but should not be the cognizant engineer or expert for that process.

- At least one member of the ISA team has thorough, specific, and detailed experience in the process under evaluation.

- The team represents a variety of process design and safety experiences in those particular safety disciplines relevant to hazards that could credibly be present in the process, including, if applicable, radiation safety, nuclear criticality safety, fire protection, and chemical safety disciplines.

- A manager provides overall administrative and technical direction for the ISA.

  The staff review finds that the descriptions provided by the applicant regarding the ISA team requirements and qualifications are sufficient to meet the 10 CFR 70.65(b)(5) requirement and the guidance contained in SRP Section 5.4.3.2.B.iv regarding a description of the team and the team qualifications.

<u>ISA Methods (SRP Section 5.4.3.2.B.v)</u>

The ISA Summary describes the applicant's ISA methods, as required by 10 CFR 70.65(b)(5). The applicant provided a description of the ISA process in Section 5.1 of the ISA Summary. Chapter 5 of the LA also provided commitments regarding the ISA methods. To ensure that all event sequences with consequences exceeding the low consequence threshold of 10 CFR 70.61 meet the performance requirements identified in 10 CFR 70.61, the following qualitative design criteria and commitments are applied to those events and the associated IROFS:

- the single-failure criterion or double contingency principle (for nuclear criticality)

- the requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and NQA-1

- industry codes and standards

- management measures, including surveillance of IROFS (i.e., failure detection and repair or process shutdown capability)

The first design criterion, application of the single-failure criterion or double-contingency principle, is the most important attribute in providing adequate risk reduction for event sequences and consequently ensuring that each respective event sequence is ultimately rendered highly unlikely. This design criterion ensures that even in the unlikely event of a failure of a single contingency, another unlikely, independent, and concurrent failure or process change is required before the occurrence of the event. This design criterion ensures that means are provided to protect against an event that could exceed the requirements of 10 CFR 70.61, including an inadvertent nuclear criticality.

The single-failure criterion for MFFF requires that IROFS be capable of carrying out their functions given the failure of any single active component within the system or in an associated system that supports its operation. Multiple failures resulting from a single occurrence are considered to be a single failure (also referred to as a common-mode or common-cause failure). Application of the single-failure criterion is not required for IROFS performing a passive safety function (e.g., a glovebox providing confinement). The following hierarchy of controls has been established by the applicant regarding the application of IROFS with respect to the single failure criterion:

- protection by a single passive safety device, functionally tested on a predetermined basis

- protection by independent and redundant active-engineered features, functionally tested on a predetermined basis

- protection by a single hardware system or engineered feature, functionally tested on a predetermined basis

- protection by enhanced administrative controls

- protection by simple administrative controls

To ensure adequate implementation of the single failure criterion, the following principles are applied by the applicant to the design of IROFS:

- Redundant equipment or systems—A piece of equipment or a system is redundant if it duplicates the operation of another piece of equipment or system to the extent that either may perform the required function (either identically or similarly), regardless of the state of operation or failure of the other.

- Diversity—Equipment or systems may satisfy the single-failure criterion by providing diverse means of performing an IROFS safety function. This diverse means of performing the safety function is equipment that does not duplicate the operation of another piece of equipment (redundancy), but still achieves the reliability required for the safety function. Each diverse system (e.g., means, paths, trains) or component is not required to provide for additional redundancy.

- Independence—IROFS are designed to ensure that the effects of natural phenomena and of normal operating, maintenance, testing, and postulated accident conditions on redundant equipment or systems do not result in the loss of their safety function or are demonstrated to be acceptable on some other defined basis.

- Separation—IROFS are separated to the extent that failure of a single system component, or failure or removal from service of any IROFS that is common to the other systems and the IROFS, leaves intact an IROFS satisfying applicable reliability, redundancy, and independence requirements.

- Fail safe—IROFS are designed to fail into a safe state or into some other nonthreatening defined basis if conditions such as disconnection of a system, loss of energy, or loss of pressure occur.

In cases in which a single active system, component, or activity of personnel is the sole IROFS preventing or mitigating an accident sequence that exceeds the performance requirements of 10 CFR 70.61, a descriptive list must be provided to demonstrate that the IROFS is designed to perform its safety function in accordance with 10 CFR 70.65(b)(8). This may include a discussion of additional management measures (e.g., increased surveillance frequencies), fail-safe characteristics, highly reliable components, or the application of noncredited additional protection features. Passive structures and components (such as buildings or tanks) are not designated as sole IROFS in the event that their design, or the design of any associated IROFS passive structure or component, precludes their failure under all credible natural phenomena and process conditions. However, these components, if relied upon, are designated by the applicant as IROFS and will be constructed as quality level items.

The second design criterion, application of the MPQAP, ensures that the requirements for IROFS are correctly translated into specifications, drawings, procedures, and instructions. The highest level of quality assurance (QL-1) and quality control are applied to all IROFS. This ensures a comprehensive application of quality assurance requirements covering all phases of the project, including the design process, configuration control; records management; procurement; materials control; installation; use of measurement and test equipment; and computer software and hardware.

Within the MPQAP, quality assurance grading can also be used to identify the controls applied to IROFS and activities that support the MPQAP based upon an evaluation of the complexity and importance of the activity compared to quality, safety, risk, and the environment. Quality levels can be used to establish the level of programmatic requirements and procedural controls which are applied to SSCs and associated activities. The rigor of quality assurance controls is commensurate with, but not limited to, the following criteria:

- the function or end use of the safety controls
- the importance and end use of the data collected or analyzed
- the consequence and likelihood of failure
- the complexity or uniqueness of the design, fabrication, or implementation
- the reproducibility of the results
- the reliability of the process
- the necessity for special controls or processes
- the ability to demonstrate functional compliance with applicable regulations

The extent of quality assurance controls applied to a safety control or activity varies as a function of the degree of confidence needed to achieve the desired quality. The grading process provides the flexibility to design and implement controls that best suit the facility or activity, but is not intended to reduce or in any way degrade compliance with applicable requirements.

The third design criterion, application of recognized industry codes and standards, provides confidence in the ability of IROFS to perform their functions. The codes and standards provide the foundation for ensuring that IROFS are robust and incorporate lessons learned from the nuclear, mechanical, electrical, and instrumentation and control disciplines. They provide an effective set of engineering and procedural guidelines used to design, construct, and operate the IROFS. Application of codes and standards provides assurance that controls utilized to implement the single-failure criterion or double-contingency principle are sufficiently reliable.

The fourth design criterion, application of management measures, is particularly important in the context of IROFS failure detection. IROFS failure detection is meant to include detection of IROFS failures and repair of the IROFS or the process is shutdown. As described in the SRP, IROFS failure detection can significantly reduce the likelihood of an accident scenario. For an accident scenario to proceed to completion, failure of one IROFS must occur, its failure must go undetected, and a second IROFS must fail.

Management measures are applied to the identified IROFS to ensure that they are reliable and available on demand. The MPQAP specifically describes the quality assurance requirements, implementing procedural controls, and documentation requirements to address management measures as described in the SRP. The set of applied management measures consists of applicable elements of the following management measures programs: quality assurance, configuration management, maintenance, training and qualification of plant personnel, plant procedures, audits and assessments, incident investigations, and records management.

Management measures are assigned based on the following types of IROFS classifications and the risk reduction level attributed to that particular IROFS as stated in Chapter 5 of the LA:

- Passive Engineered Controls—A device that uses only fixed physical design features to maintain safe process conditions without any required human action.

- Active Engineered Controls—A physical device that uses active sensors, electrical components, or moving parts to maintain safe process conditions without any required human action.

- Enhanced Administrative Controls—A procedurally required or prohibited human action, combined with a physical device that alerts the operator that the action is needed to maintain safe process conditions, or otherwise adds substantial assurance of the required human performance (i.e., augmented administrative control).

- Administrative Controls—A procedural human action that is prohibited or required to maintain safe process conditions (i.e., a simple administrative control).

Chapter 15 of the LA provides the specific elements of the various management measure programs assigned to each IROFS classification. The MPQAP illustrates how management measures are applied to the above IROFS classifications. For the enhanced administrative controls, the specific management measures for the physical device are covered under the active engineered controls classification.

Effective application of these well-defined qualitative criteria will ensure that event sequences are highly unlikely. The application of the single-failure criterion or double-contingency principle and IROFS failure detection ensure that multiple undetected failures are required for an accident sequence to proceed to conclusion. Application of appropriate codes and standards and an NQA-1 quality assurance program ensure that IROFS will be designed, operated, and maintained in a reliable manner. The application of these qualitative design criteria ensures that adequate risk reduction is achieved to satisfy the requirements of 10 CFR 70.61.

In addition to the four qualitative criteria discussed in Section 5.2.6.2, the following IROFS characteristics and qualities are defined and documented by the applicant to ensure the reliability and availability of the IROFS:

- Safety function—The credited safety function of each IROFS is stated in the safety evaluation in the ISA with a description of the controlled safety parameter.

- Quality classification—IROFS are classified to the highest level of quality

- Operating range and limits—The functional range of the IROFS is ensured to encompass both the normal operating range and the safety limit with an acceptable sensitivity over this full range.

- Emergency capabilities—Operational requirements for an IROFS under emergency conditions (e.g., loss of power) is identified and demonstrated to be implemented in the design.

- Testing and maintenance requirements—Testing and maintenance requirements are specified for each IROFS, including a description of the means to detect failures, if available, and the applied management measures.

- Environmental design factors—Environmental design characteristics necessary to ensure that the IROFS remains available and reliable to perform its safety function are identified for each IROFS. These characteristics account for both short-term and long-term exposures to environmental conditions potentially detrimental to the operation of an IROFS (such as long-term chemical degradation impacts or short-term temperature transient impacts).

- Natural phenomena response—Operational requirements of an IROFS during and after NPHs (e.g., earthquakes) are specified.

- Required instrumentation—Instrumentation necessary to ensure an IROFS operation is specified.

- Applicable codes and standards—The design codes and standards applied to an IROFS (e.g., Institute of Electrical and Electronics Engineers, American Society of Mechanical Engineers, American Nuclear Society) are identified.

- Reliability—IROFS are procured under a 10 CFR Part 50, Appendix B, NQA-1 quality assurance program.

- Protection from fires and explosions—Fires and explosions are specifically addressed in separate safety evaluations.

The following system-level parameters are also considered in the safety evaluations and the PSI:

- Safety margin, a comparison of the process parameter under normal conditions with the parameter's safety limit, is described.

- The type of control—passive, active, enhanced administrative control, or administrative control—is noted.

- Management measures are discussed.

- Fail-safe position, self-announcing fault, or surveillance measures to limit down time are identified.
- Failure modes, if credited, are described.

- Demand rate, where specifically credited, is noted.
- IROFS failure rate is ensured by the implementation of a 10 CFR Part 50, Appendix B, NQA-1 quality assurance program and commitments to industry codes and standards and provides confidence that IROFS are at least unlikely to fail.

In addition to the individual qualities of each IROFS listed above, other reliability and availability qualities are related to the characteristics of the whole system of IROFS utilized to protect against an accident sequence. The following information is also addressed in the ISA safety evaluations and the PSI:

- Defense-in-depth features are described. These features may include normal process controls that are nearly identical to IROFS controls, but with lower setpoints, that reduce the potential demands on the IROFS.

- Degree of redundancy is identified. Usually, the degree of redundancy is dual, although diverse independent controls are sometimes used.

- Degree of independence is specified, usually by the use of two independent controls.

- Diversity is described, when applicable. Often diversity is not practical, in which case independent controls are provided.

- Vulnerability to common-cause failure is assessed and limited by having independent or diverse controls.

The MFFF Operating Limits Manual (OLM) defines IROFS operability. A system, subsystem, component, or device is operable or has operability when it is capable of performing its specified function and when all necessary support equipment required for the system, subsystem, component, or device to perform its specified IROFS function is also capable of performing its related support functions. The MFFF OLM defines operational modes, operability requirements, limiting conditions for operation (LCOs) and associated completion times, and required surveillances and frequencies. The LCO for MFFF IROFS components or systems is defined as the lowest functional capability or performance level of the SSCs required for safe operation. The MFFF OLM also defines the use of compensatory measures in coordination with the LCOs for an IROFS. Compliance with the LCOs ensures that the performance requirements of 10 CFR 70.61 are met.

The reliability and availability qualities of IROFS are assessed in the ISA safety evaluations and included in the PSI. These safety evaluations ensure that the IROFS are sufficient and capable of performing their safety functions, as described in the safety evaluations, with sufficient reliability and availability to ensure that each IROFS is at least unlikely to fail, thereby ensuring that the performance criteria of 10 CFR 70.61 are satisfied.

The staff review of the descriptions of the methods used in the ISA finds that the applicant provided sufficient detail regarding the implementation of the methods to understand the methods applied, as well as clear documentation of the process, so that there is reasonable assurance that the method can be implemented in the future. The staff therefore finds that the applicant has met the requirements of 10 CFR 70.65(b)(5) and the guidance in SRP Section 5.4.3.2.B.v.

IROFS Description (SRP Section 5.4.3.2.B.xi)

As required by 10 CFR 70.65(b)(6), the applicant must provide a brief list describing each IROFS. The ISA Summary provided detailed tables of engineered and administrative IROFS for each accident event analyzed by the applicant that required the application of IROFS. These tables, provided in Chapter 5 of the ISA Summary, contain all the events for which each IROFS was associated with, the safety function of the IROFS, and the applicable process units in which the IROFS would be implemented. The staff therefore finds that the applicant has met the requirements of 10 CFR 70.65(b)(6) and the guidance in SRP Section 5.4.3.2.B.xi by providing a list describing each IROFS in sufficient detail to understand its function in relationship to the performance requirements.

Description of the Proposed Quantitative Standards (SRP Section 5.4.3.2.B.vi)

As required by 10 CFR 70.65(b)(7), the applicant must describe the quantitative standards used to assess the consequences to an individual from acute chemical exposure to licensed material or chemicals produced from licensed materials which are on-site, or expected to be on-site as described in 10 CFR 70.61(b)(4) and (c)(4). The applicant included the standards for chemical consequence levels in Chapter 5 of the ISA Summary. Chapter 8 of this SER provides a detailed evaluation of the acceptability of those levels. In terms of meeting the requirement to include the data in the ISA Summary, as required by 10 CFR 70.65(b)(7) and discussed in SRP Section 5.4.3.2.B.vi, the staff finds that the requirement has been acceptably met.

Sole IROFS (SRP Section 5.4.3.2.B.xii)
The list of all IROFS that are sole IROFS is required by 10 CFR 70.65(b)(8). The applicant provided tables of sole IROFS in Section 5.3 of the ISA Summary for the event types that credit sole IROFS. The staff therefore finds that the applicant has met the requirements of 10 CFR 70.65(b)(8) and the guidance in SRP Section 5.4.3.2.B.xii by providing lists of sole IROFS for the events that depend on them.

Definitions of Likelihood (SRP Section 5.4.3.2.B.vii)

The applicant has described its definitions of unlikely, highly unlikely, and credible, as required by 10 CFR 70.65(b)(9). Section 5.1.2.5 of the ISA Summary provides the applicant's definitions of unlikely, highly unlikely, and credible. The applicant originally proposed these definitions in the CA licensing phase and they were accepted by the staff in the SER for the CAR. The applicant used the following definitions:

- Not Unlikely—Events that may occur during the lifetime of the facility.
- Unlikely—Events that are not expected to occur during the lifetime of the facility or events originally classified as not unlikely to which sufficient IROFS are applied to further reduce their likelihood to an acceptable level.
- Highly Unlikely—Events originally classified as not unlikely or unlikely to which sufficient IROFS are applied to further reduce their likelihood to an acceptable level.
- Credible—Events that do not meet the definition of not credible.

- Not Credible—The definition includes one of the following:
  - Natural phenomena or external manmade events with an extremely low initiating event frequency, conservatively estimated as less than once in a million years.
  - A process deviation that consists of a sequence of many unlikely human actions or errors for which there is no reason or motive and no such sequence of events can ever have actually happened in any fuel cycle facility.
  - Process upsets for which there is a convincing argument, based on physical laws, that are not possible or are unquestionably extremely unlikely.

The applicant described these likelihood definitions in a manner such that the application of these definitions will ensure that the performance requirements of 10 CFR 70.61 are satisfied. These definitions and methodology rely on specific identifiable characteristics of the process design that may affect the likelihood of an accident sequence, rather than subjective judgments of adequacy. In applying the above definitions to address the performance requirements of 10 CFR 70.61, initiating events are assumed to be not unlikely. Postulated credible intermediate or high consequence events are made highly unlikely based on the application of IROFS features or controls without crediting the likelihood of the initiating event.

The staff review finds that the definitions provided in the ISA Summary are consistent with the definitions previously approved by the staff and meet the requirements of 10 CFR 70.65(b)(9) and the guidance in SRP Section 5.4.3.2.B.vii by inclusion in the ISA Summary.

5.3.4 Baseline Design Criteria

The information demonstrating compliance with the baseline design criteria, required by 10 CFR 70.64(a)(1–5) and 10 CFR 70.64(a)(7–10) for new facilities and by 10 CFR 70.65(b)(4), is found in the individual technical sections of the ISA Summary. The technical sections of this SER provide the detailed review by the staff of the 10 CFR 70.64 requirements. The applicant has committed to evaluate baseline design criteria as described in the ISA methodology provided in Chapter 5 of the LA and Chapter 5 of the ISA Summary. The staff review therefore finds that the applicant’s commitment to apply baseline design criteria in performing an ISA meets the requirement in 10 CFR 70.65(b)(4) and is therefore acceptable.

5.4 Evaluation Findings

The staff performed a review of the ISA programmatic commitments, as described in the LA, and for the ISA results, as described in the ISA Summary. The review was coordinated with other technical staff reviewers to ensure consistency between the review conducted in this chapter and the review conducted in other chapters. The review was also coordinated with the staff reviewing quality assurance and management measures to ensure that the review of the MPQAP and the applicant’s proposed management practices are consistent with the material submitted in Chapter 15 of the LA.

The staff conducted its review of the ISA programmatic commitments in the LA and ISA Summary for the MFFF to possess and use special nuclear material in accordance with the guidance provided in Chapter 5.0 of the SRP. The staff confirmed that the applicant’s LA contains appropriate commitments, including commitments to (1) perform and maintain an ISA,

(2) compile and maintain PSI, (3) engage personnel with appropriate training to conduct the ISA, (4) use appropriate methods to conduct the ISA, and (5) implement appropriate measures and procedures to ensure that the ISA stays accurate and up to date.

The staff has also verified that the applicant performed an ISA to identify and evaluate the hazards and potential accidents associated with the facility and to establish engineered and administrative controls to ensure that facility operation will be within the bounds of the 10 CFR 70.61 performance requirements. The staff confirmed that the applicant's ISA Summary (1) identified the hazards at the facility, (2) analyzed for accident sequences through the use of process hazards analysis, (3) evaluated and assigned consequences to the accident sequences, and (4) evaluated the likelihood of each accident consistent with the guidance in the SRP. Moreover, the applicant identified all IROFS, including administrative and engineered controls, and provided a listing of these controls, including sole IROFS, in its ISA Summary. As a result, the NRC staff has concluded that there is reasonable assurance that the applicant's postulated accidents resulting from the facility hazards that may be anticipated to occur should be in compliance with the performance requirements of 10 CFR 70.61.

The staff further concludes that the identification and evaluation of the hazards and accidents as part of the ISA and establishment of controls to maintain safe facility operation from their consequences meet the requirements for a license to possess and use special nuclear material under 10 CFR Part 70 and provide reasonable assurance that the health and safety of the public, the workers, and the environment will be adequately protected.

REFERENCES

(MOX, 2010a) Shaw Areva MOX Services, "MFFF—License Application," Aiken, SC, March 2010.

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(NRC, 2000) U.S. Nuclear Regulatory Commission, NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility," Washington, DC, August 2000.

(NRC, 2005) U.S. Nuclear Regulatory Commission, NUREG-1821, "Final Safety Evaluation Report on the Construction Authorization Request for the Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina," Washington, DC, March 2005.

(NRC, 2009) U.S. Nuclear Regulatory Commission, "Safety Evaluation Report of the MPQAP," Washington DC, April 2010.

(AICHE 1992) American Institute of Chemical Engineers, Center for Chemical Process Safety, "Guidelines for Hazard Evaluation Procedures—Second Edition—With Worked Examples," New York, NY, 1992.

(NRC, 1999) U.S. Nuclear Regulatory Commission, NUREG-1513, "Integrated Safety Analysis Guidance Document," Washington, DC, 1999.

10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities.

10 CFR Part 70, Domestic Licensing of Special Nuclear Material.

6.0 NUCLEAR CRITICALITY SAFETY

6.1 License Application Review

This chapter of the safety evaluation report (SER) contains the U.S. Nuclear Regulatory Commission (NRC) staff's review of the nuclear criticality safety (NCS) program described by the applicant, Shaw AREVA MOX Services (MOX Services), in Chapter 6 of the license application (LA) (MOX, 2010a), Section 5.3.7 of the Integrated Safety Analysis (ISA) Summary (MOX, 2010b), and other supporting documents that the staff evaluated during in-office reviews at the applicant's facilities.

6.1.1 Organization and Administration

Section 6.1 of the LA discusses the nuclear criticality safety organization and administration at the Mixed Oxide Fuel Fabrication Facility (MFFF), including the technical qualifications, training, and experience of the applicant and its staff to engage in NCS activities. The applicant's organizational commitments include designating organizational functions important to criticality safety, assigning responsibility and authority to these positions to carry out NCS functions, and stating the requirements for the qualification of NCS personnel. The NCS organization is administratively independent of production responsibilities and has the authority and responsibility to shut down potentially unsafe MFFF operations. Specific responsibilities of the NCS organization and its personnel include the following:

- Establish the NCS Program, including design criteria, procedures, and training.
- Provide NCS support for ISA and configuration control.
- Assess normal and credible abnormal conditions.
- Determine criticality safety limits for controlled parameters.
- Develop and validate methods to support nuclear criticality safety evaluations (NCSEs).
- Perform criticality safety calculations and prepare NCSEs.
- Review and approve proposed changes in process conditions or equipment involving fissionable material as part of the MFFF configuration management and design change process to determine whether the facility changes require prior NRC approval in accordance with the criteria of Title 10 of the *Code of Federal Regulations* (10 CFR) 70.72, "Facility Change Process."
- Specify NCS control requirements and functionality.
- Review and approve MFFF operations and operating procedures that involve fissionable material.
- Support emergency response planning and events.

- Assess the effectiveness of the NCS Program through the audit and assessment program.

- Identify NCS posting requirements that provide administrative controls for operators in applicable work areas.

- Maintain NCS programs for the MFFF in accordance with applicable regulatory guides (RGs) and industry standards.

- Serve as the single point of contact for nuclear criticality issues with internal and external groups or agencies, coordinating with and taking direction from the manager of the regulatory function.

The manager of the environmental safety and health (ES&H) licensing function is independent of the production function and is directly responsible for the health, safety, and environmental functions, including fire safety, radiation protection, chemical safety, criticality safety, nuclear safety analysis, and environmental protection. The ES&H licensing manager reports directly to the President of MOX Services. The ES&H licensing manager is responsible for maintaining the MOX Services special nuclear material (SNM) possession and use license, planning and executing licensing and regulatory compliance activities, maintaining licensing-related documents, and interfacing with the NRC and other regulatory agencies regarding licensing matters. These functions are accomplished by delegating and assigning responsibility to qualified personnel. In addition, the manager of the ES&H licensing function has the authority to make commitments to the NRC.

The NCS organization reports to the manager of the ES&H licensing function. The NCS organization commits to implement the applicable NCS practices of American National Standards Institute/American Nuclear Society (ANSI/ANS)-8.1-1998, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors." The NCS organization also commits to implement the NCS administrative practices, as described in ANSI/ANS-8.19-2005, "Administrative Practices for Nuclear Criticality Safety."

The manager of the NCS function has the authority and responsibility to assign and direct activities associated with the NCS function. Senior NCS engineers have the authority and responsibility to conduct activities assigned to the criticality safety function, as directed by the manager of the NCS function. NCS engineers have the authority and responsibility to conduct activities assigned to the criticality safety function, with the exception of independent verification of NCSEs.

The minimum requirements for the manager of the NCS function are a baccalaureate degree, or equivalent, with a science or engineering emphasis and 3 years of nuclear industry experience in criticality safety. The manager of the NCS function has appropriate knowledge of NCS and its administration. The minimum requirements for a senior NCS engineer are a baccalaureate degree, or equivalent, with a science or engineering emphasis and 3 years of nuclear industry experience in criticality safety. Senior NCS engineers have the authority and responsibility to conduct activities assigned to the criticality safety function, as directed by the manager of the NCS function. The minimum requirements for an NCS engineer are a baccalaureate degree, or equivalent, with a science or engineering emphasis and 1 year of nuclear industry experience in criticality safety. NCS engineers have the authority and responsibility to conduct activities assigned to the criticality safety function, with the exception of independent verification of NCSEs.

The staff has reviewed the MFFF NCS organizational structure and finds it acceptable because the NCS organization is independent of the production staff, education and experience levels for NCS staff members are sufficient to provide the requisite skills and knowledge to perform their technical duties, and the NCS organization is consistent with the requirements in ANSI/ANS-8.19-2005. The staff finds that the commitments in this section are consistent with the guidance in Section 6.4.3.1 of NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility," issued August 2000, and with standard industry practice.[2] Therefore, the applicant's commitments are acceptable to the staff.

6.1.2 Management Measures

MFFF maintains several programs, systems, and functions to ensure that all items relied on for safety (IROFS) will be available and reliable, will remain available and reliable, and will be under surveillance for malfunction detection and appropriate corrective action. These management measures include training, audits and assessments, and procedures.

The applicant commits in Chapter 6 of the LA to implement the training requirements of ANSI/ANS-8.19-2005 and ANSI/ANS-8.20-1991 (R1999), "Nuclear Criticality Safety Training," at the MFFF. The training is appropriately tailored to the staff's function within the MFFF.

In addition, the MFFF NCS staff develops the following:

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

The applicant conducts program and process assessments to compare established NCS standards to MFFF performance. The assessments take the form of program audits and compliance inspections. MOX Services commits to meet the requirements of ANSI/ANS-8-19-2005, as they relate to audits and assessments.

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2 In the context of this chapter of the SER, "standard industry practices" refer to historical NCS good practices, as described in industry consensus standards (e.g., ANSI/ANS-8.1-1998 and ANSI/ANS-8.19-2005), industry-accepted handbooks and safety guides, regulatory guidance, and commitments in fuel facility licenses and internal documents. Standard industry practices encompass organization and administration of the NCS Program, management measures, and technical practices.

The NCS Program will be evaluated annually by either periodic audits or assessments. At a minimum, regularly scheduled internal audits of the NCS functional area quality-affecting activities will be performed at least once every 2 years. The frequency for audits of operational phase IROFS-related activities will be based on a risk-informed methodology determination which will consider the safety significance of the activity and the results of the ISA or performance history, or both, so that each area is evaluated annually (assessment or audit) and audited at least once every 2 years. Personnel conducting audits will be independent of the direct responsibility for performing the work being audited.

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

The applicant established and implemented NCS procedures in accordance with ANSI/ANS-8.19-2005 such that the double contingency principle is maintained. NCS posting requirements at the MFFF identify administrative controls applicable and appropriate to the activity or area. NCS procedures and postings are controlled to ensure that they are maintained current. Procedures and their implementation are reviewed periodically, but at least once every 2 years, to ascertain that procedures are being followed and that process conditions have not been altered to adversely affect NCS requirements or controls. The frequency for procedure reviews, if less than annually, will be based on a risk-informed methodology determination which will consider the safety significance of the activity, results of the ISA, and performance history. MFFF staff knowledgeable in NCS conduct the reviews, in consultation with operating personnel.

The staff has reviewed the applicant's commitments to NCS management measures and finds that they are acceptable because the applicant commits to (1) provide training to personnel, (2) conduct activities involving SNM with written and approved procedures, (3) conduct NCS walkdowns using a graded approach based on the ISA, (4) conduct NCS audits such that all processes and all aspects of the program are audited within 2 years, and (5) evaluate procedures to ensure that they consider the double contingency principle. The staff finds that the commitments in this section are consistent with the guidance in Section 6.4.3.2 of NUREG-1718 and standard industry practice. Therefore, the applicant's commitments are acceptable to the staff.

6.1.3 Technical Practices

The technical practices, as described in LA Section 6.4, describe the applicant's methodology for performing NCSEs and their supporting calculations, selecting and controlling the criticality controlled parameters (i.e., criticality control modes), and validating calculational techniques used to derive subcritical limits on those parameters. Significant aspects of these technical practices were used in designing the facility, and the NRC staff examined these practices during its review of the construction authorization review (CAR) (DCS, 2005). The following sections discuss those aspects unique to the operation of the MFFF.

6.1.3.1 Nuclear Criticality Safety Evaluations and Analytical Methodology

LA Section 6.4.1 discusses the criteria for determining when an NCSE is needed, the process for developing NCSEs, and the contents of NCSEs. NCSEs are required whenever a facility system or component containing fissile material is designed or modified so as to potentially affect credible criticality sequences. Determining whether an NCSE is required will involve an evaluation of all changes to a fissile material process to assess whether the potential exists to change or create an accident sequence leading to criticality. Changes having no potential criticality consequences do not need controls to be established and thus do not require an NCSE.

The described process for developing and approving NCSEs includes performance by qualified individuals, peer review, and approval by the NCS manager and affected line organizations. The described process is consistent with the guidance in Section 6.4.3.1 of NUREG-1718 and standard industry practice and is therefore acceptable to the staff. NCSEs will provide the basis for meeting the double contingency principle and ensuring that processes are subcritical under normal and credible abnormal conditions (as required by 10 CFR 70.61(d)). The NCSEs will also demonstrate that accident sequences leading to criticality will be highly unlikely (as required by 10 CFR 70.61(b)). The ISA will designate all criticality controls relied on to meet the above requirements as IROFS. (For a detailed discussion of the role that NCSEs play in performance of the ISA, see Section 6.2 of this SER.)

The selection of controls and controlled parameters follows a preferential hierarchy: passive controls are preferred over active engineered and engineered controls are preferred over administrative controls. The use of geometry control (in some instances in conjunction with neutron absorbers) is preferred. Controls are established to maintain parameters within their safety limits under normal and credible abnormal conditions. In response to Request for Additional Information (RAI) NCS-17[3] (NRC, 2008e), the applicant explained that sufficient controls and defense in depth will be employed to prevent the loss of controlled parameters. The applicant provided a practical example of a failure of the nonsafety control system, which could lead the system to attempt to overfill a jar, and even though one of the redundant IROFS safety systems could hypothetically fail, the other redundant IROFS would prevent overfilling of the jar and exceeding the mass limit. In other words, the two redundant IROFS provide defense in depth against exceeding the parameter under normal (typified by the nonsafety system failure) or abnormal (typified by the IROFS failure) conditions. In this case, even if the mass limit were to be exceeded, a loss of moderation control would also be required before criticality could become possible. Thus, the example does not mean that compliance with the double contingency principle would be based on a single parameter (which would violate the principle of diversity), but that dual controls are established on each parameter to ensure that its loss is sufficiently unlikely. For criticality parameters that are not controlled, or were controlled and then failed, the evaluation assumed optimum or worst-credible conditions. The applicant defined "worst-credible conditions" to mean the most reactive conditions that can reasonably be expected to occur without the need for any formal controls. An example would be the assumed maximum densities of different reference fissile materials based on experimental and historical evidence (e.g., it is not credible that unsintered powder would have densities in excess of 7 grams per cubic centimeter (g/cc)). The applicant's use of a preferential hierarchy, defense-in-depth practices, and conservative assumptions with regard to controlled parameter values is

[3] The reference section of this chapter of the SER (i.e., Section 6.4) includes the NRC's RAIs. Criticality safety-related RAIs are numbered sequentially and designated by the prefix "NCS." RAIs NCS-01 through NCS-60 are related to the LA review, and RAIs NCS-61 through NCS-96 are related to the ISA Summary review.

consistent with the guidance in Section 6.4.3.3.2 of NUREG-1718 and standard industry practices. Therefore, the applicant's approach is acceptable to the staff.

6.1.3.2 Criticality Controls and Controlled Parameters

LA Section 6.4.4 reiterates the preferential control hierarchy (adding the preference for enhanced over simple administrative controls), lists the 13 criticality parameters, and provides some general guidance for their use. LA Sections 6.4.4.1–6.4.4.13 list specific commitments applicable to each of the 13 parameters.

LA Section 6.4.4 states that MOX Services will follow the preferential control hierarchy "to the extent practical." In response to RAI NCS-21, the applicant stated that it meant that it would follow the hierarchy whenever practicable, given the nature of the process. As an example, the applicant explained that geometry control cannot be used in powder operations for sequences involving spills because the initiating event is caused by a loss of equipment geometry. In addition, laboratory, waste, and maintenance operations cannot practically be controlled by passive or active engineered means. Dual active controls on mass are used to detect powder spills, and manual processes are controlled by administrative means. During the CAR review, the staff evaluated the applicant's criticality control strategy (the dominant controlled parameters) for each of the major process units and determined that this strategy was consistent with standard industry practice for establishing controls for similar operations (as documented in Sections 6.1.3.4.1 and 6.1.3.4.2 of NUREG-1821, "Final Safety Evaluation Report on the Construction Authorization Request for the Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina," issued March 2005 (NRC, 2005a). The staff has similarly determined that the detailed facility design, including specifically the choice of controlled parameters, the means of controlling them, and the implementation of the double contingency principle, complies with the approach outlined in the CAR and is consistent with the guidance in Section 6.4.3.3 of NUREG-1718 and standard industry practice (for a more detailed discussion, see Section 6.2 of this SER).

For each of the 13 controlled parameters, tolerances are conservatively taken into account in establishing operating limits and controls. The applicant has committed to develop an operating limits manual, which will take process variability and uncertainties, including instrument errors, into account in accordance with standard industry practices (e.g., industry standards for setpoint determination). The staff has determined that correct implementation of these commitments will provide reasonable assurance of safety during operations. The NRC staff will verify completion of the operating limits manual as part of the verification of the principal structures, systems, and components (PSSCs), as required by 10 CFR 70.23(a)(8) and as discussed in Chapter 1 of this SER.[4] The applicant performed sensitivity studies to demonstrate that processes will be subcritical under all credible conditions. Sections 6.4.4.1–6.4.4.13 of the LA further supports this approach, stating that "limits are established in a manner that ensures an adequate margin of subcriticality (including margins to protect against uncertainties in process variables and against limits being accidentally exceeded)." The staff's review of criticality calculations, as part of the ISA Summary review (see Section 6.2 of this SER), indicated that facility calculations took process variability and uncertainty into account and modeled the most reactive combination of geometric and material tolerances. This approach is consistent with the guidance in

[4] The relevant PSSC for criticality safety is "criticality control," which consists of all measures relied on to meet the performance requirements of 10 CFR 70.61(b) and 10 CFR 70.61(d) and the double contingency principle of 10 CFR 70.64(a)(9) (i.e., all criticality safety-related IROFS).

Sections 6.4.3.3.1 and 6.4.3.3.2 of NUREG 1718 and standard industry practice. Therefore, the applicant's approach is acceptable to the staff.

LA Sections 6.4.4.1–6.4.4.13 discuss each of the 13 controlled parameters individually. For each parameter, the LA specifies the values included in the calculations, as well as the acceptance criteria for implementing parameter controls in the field. The staff reviewed these criteria against those in Section 6.4.3.3.2 of NUREG-1718 and determined that they were consistent with NUREG-1718 and with standard industry practice. Where ANSI/ANS-8 series standards endorsed in RG 3.71, Revision 1, "Nuclear Criticality Safety Standards for Fuels and Material Facilities," issued October 2005 (NRC, 2005b), apply to a particular parameter (e.g., moderation, neutron absorbers), the applicant has committed to follow all "shall" statements in the standard. The NCSEs and the ISA Summary will control the process variables by setting limits that can affect the value of controlled parameters. Limits will be derived using validated criticality calculational methods, rather than standards or handbooks (with the exception of mass—mass limits given in standards and handbooks are typically derived using very conservative assumptions including spherical geometry, optimum moderation, and full reflection). Specific issues regarding the individual parameters are discussed below.

The applicant stated in LA Section 6.4.4.4 that "determination of isotopic content is based on compliance with the double contingency principle." The fact that incoming feed material has an isotopic composition within certain ranges is a key bounding assumption upon which all of the criticality safety analyses and ISA assume. MOX Services explained in its response to RAI NCS-28 (NRC, 2009c) that it will verify the isotopic vector by two independent isotopic measurements whenever less than the bounding isotopic content is assumed. This approach is the same as that used when a criticality accident sequence relies on a single parameter that must be administratively controlled (such as reliance on dual concentration or density controls). The NRC staff reviewed the implementation of dual sampling for such controls during its ISA review (see Section 6.2 of this SER).

Some parametric limits are properties of the incoming pit disassembly and conversion facility (PDCF) or alternate feedstock (AFS) material, and as such, constitute global assumptions that apply to numerous processes and accident sequences. These assumptions are not generally repeated explicitly for each accident sequence. The main parametric limits of this kind include specifications for the isotopic vector (abundance of the various plutonium isotopes, ^{235}U enrichment, and the ratio of uranium to plutonium (U/Pu)) and the density, moderation, and level of chemical impurities of received feed powder. [redacted]

The staff reviewed the applicant's approach for ensuring that incoming feed material would be within the required specifications, as described in its response to RAI NCS-29 (NRC, 2009c). The vendor for the feedstock, the U.S. Department of Energy (DOE), will be required to undergo a qualification audit to ensure that its quality assurance plan meets the applicable requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and nuclear quality assurance (NQA-1), as well as periodic performance evaluations and audits. The applicant will perform surveillance testing of the incoming feed material during the first production run, and again after 6 months. Further surveillance will depend on these initial results. An interface control document will specify the requirements for DOE to perform its own testing of the product to be supplied to the MFFF and the contents of the documentation provided along with the material. The exact details of these requirements, including how compliance with the double contingency principle will demonstrated, have not yet been determined. The staff will verify completion of the requirements for ensuring that the vendor will comply with bounding specifications for the incoming feed material, consistent with the double contingency principle, as part of the verification of the PSSCs.

Besides performing periodic audits of the vendor's quality assurance program, the applicant will also independently confirm certain credited characteristics of the material. Some of the aforementioned characteristics—notably the moisture content—cannot be measured before opening the cans. Before measurement, the vendor's quality assurance program will be the primary means of ensuring that the material is within specifications. However, the 3013 cans are of robust construction and the analysis done in developing the 3013 standard showed the cans to be safe (i.e., the cans have been analyzed safe from a criticality perspective) under worst-case conditions, so the risk of criticality is negligible until the material is introduced into the process (by which time the measurements will have been performed). Nevertheless, the applicant stated that it will independently confirm the fissile content of incoming material (which, in conjunction with can weight or calorimetric measurements, will provide the plutonium isotopic ratio) using gamma spectroscopy. X ray measurements of the level within the cans, in conjunction with can weight, will be used to determine the density of unopened cans. The impurity level of the material will be determined using nondestructive analysis (NDA) measurements, with any weight unaccounted for conservatively assumed to be chemical impurities.

The staff reviewed the results of preliminary tests of the density measurement methodology (NRC, 2010a) and finds that there is reasonable assurance that the applicant will be able to determine that material is within required specifications. The staff concluded that license commitments with regard to measurement of these specifications represented a valid approach. However, the applicant stated that it would confirm final acceptability of its measurement techniques during startup testing. This includes calibrating the NDA measurements to confirm the mass and isotopic characteristic of incoming feed material, which must be verified because of the different isotopic characteristics between the MFFF (PDCF and AFS) feed material and the reference plant material. The staff will confirm completion of methods for independently verifying the acceptability of incoming feed material (including demonstration of the feasibility of measurement methods) as part of the verification of the PSSCs.

With regard to the applicant's commitments concerning reflection control in LA Section 6.4.4.5, the staff asked the applicant to explain the kinds of conditions that constitute a loss of reflection control. Statements in the ISA Summary referred to a loss of reflection control as involving "leak of a process material achieving an unsafe volume" and "leak of solution that could provide additional reflection of fissile bearing process equipment." In its written response to RAI NCS-30 (NRC, 2009c) and subsequent verbal discussions, the applicant clarified that loss of reflection control involved only leaks from pipes with nonfissile liquids; leaks of fissile solutions would typically involve a loss of geometry control. The applicant also clarified in its responses to RAIs NCS-31 and NCS-32 (NRC, 2009c) that "sufficient water reflection" meant that (1) at a minimum, a 1-inch (2.5-centimeter) tight-fitting water jacket would be assumed to bound transient reflection, such as personnel, when reflection is controlled, and (2) a 12-inch (30-centimeter) tight-fitting water jacket would be assumed when reflection was not controlled.

These criteria are consistent with the guidance in Section 6.4.3.3.2.5 of NUREG-1718 and standard industry practice. Therefore, the applicant's criteria are acceptable to the staff.

The applicant made statements in LA Section 6.4.4.6 concerning "restrictions placed on the use of moderating materials" as fire suppressants and observed that "the effect of credible fire events and the consequences associated with the potential use of moderating material in mitigating such fires are evaluated." In its response to RAI NCS-36 (NRC, 2009c), the applicant noted that LA Section 7.3.3.1 states, "Due to nuclear criticality safety concerns, hydrogenous material (e.g., water) is not used as a suppression agent in process rooms and in areas that contain nuclear material." The applicant clarified that its intent is to prohibit hydrogenous material in such areas and provided a cross-reference to LA Section 7.3.3.1 in Section 6.4.4.6 to further clarify its commitment.

The applicant stated in LA Section 6.4.4.7 that, "When sampling of the concentration is specified, a program based on dual independent sampling and analysis using independent verification sampling methods using two people is implemented." In its response to RAI NCS-39 (NRC, 2009c), the applicant clarified that these conditions applied when concentration sampling alone was relied on to meet the double contingency principle. The review of the ISA Summary showed that, when the basis of criticality safety included concentration control, the two legs of double contingency were generally redundant sampling controls. Section 6.2 of this SER discusses the staff's review of the applicant's dual sampling methods. The applicant included a reference to its Long-Term Fissile Material Accumulation Program (LTFMAP) as part of its response to RAI NCS-40 (NRC, 2009c). This program consists of non-IROFS radiation monitoring to detect long-term accumulation that could eventually result in overconcentration in process tanks.

In LA Section 6.4.4.10, the staff noted that single-parameter volume limits are based upon worst-case geometry and material composition (unless these parameters are controlled by IROFS). The staff noted that, unlike the discussion for other parameters, this section did not state that worst-case moderation and conservative reflection conditions would be used. However, the staff considers this to be addressed by the general commitment that "optimum or worst-credible conditions are assumed for parameters unless they are specifically controlled" (LA Section 6.4.4). The discussion of single-parameter limits also does not mention a specific percentage of the critical volume to be used, as included in Section 6.4.3.3.2.2 of NUREG-1718. However, in its response to RAI NCS-43 (NRC, 2009c), the applicant stated that single-parameter limits will be based on explicit calculations (which are required to be validated). The use of critical values from standards and handbooks, with appropriate margin, is one acceptable method of ensuring subcriticality; the use of validated calculational methods, with appropriate margin, is another. Therefore, the lack of specific margins in the volume limits discussion is acceptable, based on the applicant's commitment to perform explicit calculations that are subject to NRC staff review.

LA Section 6.4.4.10 concerned process variable control for criticality safety. The staff had asked the applicant to provide examples of what constituted process variable control. In its response to RAI NCS-46 (NRC, 2009c), the applicant stated that it did not currently rely on process variable control and therefore could not provide any specific examples. There were several cases in the ISA in which parameters other than the standard 13 were controlled for criticality safety (e.g., temperature and nitric acid pH for polymerization prevention, furnace temperature, reagent addition controls). However, in each case, the process variable was tied to one of the standard 13 parameters. The temperature and acidity of plutonyl nitrate was credited with maintaining the physicochemical form of the material as a solution. The furnace temperature was credited with controlling the moderation level in PuO_2 or MOX product. Reagent addition controls were credited with controlling concentration, physicochemical form, and other parameters. Control over those process variables that can affect other controlled parameters is essential to ensure that all processes will be subcritical under normal and credible abnormal conditions. To this end, the applicant has committed (in each applicable section which discusses one of the other controlled parameters) that, when process variables can affect one of the other controlled parameters, they are defined and controlled as IROFS in the NCSEs and ISA Summary.

The staff finds that the applicant's approach to implementing criticality controls and limits on controlled parameters is consistent with standard industry practice and the guidance in Sections 6.4.3.3.2.0 through 6.4.3.3.2.12 of NUREG-1718. Therefore, the approach is acceptable to the staff.

6.1.3.3 Requirements in 10 CFR 70.24, "Criticality Accident Requirements"

LA Section 6.3 describes the criticality accident alarm system (CAAS) as a monitoring system composed of groups of detectors called monitoring units that will activate audible and visual alarms in case of a criticality accident. The evaluation of the effectiveness of CAAS detectors (detection criteria and location/spacing) takes into account the effect of existing shielding. CAAS detector coverage is determined using three-dimensional radiation transport codes. The CAAS is 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 (R2003), "Criticality Accident Alarm System." The applicant maintains a CAAS consistent with the requirements of 10 CFR 70.24 and ANSI/ANS-8.3-1997 (R2003) (as endorsed by RG 3.71, Revision 1).

The applicant maintains a CAAS that is designed to remain operational in the event of a seismic shock equivalent to the MFFF design-basis earthquake. CAAS components are protected so as to reduce the potential for damage in case of fire, explosion, corrosive atmosphere, or other probable extreme conditions. The CAAS is designed to reduce the potential of failure, including false alarms, and provides a visual or audible warning signal to indicate system malfunction or the loss of primary power.

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

The MFFF staff maintains an emergency procedure which covers the entire facility, including locations where licensed SNM is handled, used, or stored, to ensure that personnel can be withdrawn to a safe area upon the actuation of the CAAS alarm notification. The nuclear criticality accident onsite emergency planning and response for the MFFF staff follows the guidance in ANSI/ANS-8.23-1997, "Nuclear Criticality Accident Emergency Planning and Response."

The staff has reviewed the applicant's commitment to the CAAS requirements identified in 10 CFR 70.24 and finds it acceptable because MOX Services maintains a CAAS capable of energizing a clearly audible alarm signal if accidental criticality occurs. In addition, the applicant maintains emergency procedures for each area in which SNM is handled, used, or stored to ensure prompt personnel evacuation upon sounding of the alarm. Furthermore, the staff finds that the commitments in this section are consistent with the guidance in Section 6.4.3.3.3 of NUREG-1718 and standard industry practice (especially ANSI/ANS-8.3-1997), as well as with the requirements of 10 CFR 70.24(a). Therefore, these commitments are acceptable to the staff.

6.1.3.4 Criticality Methods and Code Validation

Subcriticality is demonstrated using validated criticality calculational methods. LA Section 6.4.5.2 describes the SCALE Monte Carlo code package the applicant used to perform the criticality calculations. The staff noted that the spacing limits for hand-carried items in the RCA unit were derived using the solid angle technique, a hand calculation method. In its response to RAI NCS-48 (NRC, 2009c), the applicant explained that the spacing limits were ultimately based on validated computer methods. The solid angle technique is a widely accepted method for determining subcriticality, (as documented in Knief, 2000). Because of this, the applicant’s use of the solid angle method, as confirmed by validated computer methods, is acceptable to the staff.

The staff reviewed the applicant’s description of its validation methodology and requirements for documenting the results in a formal validation report. This information agreed with the acceptance criteria in Section 6.4.3.3.4 of NUREG-1718 and with standard industry practice, with one exception. The LA does not contain a commitment to describe the validated areas of applicability (AOAs) in the validation report. Table 6.4-1 of the LA describes the AOAs; therefore, there is no need to duplicate this information in the validation reports themselves.

Part of defining the code’s AOA involves specifying the code options that were used to analyze the critical benchmark experiments. The staff noted that some of the calculations in the KCD unit made use of the concrete albedo option, which had not been included in the criticality code’s validation. Albedos are a way of representing the effect of external absorbers that can shorten running times by avoiding tracking neutrons explicitly in the surrounding material. In response to RAI NCS-80 (NRC, 2009c), the applicant stated that it will revise these calculations to explicitly track neutrons in the external reflectors rather than using the albedo option. The staff will verify completion of criticality calculations to support operation of the MFFF as part of the verification of the PSSCs.

LA Section 6.4.5.4 describes the applicant’s statistical methodology used to determine the bias and uncertainty. The staff reviewed this method, as well as the applicant’s validation reports, during its evaluation of the CAR (as documented in Section 6.1.3.5.1 of NUREG-1821). The applicant has also committed in the LA to include the validation report in its configuration management program and to maintain the software in accordance with the MOX Project Quality Assurance Plan (MPQAP). The staff will verify the applicant’s development of the procedures for maintaining configuration control over calculational methods as part of the verification of the PSSCs.

The staff also reviewed a sample of several of the applicant’s calculations in the ISA review and determined that they were within the validated AOAs. The staff finds that the applicant’s implementation of validation is consistent with the approach reviewed and approved at the CAR stage (in Section 6.1.3.5.1 of NUREG-1821). Therefore, this approach is acceptable to the staff.

6.1.3.5 Criticality Safety in the Integrated Safety Analysis

LA Section 6.4.6 describes the process for review and approval of NCSEs, which are considered part of the ISA. The NCSEs identify IROFS as controls that are relied on to demonstrate that processes are subcritical under normal and abnormal conditions and that criticality is highly unlikely. Section 6.2 of this SER includes a detailed discussion of how accident sequences are developed in the process hazard analyses (PrHAs), evaluated in the NCSEs, and incorporated into the ISA Summary. The accident sequences mainly consist of initiating events that result in a loss of one of the controlled parameters, together with two trains of preventive IROFS to ensure double contingency protection. The tables in Section 5.3.7 of the ISA Summary describe these accident sequences. The commitments in this section are consistent with the guidance in Chapter 5 and Section 6.4.3.3.6 of NUREG-1718 and standard industry practice and are therefore acceptable to the staff.

6.1.4 Regulatory Guidance

The staff reviewed the applicant's commitment to the ANSI/ANS-8 series consensus standards related to criticality safety, as described in LA Section 6.5. The applicant commits to using the ANSI/ANS-8 series standards endorsed in RG 3.71 in the design of the facility. LA Section 6.5 commits, in general, to "comply with the guidance (shall statements) and implement the recommendations (should statements)" of applicable standards, but identified several clarifications to specific commitments within certain of the standards. Clarifications to specific commitments to follow these standards are summarized below:

- <u>ANSI/ANS-8.1-1998</u>: The applicant commits to comply with the guidance (i.e., the "shall" statements) of ANSI/ANS-8.1-1998 and implement the recommendations (i.e., the "should" statements), with clarification of three provisions in Sections 4.2.2, 4.2.3, and 4.3.2 of the ANSI standard. For Section 4.2.2, the applicant commits to follow the double contingency principle, 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." Compliance with the double contingency principle will be demonstrated for those processes and areas in which criticality is determined to be credible during the performance of the ISA. The staff 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 accordance with the definitions in Section 6.8 of NUREG-1718. The staff evaluated this definition and found it to be acceptable during its review of the CAR (see Section 6.1.4.2 of NUREG-1821). Therefore, the staff finds the applicant's approach to be acceptable.

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 (i.e., passive controls are preferred to active; engineered controls are preferred to administrative) and is therefore acceptable to the staff. Also in terms of Section 4.3.2 of the ANSI standard, the applicant committed that, where an extension to the AOAs is required, the calculational method will be supplemented by other calculational methods to provide a better estimate of bias in the extended areas or through an increase in the margin of subcriticality. In making the extension, trends in the bias and any additional uncertainty must be considered in determining the appropriate amount of margin. Section 4.3.2 of the ANSI standard states that the AOA may be extended by making use of trends in the bias, and Section 4.3.3 states that the uncertainty in the bias shall contain allowances for extensions to the AOA. These commitments are consistent with the standard. The NRC has endorsed this standard; thus, the process is acceptable to the staff. This allowance could be construed as permitting the applicant's AOA to be broadened beyond the point described in the NRC-reviewed validation reports. However, because Table 6.4-1 of the LA describes the AOAs, they cannot be changed without using the applicant's change process which may require NRC review and approval.

With regard to the subcritical limits in ANSI/ANS-8.1-1998, the staff considers the use of subcritical limits from the standard in lieu of explicit calculation to be an acceptable practice. RG 3.71 has endorsed these results, and as single parameter limits, the NRC finds that they are conservative.

- <u>ANSI/ANS-8.3-1997 (R2003)</u>: The applicant commits to comply with the guidance of the standard and implement the recommendations, as modified by RG-3.71. The staff finds this approach acceptable.

- <u>ANSI/ANS-8.5-1996, "Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material"</u>: The applicant does not consider this standard to be part of the design basis of the facility. In addition, 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 (R2001), "Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors." The staff concurs that commitment to this standard is not applicable to the design of the facility and, therefore, is not part of the design basis of the facility.

- <u>ANSI/ANS-8.6-1989, "Safety in Conducting Subcritical Neutron-Multiplication Measurements In Situ"</u>: The applicant does not consider ANSI/ANS-8.6-1989 to be part of the design basis of the facility. The applicant does not intend to conduct subcritical neutron multiplication measurements at the facility. The staff concurs that commitment to this standard is not applicable to the design of the facility and, therefore, is not part of the design basis of the facility.

- <u>ANSI/ANS-8.7-1998, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials"</u>: 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-8.1-1998 and technical practices, as described in LA Sections 6.3 and 6.4.4, are sufficient to ensure that criticality safety is appropriately provided for fissile material storage areas. The staff concurs that commitment to this standard is not applicable to the design of the facility and, therefore, is not part of the design basis of the facility.

- <u>ANSI/ANS-8.9-1975 (R1995), "Guide for Nuclear Criticality Safety for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Materials"</u>: 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. The applicant will evaluate piping configurations containing aqueous solutions of fissile material by calculation, in accordance with ANSI/ANS-8.1-1998. 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 (R2005), "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. The facility's approach is to prevent criticality in accordance with the double contingency principle, rather than to rely on shielding and confinement for dose mitigation. 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 (R2002), "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-8.1-1998 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 (R1995), "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-8.1-1998. Because using validated methods to determine subcritical limits is an acceptable methodology, the staff determined that this approach was acceptable.

- ANSI/ANS-8.17-2004, "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors": The applicant commits to comply with the guidance (i.e., the "shall" statements) of ANSI/ANS-8.17-2004 and implement the recommendations (i.e., the "should" statements), with clarification of two provisions in Sections 4.11 and 5.1 of the standard. For Section 4.11, the applicant commits to use of the double contingency principle for the handling, storage, and transportation of fuel units and rods. As required by 10 CFR 70.64(a)(9), the facility must comply with the double contingency principle; thus, this commitment is acceptable to the staff. For Section 5.1 of the standard, the applicant committed that when an extension to the AOA is required, the calculational method will be supplemented by other calculational methods to provide a better estimate of bias in the extended area, or through an increase in the margin of subcriticality. This is consistent with the endorsed standard and is, therefore, acceptable to the staff. MOX Services intends to adhere to the exception noted in RG 3.71, which states that licensees and applicants may take credit for fuel bum up only when the amount of burnup is confirmed by physical measurements that are appropriate for each type of fuel assembly in the environment in which it is to be stored. MOX Services has stated that fresh fuel isotopics will be used throughout the facility, and therefore commitments related to burnup credit are not applicable.

- ANSI/ANS-8.19-2005, "Administrative Practices for Nuclear Criticality Safety", The applicant commits to comply with the guidance of ANSI/ANS-8.19-2005 and implement the recommendations, with the exception that no commitments are made related to Section 10 of the standard regarding emergency response to criticality accidents. The staff notes that the applicant has committed to ANSI/ANS-8.23-1997, which contains many of the same requirements as specified in Section 10; the staff finds this acceptable.

- ANSI/ANS-8.20-1991 (R1999). "Nuclear Criticality Safety Training", The applicant commits to comply with the guidance of ANSI/ANS-8.20-1991 (R1999) and implement the recommendations (i.e., the "should" statements) without exception or clarification. The staff finds this acceptable.

- ANSI/ANS-8.21-1995 (R2001), "Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors": The applicant commits to comply with the guidance of ANSI/ANS-8.21-1995 (this standard contains no recommendations) without exception or clarification. The staff finds this acceptable.

- ANSI/ANS-8.22-1997, "Nuclear Criticality Safety Based on Limiting and Controlling Moderators": The applicant commits to comply with the guidance and implement the recommendations (i.e., the "should" statements) of ANSI/ANS-8.22-1997 without exception or clarification. The staff finds this acceptable.

- ANSI/ANS-8.23-1997: The applicant commits to comply with the guidance (i.e. the "shall" statements) and implement the recommendations of ANSI/ANS-8.23-1997 without exception or clarification. The staff finds this acceptable.

ANSI is developing and revising its standards continually. ANSI published several new standards during the lifetime of the review, but they were not used in the design of the MFFF. The above list includes all applicable NRC-endorsed standards.

In summary, the design basis of the facility includes the following NRC-endorsed standards, in whole or in part: ANSI/ANS-8.1-1998, ANSI/ANS-8.3-1997 (R2003), ANSI/ANS-8.17-2004, ANSI/ANS-8.19-2005, ANSI/ANS-8.20-1991 (R1999), ANSI/ANS-8.21-1995 (R2001), and ANSI/ANS-8.22-1997; the applicant has also committed to ANSI/ANS-8.23-1997. The other standards were either not applicable to the design of the facility or adequately covered by other commitments. The staff considers this list to represent an acceptable set of design bases of the facility. However, nothing precludes the use of any other standards endorsed in RG 3.71. As stated in the discussions of each applicable ANSI/ANS standard above, the staff finds that the applicant's commitments are consistent with the guidance in Section 6.4.2 of NUREG-1718.

6.2 ISA Summary Review

The staff reviewed selected portions of the MFFF ISA Summary and supporting ISA process safety documentation (e.g., NCSEs, nuclear safety evaluations (NSEs), criticality calculations, equipment drawings, piping and instrumentation diagrams (P&IDs)) to determine whether the applicant had met the criticality hazard requirements for licensing related to performance of the ISA (as listed in 10 CFR 70.66(a)). Specifically, the staff reviewed the aforementioned ISA documentation to determine whether the applicant had (1) identified all credible accident sequences leading to inadvertent criticality, (2) established sufficient engineered and administrative IROFS to ensure that those sequences will be highly unlikely, (3) ensured that all processes will be subcritical under both normal and credible abnormal conditions, and (4) complied with the double contingency principle.

The applicant divided the MFFF into 53 different process units for ease in performing the criticality analysis. Many of these contained similar hazards or repetitive controls. Thus, the staff primarily reviewed those process units judged to have the highest potential risk of inadvertent criticality, as well as some selected process units posing lower risk. This approach enabled the staff to review the applicant's implementation of its ISA methodology across a wide spectrum of systems having different fissile material compositions, physical characteristics, and criticality controls. [redacted]

The staff performed a risk-informed review of the ISA; that is, the staff reviewed first, and in greatest depth, those areas that pose the highest unprevented risk (risk based on the type and quantity of material present, without taking into account risk reduction produced by IROFS). [redacted]

- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]

The basis for this ranking is the known physical and neutronic properties of fissionable materials, as well as experience with the types and quantities of materials in accidents that have occurred (Los Alamos, 2000).

Besides the unprevented risk, the staff used several other factors to decide in what areas to concentrate its review. These factors included complexity of the controls (e.g., an active engineered interlock is much more complicated than a passive geometry control), diversity, redundancy, significance for downstream units, and safety margin. The staff spent more time reviewing complex IROFS, such as active interlocks and enhanced or multistep administrative controls, than simple controls for which it is relatively easy to assess adequacy. Since numerous accident sequences have similar control strategies (e.g., passive geometry with drip trays, mass limits based on IROFS scales in conjunction with moderation), once the staff reviewed and obtained a reasonable assurance of safety with a given control strategy, it focused its review on other types of control strategies. This approach allowed the staff to review a diverse cross-section of passive engineered, active engineered, and administrative controls. Those systems relying on redundant IROFS, especially redundant administrative IROFS, also received added attention because of their susceptibility to common-mode failure. Some IROFS (e.g., controls maintaining the correct isotopic blend of MOX powder) received in-depth review because failures would propagate to many process units downstream. Lastly, those control systems having a small safety margin (especially for administrative IROFS) received more in-depth review. These criteria were based on lessons learned during the staff's ISA reviews for other licensees and applicants and historical events at fuel facilities. Finally, following its visit to the reference facilities in France, the staff focused its review on the processes and control systems that differed from those at the reference facilities, because differences between the reference facilities and the MFFF represent areas without an extensive track record of historical experience and thus greater uncertainty.

Once the most risk-significant portions of the facility and types of controls most likely to fail were identified, the staff performed its review in the following manner—several in-office reviews were conducted, starting with the main aqueous polishing processes. Within each area, the staff reviewed the applicable NCSE to identify the controlled parameters, controls, and limits used to ensure criticality safety. The staff reviewed accident sequences in the double contingency discussion to ensure that they met the double contingency principle; the staff also reviewed calculations to determine that all normal and credible abnormal conditions were demonstrated to be safely subcritical. The staff selectively reviewed the IROFS tables in the NCSE to determine that the IROFS met the four required criteria for being highly unlikely (single-failure criterion or double contingency principle, application of Appendix B to 10 CFR Part 50 and NQA-1, industry codes and standards, and management measures including failure detection) and reviewed the applicant's basis for classifying the sequence as a Category A, B, or C highly unlikely event. After review of the NCSE and supporting documents and calculations, the staff selectively sampled the flowdown of hazards from the PrHAs into the NCSE to determine whether sequences screened out as incredible had been properly evaluated. Following its review of the applicant's ISA documentation, the staff reviewed the ISA Summary description of the process area, accident sequences, and IROFS to ensure that it accurately reflected the process's safety basis and met all applicable regulatory requirements. The staff reviewed all process areas, with the exception of those that were essentially duplicates of another process area using a similar control strategy.

The staff documented its review of the units discussed in the following sections in a series of in-office review summaries, as listed below (NRC, 2008a; NRC, 2010a). The staff reviewed the following units and issues on the dates indicated:

- KDB, NDP, LCT (April 17–19, 2007)
- NBX/NBY, NDS, KPA (July 9–13, 2007)
- KPA, NDP, KCD (August 14–16, 2007)

- KDB, KCA, KPB, KPC, KWD, KWS, KWG, KPG (September 25–27, 2007)
- KPB, KCA, KDA/KDM, KDD, KCD (November 6–8, 2007)
- KDA/KDM, PFE/PFF, PRE/PRF (December 4–6, 2007)
- NCR, PSE/PSF/PSI/PSJ, PML, DCM, STK, SMK, VDR, TAS (January 8–10, 2008)
- DCE, NDD, VDQ, VDT (January 22–24, 2008)
- NXR, RCA (February 5–6, 2008)
- Cracked Concrete Issue (September 23, 2008)
- KPA, KDB, KCA, RCA (December 16–17, 2008)
- KPA, KPB, KCA, KDA/KDM, PRE/PRF, VDQ, VDT (December 15–16, 2009)

The vertical slice reviews consisted of (1) reviewing selected process unit NCSEs and calculation documents to assess the adequacy of criticality parameters, controls, and limits to meet the double contingency principle; (2) reviewing NCSEs and PrHA documents to assess the completeness and adequacy of accident sequences and IROFS to meet the performance requirements; (3) reviewing other supporting safety documentation to assess the applicant's ability to implement criticality controls, limits, and management measures in the plant's safety program; and (4) reviewing the applicant's ISA Summary to ensure the correct flowdown of information (i.e., the contents required by 10 CFR 70.65(b)(1)–(9) flow from these documents into the ISA Summary).

In addition to the in-office document reviews listed above, the staff also visited the two reference plant facilities in Marcoule and La Hague, France, and documented its review of operations at these two facilities (NRC, 2008c). There the staff observed ongoing processes similar to those designed for the MFFF, reviewed several additional technical documents, and discussed technical issues with the facilities' criticality safety and operations staff. Part of this review consisted of discussing the facilities' operational history, including observed failure rates of safety controls and reportable events. Based on its review of documents and operations in the above vertical slice reviews, the staff has reasonable assurance that the regulatory requirements of Subpart H, "Additional Requirements for Certain Licensees Authorized To Possess a Critical Mass of Special Nuclear Material," of 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," will be met for criticality safety at the MFFF, as discussed below.

6.2.1 Programmatic Aspects of Review

As stated in 10 CFR 70.66(a), an LA will be approved if the applicant has complied with the requirements of 10 CFR 70.21, "Filing"; 10 CFR 70.22, "Contents of Application"; 10 CFR 70.23, "Requirements for the Approval of Applications"; and 10 CFR 70.60, "Applicability," through 10 CFR 70.65, "Additional Content of Applications." This section only discusses the staff's review of the ISA and ISA Summary, addressing the requirements of Subpart H (10 CFR 70.60 through 10 CFR 70.65), as applied to preventing criticality hazards. Other chapters of the SER address the application of Subpart H requirements to other types of hazards.

The applicant performed its ISA for all hazards at the MFFF in accordance with the methodology discussed in Chapter 5 of this SER; thus, aspects of the methodology common to all safety disciplines will not be discussed further. Several unique considerations, which apply only to criticality hazards, are discussed in greater detail below. These aspects of the methodology are in large part the result of discipline-specific considerations and regulatory requirements.

<u>10 CFR 70.61(b), 10 CFR 70.61(d), and 10 CFR 70.64(a)(9)</u>

Throughout the nuclear industry, a criticality accident is generally presumed to be a high-consequence event because the potential for radiological doses to workers exceeding the threshold of 10 CFR 70.61(b)(1) cannot be discounted. At the MFFF, a criticality is also defined to be a high-consequence event. Neither the LA nor the ISA Summary explicitly states that criticality is a high-consequence event. However, the applicant's methodology requires that criticality accidents be demonstrated to be highly unlikely, so the applicant is effectively treating criticality as a high-consequence event. Consequence determination therefore is trivial and demonstrating compliance with the performance requirements consists only in determining the likelihood for all credible accident sequences leading to criticality. Because of this, criticality-related IROFS can only be preventive and not mitigative in nature; no credit is taken for shielding or other natural or design features that could reduce the consequences to less than the threshold of 10 CFR 70.61(b)(1). This is consistent with the requirement of 10 CFR 70.61(b)(1), as well as that of 10 CFR 70.61(d), which states that all nuclear processes must be shown to be subcritical (regardless of whether there is a dose) under both normal and credible abnormal conditions and that criticality control must rely on primarily preventive means.

To satisfy the requirement that all nuclear processes be subcritical, including use of an approved margin of subcriticality for safety, the applicant used a minimum margin of subcriticality of 0.05. The staff reviewed the basis for this at length in the review of the CAR, as documented in that SER (Section 6.1.3.5.2 of NUREG-1821). Margin was also provided in making use of conservative calculational assumptions and technical practices, as discussed in Section 6.1.3 of this SER. During the ISA review, the staff verified that calculations were within the AOAs that were reviewed and approved in the CAR review. The staff also verified that the calculations were consistent with technical practices specified in the LA. In some calculations, particularly with regard to the inclusion of certain neutron poisons, conditions were slightly outside the approved AOA. The staff noted that relevant NCSEs provided the justification, which generally relied on the fact that the calculated k_{eff} was substantially subcritical or that the worth of the absorber was sufficiently small that even a gross error in the absorption cross-sections would not be sufficient to render the system critical. The staff found these arguments to be sound.

The staff did note, however, that in some cases calculations assumed parameter values identical to so-called "nominal" design values, without any apparent margin. Although the exact dimensions of fixed equipment have been largely determined, several limits (e.g., drip tray level setpoints, radiation trip points) have yet to be determined, so the margins associated with these controls are not yet known. In its responses to RAIs NCS-61 and NCS-62, the applicant committed to developing operating limits manual to ensure adequate margin and provide details as to how it will determine applicable safety limits, analytical limits, instrument setpoints, and operating limits in its responses (NRC, 2009c). The staff determined that the methodology described in the applicant's RAI responses was consistent with standard industry practice (as described in RG 1.105, "Setpoints for Safety-Related Instrumentation," and American National Standards Institute/International Society for Automation (ANSI/ISA) S67.04.01-2000, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation"). Therefore, the staff finds the applicant's methodology to be acceptable. The staff will confirm that MOX Services has developed an operating limits manual to establish sufficient margin to meet the performance requirements during the verification of the PSSCs, as discussed in Section 6.1.3.2 of this SER.

In addition, because the MFFF is a new facility, the design must comply with the double contingency principle, in accordance with 10 CFR 70.64(a)(9). To do this, MOX Services established at least two independent IROFS on each credible accident sequence leading to criticality, supported by an analysis (a NCSE) to demonstrate that at least two unlikely, independent, and concurrent changes in process conditions would be required before criticality is possible.

Ensuring that high-consequence events are highly unlikely is based on the following four qualitative criteria:

(1) application of the single-failure criterion (double contingency principle for criticality)

(2) application of the MPQAP, which meets the requirements of Appendix B to 10 CFR Part 50 and NQA-1

(3) application of industry codes and standards

(4) management measures, including surveillance of IROFS

These criteria apply to accident sequences in all safety disciplines, but because of the more stringent regulatory requirements related to criticality hazards, the applicant provided the following additional criteria in LA Section 5.2.2.7.2. For some accident sequences relying on robust passive controls, the following is sufficient to demonstrate that the sequence is inherently highly unlikely:

> For robust passive features with no credible failure leading to criticality, the equipment must be specified as an IROFS, must be evaluated and shown to be subcritical under all credible process conditions, and must be under the facility's configuration management program.

For all other systems that have credible accident sequences leading to criticality, the following criteria will apply:

- application of at least two independent, robust (unlikely to fail) controls
- active or passive engineered controls, which are unlikely to fail, based on consideration of all applicable "available and reliable" qualities, in accordance with NUREG-1718, and which must be classified as quality level QL-1
- administrative controls which are robust and unlikely to fail, based on consideration of all applicable "available and reliable" qualities, in accordance with NUREG-1718, and which must be simple and unambiguous

These criteria, which are discussed further in Chapter 5 of this SER, are often referred to as the "four pillars" that support the determination that an accident sequence is "highly unlikely," for all facility hazards. In addition, criticality hazards must be supported by one additional pillar, which requires additional failure detection, subcritical margin, or other comparable assurance that controls are robust enough to be used as the basis for meeting the double contingency principle (which, together with the subcriticality requirement of 10 CFR 70.61(d), is unique to criticality). All credible criticality accident sequences are therefore classified into one of three "highly unlikely categories," defined as follows:

- Category A events have the means to detect failure of the control within a specified period of time.

- Category B events include a safety margin that demonstrates that multiple (i.e., three or more) failures of each independent robust control will not result in a loss of subcriticality.

- Category C events possess some other means, with justification, which provide comparable assurance to Categories A or B above.

The staff reviewed and approved these criticality-specific portions of the applicant's ISA methodology during the CAR review (see Section 6.1.4.2 of NUREG-1821), consistent with the guidance contained in Section 6.4.3.3.6 and Chapter 5 of NUREG-1718. The staff's review of the ISA and ISA Summary focused on whether the applicant had implemented its methodology adequately, so as to demonstrate compliance with the performance requirements and double contingency principle.

<u>10 CFR 70.62(a–d)</u>

As required by 10 CFR 70.62(a), each applicant must establish and maintain a safety program demonstrating compliance with the provisions of 10 CFR 70.61, "Performance Requirements." For criticality safety, the safety program meeting this is the applicant's NCS Program, discussed earlier in Section 6.1 of this SER. In addition, 10 CFR 70.62(b) requires each applicant to maintain process safety information, and 10 CFR 70.62(c) requires each applicant to conduct and maintain an ISA. Furthermore, 10 CFR 70.62(d) requires each applicant to establish management measures to ensure compliance with 10 CFR 70.61. Chapter 5 of this SER discusses the overall review of these three elements of the safety program (i.e., process safety information, ISA, and management measures). The staff confirmed the adequacy of the implementation of these elements to criticality safety hazards during its in-office licensing reviews, as discussed below.

<u>10 CFR 70.65(b)(1)–(9)</u>

The provisions of 10 CFR 70.65(b)(1)–(9) specify the required contents of the applicant's ISA Summary. The staff determined, based on its review of commitments in the LA and its review of the ISA Summary, including the vertical slice review, that the applicant had met the requirements specified in 10 CFR 70.65(b)(1)–(9) with regard to criticality hazards, as summarized below.

The regulation at 10 CFR 70.65(b)(1) requires a general description of the site, with an emphasis on the factors that could affect safety. The regulation at 10 CFR 70.65(b)(2) requires a general description of the facility, with an emphasis on the areas that could affect safety. This information, which is contained in Chapters 2 and 3 of the ISA Summary, is general in nature and not specific to criticality safety. See Section 1.3 of this SER for more specific information.

The provisions of 10 CFR 70.65(b)(3) require a description of each process analyzed in sufficient detail to understand the theory of operation, the hazards, and a general description of the types of accident sequences. Chapter 4 of the ISA Summary includes detailed process descriptions, including detailed schematics. Section 5.3.7 of the ISA Summary discusses the parameters which are controlled for criticality safety, as well as the major process features that control them. The staff reviewed this information along with information available on site in the NCSEs, system description documents, criticality calculation documents, mechanical drawings, P&IDs, and other related documents. Based on its review of this information, the staff concludes that there is sufficient detail to understand the process and its theory of operation. The PrHA document describes the process hazards and other initiating events postulated to lead to an inadvertent criticality for the various process units. Not all scenarios considered were determined, upon further evaluation, to credibly lead to criticality. Some scenarios considered in the PrHAs were determined to not be credible or to lead to conditions subsequently shown to be subcritical. The tables in the ISA Summary did not explicitly include those scenarios that were not credible, or which did not lead to criticality. The tables also did not include those scenarios determined to be "inherently highly unlikely," as described above. However, the controlled parameter discussion in Section 5.3.7 of the ISA Summary did discuss the corresponding scenarios and identified the IROFS limiting those parameters by underscoring them. For all other events, the tables in Section 5.3.7 of the ISA Summary explicitly included highly unlikely and double contingency demonstrations. For each of these accident sequences, the initiating event, postulated causes, and the sequence of subsequent events that could credibly lead to criticality were described in one table. Another table described in detail the two preventive barriers (each consisting of one or more IROFS) constituting the two legs of double contingency.

During the various in-office reviews, the staff performed a vertical slice from the PrHA to the NCSEs to the ISA Summary. The PrHA is first performed as part of the ISA process, and events that could credibly lead to criticality are carried forward into the NCSE. Those scenarios that can credibly lead to criticality are summarized in the controlled parameter discussion or in accident sequence tables in the ISA Summary. The staff performed a detailed review of selected sequences and determined that only those PrHA events that were not screened out as incredible or not leading to a critical configuration were treated further in the NCSE and carried forward into the NCSE. Those that were screened out were discussed in summary fashion in NCSE tables. In many cases, several individual PrHA events involving the same parameter were grouped together as a single bounding accident sequence, which constituted one of the "general types of accident sequences" discussed above. In all cases, the staff determined that the description of the process, the hazards, and the general types of accident sequences was appropriate.

The regulation at 10 CFR 70.65(b)(4) requires information demonstrating the applicant's compliance with the performance requirements of 10 CFR 70.61, including a description of management measures, criticality monitoring, and alarms, as required in 10 CFR 70.24, and, if applicable, the requirements in 10 CFR 70.64, "Requirements for New Facilities or New Processes at Existing Facilities." The aforementioned accident sequence tables in the ISA Summary contain information demonstrating that all credible accident sequences leading to criticality are highly unlikely. The information in these tables includes a brief discussion as to why the event is highly unlikely, including categorization as either a Category A, B, or C event. The discussion of Category A events states that the IROFS will be subject to periodic failure detection. For Category B events, the discussion states that there is sufficient margin so that at least three failures are required before criticality is possible. The discussion for Category C events is more varied, but must show "comparable assurance" to that provided by failure detection and margin. Because this is a new facility handling more than a critical mass of fissionable material, the requirements of 10 CFR 70.64 also apply. The design provides for criticality safety by the use of controlled parameters and IROFS. Double contingency is ensured by having two preventive control barriers, which are independent and unlikely to fail. The ISA tables in Section 5.3.7 justify their likelihood based on "consideration of all applicable 'available and reliable' qualities per NUREG-1718." The NCSEs expand upon these discussions, which describe the monthly failure detection, calculational basis for the safety margin, and other considerations relied on to place the event in one of the three highly unlikely categories. The staff observed that most of the events fell into Category A or Category C.

The staff reviewed the acceptability of the applicant's highly unlikely determination by independently confirming whether a selected sample of accident sequences exhibited adequate failure detection, margin, or other valid justification that provided comparable assurance to failure detection or margin. The staff reviewed the applicant's documented justification and technical references (especially calculations for the margin justification) to determine whether adequate controls and management measures had been provided to ensure that accident sequences were highly unlikely. As an additional check, the staff independently assessed the selected accident sequences against the criteria in Appendix A to NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," to determine whether the factors credited in the highly unlikely demonstration would have produced a numerical likelihood index corresponding to "highly unlikely."

Category A events were mainly those that relied on active engineered or administrative controls. Those Category A events relying on active engineered controls were generally justified by requiring at least monthly failure detection on the active components. The applicant based its justification of a majority of the Category A events relying on administrative controls on supervisory oversight of required operator actions. The applicant also justified some events based on the "continuous monitoring" of plant operations by control room personnel (which the staff observed directly at the reference facilities). This approach is consistent with the index scoring technique of NUREG-1718, Appendix A. Using this technique, two active or administrative IROFS combined with at least monthly failure detection (providing a duration index in NUREG-1718 of -1) would generally produce a numerical likelihood index corresponding to "highly unlikely."

[redacted]

Based on the above sampling of Category A criticality accident sequences, the staff has reasonable assurance that these accident sequences meet the requirement of 10 CFR 70.61(b).

The applicant justified a majority of Category C events by establishing at least two passive engineered controls, each of which was "more than unlikely" to fail. This approach is consistent with the index scoring technique of NUREG-1718, Appendix A. Using this technique, the combination of two passive IROFS would generally produce a numerical likelihood index corresponding to "highly unlikely." The staff also reviewed other examples of Category C events during the in-office licensing reviews and determined that they provided assurance comparable to monthly failure detection or safety margin. (The applicant stated that it is reducing the number of such events by recategorizing them as either Category A or B, as its likelihood arguments become more refined.)

[redacted]

Based on the above sampling of Category C criticality accident sequences, the staff has reasonable assurance that these accident sequences meet the requirement of 10 CFR 70.61(b).

There were also a small number of Category B events. These events were mainly those occurring in operations relying on mass or moderator control, in which there was a large margin of safety in the controlled parameter. The applicant justified a majority of Category B events on the basis of calculations demonstrating that many successive failures of either mass or moderation control (or sometimes both) would be needed before criticality would be possible. This approach is consistent with the index-scoring technique of Appendix A to NUREG-1718. Using this technique, the occurrence of three independent administrative failures (the most likely type and the type generally involved in mass or moderator control) would generally produce a numerical likelihood index corresponding to “highly unlikely.”

[redacted]

Based on the above sampling of Category B criticality accident sequences, the staff has reasonable assurance that these accident sequences meet the requirement of 10 CFR 70.61(b).

The IROFs tables in the NCSEs describe the specific management measures, codes and standards, and available and reliable qualities (e.g., redundancy, diversity, safety margin, failure detection, and surveillance) relied on to make the double contingency demonstration. Specific details about implementation of these management measures (e.g., surveillance frequencies, maintenance, and functional test procedures) are to be determined. The staff will verify the determination of specific management measures to be applied to ensure the reliability and availability of IROFS as part of the verification of the PSSCs. The staff determined, during the in-office licensing reviews, that the information in the NCSE IROFS tables was sufficient to demonstrate that (1) criticality would be highly unlikely, (2) all nuclear processes would be subcritical under both normal and credible abnormal conditions, and (3) double contingency would be met, provided that suitable management measures are established. The applicant stated, in its response to RAI NCS-71 (NRC, 2009c), that it did not expect active engineered instrumentation to be adversely affected by environmental conditions, such as vibration, humidity, high temperature, and radiation. However, it will environmentally qualify such electronic instrumentation in accordance with applicable industry standards. The staff will verify implementation of environmental qualification to ensure that the reliability and availability of IROFS will be verified as part of the verification of the PSSCs.

With regard to the use of a CAAS, the staff reviewed the adequacy of the alarm system as part of the LA review. The staff determined that the applicant's license commitments were sufficient to meet the requirements of 10 CFR 70.24, as discussed in Section 6.1.3.3 of this SER.

The provisions of 10 CFR 70.65(b)(5) require a description of the team, qualifications, and methods used to perform the ISA. As stated above, Chapter 5 of this SER discusses the aspects of the ISA methodology applicable to all safety disciplines. Aspects specific to criticality safety are discussed above. The ISA team included qualified criticality safety personnel, as appropriate, to ensure adequate treatment of criticality hazards, as mentioned in Section 5.2 of the ISA Summary.

As required by 10 CFR 70.65(b)(6), the applicant must provide a list briefly describing each IROFS in sufficient detail to understand its function in relation to the performance requirements of 10 CFR 70.61. The accident sequence tables in Section 5.3.7 of the ISA Summary briefly describe each IROFS in terms of its basic safety function (e.g., to limit a certain parameter). Tables 5.3.7-105 and 5.3.7-106 provide a more detailed description of engineered and administrative IROFS, respectively. In addition, the NCSEs include tables for each IROFS that describe the safety function, individual components, management measures, and other relevant features (e.g., available and reliable qualities) in greater detail. [redacted] While the level of IROFS description is at the functional rather than the component level in the ISA Summary, the individual components are identified in the IROFS tables in the NCSEs, the PrHAs, and other safety documentation such as P&IDs and drawings. The staff determined that this information is sufficient to understand how the IROFS meet the requirements of 10 CFR 70.61.

The regulation at 10 CFR 70.65(b)(7) requires a description of the proposed quantitative standards used to assess the consequences from acute chemical exposures. This is not applicable to the criticality safety review. See Chapter 8 of this SER for more information.

As required by 10 CFR 70.65(b)(8), the applicant must provide a descriptive list identifying all sole IROFS. While criticality sequences must comply with the double contingency principle, this does not preclude the use of sole IROFS so long as no single change in process conditions can lead to criticality. For all credible accident sequences leading to criticality that were included in the ISA Summary, the sequences rely on two preventive control barriers, which are independent and each are unlikely to fail. (Sequences that do not credibly lead to criticality or are inherently highly unlikely were not explicitly developed in the ISA Summary and therefore the applicant need not explicitly demonstrate that they comply with the double contingency principle. A sequence in which there is no credible failure that can lead to criticality is presumed to meet the double contingency principle without the need for specifying two separate control barriers.) The applicant provided a list of "sole IROFS" for criticality in its ISA Summary. The staff determined that when the applicant referred to a "sole IROFS," it meant a component shared between both preventive barriers for the same sequence, so that the barriers could not be considered totally independent. (Most of these are composed of active engineered controls, the various parts of which the applicant has chosen to call separate IROFS, instead of grouping them as a single IROFS.) The staff does not consider these components to be true instances of sole IROFS, because, in all cases, other components would have to fail before criticality is possible. There were no cases in which the "sole IROFS" was the only item protecting against a criticality. The staff performed a detailed review of sequences involving these items, not because they were labeled sole IROFS, but in order to obtain reasonable assurance that the sequences in question met the double contingency principle. The in-office review summaries referenced herein discuss these items in detail.

The provisions of 10 CFR 70.65(b)(9) require a description of the definitions of unlikely, highly unlikely, and credible as used in the ISA. The staff reviewed and approved the applicant's use of these terms during its review of the CAR (see Sections 5.1.5 and 6.1.4.2 of NUREG-1821). Only the definitions of "highly unlikely" and "credible" are relevant to the criticality review. The CAR defined "highly unlikely" as "events originally classified as not unlikely or unlikely to which sufficient principal SSCs are applied to further reduce their likelihood to an acceptable level." The staff determined that satisfying the above deterministic criteria constitutes an acceptable working definition of a sequence that will be highly unlikely. The CAR defined "not credible" as "natural phenomena or external man-made events with extremely low initiating frequency and process events that are not possible." The staff reviewed the applicant's highly unlikely classification of events, and its screening out of events as not credible or inherently highly unlikely, during the in-office ISA reviews, as documented in the in-office review summaries and summarized in the following section of this SER. In all cases, the staff determined that the applicant had correctly categorized or screened the events that it reviewed. (There was one exception—the applicant erroneously categorized some sequences in the grinding unit (PRE/PRF) NCSE. However, this appeared to be a typographical error that the applicant stated it would correct in the next revision of the NCSE which will be confirmed during the staff's verification of PSSCs.)

The preceding discussion addresses the required ISA Summary contents listed in 10 CFR 70.65 both programmatically and plantwide. The staff finds that the applicant's safety program, including the NCS Program, and its ISA methodology are sufficient to meet the requirements of 10 CFR Part 70 with regard to the ISA. The following section addresses the specific implementation of these requirements in individual facility processes.

6.2.2 Technical Aspects of Review

The staff reviewed the applicant's implementation of its ISA methodology for several of the most risk-significant process units in the aqueous polishing process, the MOX process, and the auxiliary systems. The staff documented these reviews in the in-office review meeting summaries and reference plant sites visit summary. The following summary provides an overview of the criticality safety control strategy and major issues discussed for each of the main process units reviewed. Because of the very large number of criticality accident sequences in the MFFF, the SER does not list the specific accident sequences reviewed; a more complete discussion can be found in the meeting summaries referenced herein. It should be noted that the applicant has analyzed ISA Summary events, which are discussed in this section of the SER, from a perspective of determining those that are highly unlikely (and meet the double contingency principle) from a criticality perspective. The applicant analyzed some aqueous polishing events to determine whether they are highly unlikely from a chemical release or radiological dose perspective. Chapters 8 and 11 of this SER discuss events not related to the evaluation of criticality. The following summary also discusses those aspects of the facility that the staff will verify as part of the required verification of the PSSCs.

6.2.2.1 Aqueous Polishing Process

The aqueous polishing process receives PuO_2 powder (consisting of both PDCF and AFS material); rids it of impurities, such as gallium and americium, through a solvent extraction process; precipitates it to plutonium oxalate; and reconverts it to purified PuO_2 for fuel fabrication in the MOX process. Most of the steps involved in this procedure consist of wet chemistry processes, in which the criticality control strategy relies primarily on maintaining plutonium solutions within geometrically favorable process equipment (e.g., columns, tanks, pipes). (Section 11.2 of this SER provides additional discussion of the aqueous polishing process.) [redacted] The staff therefore focused its review on those operational events that can cause either a loss of geometry or a loss of physicochemical control.

[redacted]

[redacted]

[redacted]

[redacted] The applicant will conduct settling studies to determine detailed requirements for representative sampling, and laboratory analysis will be subject to strict quality assurance, including redundancy and instrument calibration. The staff will verify the development of a sampling plan to ensure that IROFS samples will be reliable and will be sufficiently independent to meet the double contingency principle as part of the verification of the PSSCs. Solution transfers also are enabled through the use of dual hand switches and turn keys in the control room, which are kept under the control of operations supervisors. These will be locked out and transfers will not be authorized until automatically analyzed IROFS samples have been determined to be within acceptable limits by the computer control system (MMIS). The exact features of the control room design to ensure that solution transfer controls—and other process controls credited for safety—will be sufficiently independent is yet to be completed. The staff will verify human factors considerations (Chapter 12 of the SER) to ensure that control room operations will comply with the double contingency principle as part of the verification of the PSSCs. The staff reviewed the approach to IROFS sampling and solution transfer controls (as outlined in the in-office review summary dated August 14–16, 2007 (NRC, 2010a)) and has reasonable assurance that the applicant will ensure double contingency protection for this type of scenario.

[redacted]

The applicant stated that radiation detectors cannot, in general, accurately measure fissile isotope concentrations, but can only indicate a relative change in process conditions that could pose a potential problem. For this reason, these devices are not credited as IROFS. While not IROFS, they do provide added defense in depth against a loss of geometry control, which supports the applicant's determination that criticality is highly unlikely.

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Besides plutonium polymerization, other chemical transformations can change the form of the material. Precipitation is not a concern in most of the process, because wherever optimum moderation is assumed, any precipitation can only lead to an undermoderated condition. The staff also reviewed and agreed with calculations demonstrating that a pure plutonyl nitrate solution bounds worst-case plutonium-water-organic mixtures, and possible formation of a "third phase" (as discussed in the summary of the meeting held July 10–12, 2007 (NRC, 2010a)).

For normally dry portions of the aqueous polishing process (before PuO_2 dissolution or after plutonium oxalate precipitation), the material is handled in individual batches subject to mass and moderator control. Mass control is ensured throughout by the use of IROFS scales that either ensures that can contents are within analyzed limits or that the difference between two successive weight measurements (indicating a potential spill) is within analyzed limits. The cans are typically weighed either upon opening or after operations such as milling, transfer of the material to another container, or batching to the electrolyzer. Scale measurements are also compared to each other so as to provide a continuous functional check, such as before and after movement into an enclosure or area, when the cans are not opened (so that any mass difference is indicative of a measurement error or equipment malfunction). Moderator levels in the PuO_2 cans at the start of the aqueous polishing process are ensured upstream in the MOX process. Moderator levels in the powder leaving the aqueous polishing process are ensured by controlling the temperature and residence time (screw speed) in the KCA calcining furnace.

The Staff reviewers also considered other controlled parameters, including powder density, reflection, interaction, and use of neutron absorbers. The applicant has stated that the powder density values chosen (e.g., powder batched to the electrolyzers must be less than 7 g/cc) are generally considered to be conservative, but they will be verified during startup testing. The powder density of oxide from the PDCF will be verified based on vendor (DOE) records and subsequently confirmed, while the density of AFS oxide will be verified by methods such as nondestructive assay and weight measurements. The staff examined historical density records during its onsite review at the MELOX facility and determined that the assumed values were conservative. The staff will follow up on the completion of methods for independently verifying the acceptability of incoming feed material during the verification of the PSSCs, as discussed in Section 6.1.3.2 of this SER.

Most glovebox and process cell calculations assume nominal (1-inch) tight-fitting reflection conditions. The staff determined that this approach was acceptable given the lack of transient reflectors (e.g., personnel) because of engineered barriers in these areas and the use of drip trays to detect significant accumulations of liquid and limitations on sources of liquids within these areas.

Interaction between units was generally considered in facility calculation documents and NCSEs. The staff observed that significant numbers of "small" or "ancillary equipment," such as pipes, safe volume demisters, and pots, were not generally included in calculations. The staff reviewed facility plan and elevation drawings, as well as information from the applicant's three-dimensional CAD system, and concurs that this equipment, even if filled with optimal solution, will not have a significant effect on the reactivity of the system. The staff reviewed several sensitivity calculations that support this conclusion during its in-office reviews (see summary of meeting held December 15–16, 2009 (NRC, 2010a)) and found them acceptable.

Finally, neutron absorbers relied on for criticality safety will be procured and qualified in accordance with the facility's quality assurance program. The staff reviewed several procurement specifications for items, such as borated concrete and tanks, which incorporate fixed neutron absorbers and determined that they were adequate.

The aqueous process units reviewed included decanning (KDA) and milling (KDM); dissolution (KDB) and dechlorination/dissolution (KDD); purification (KPA); solvent recovery (KPB); homogenization, filling, and sampling (KCB); PuO_2 canning (KCC); oxalic mother liquor recovery (KCD); acid recovery (KPC); offgas treatment (KWG); stripped uranium and high alpha waste (KWD); waste organic solvent (KWS); and oxalic precipitation (KCA). The staff selected representative accident sequences and IROFS for further review and determined that they were adequate to meet both the performance requirements and the double contingency principle.

6.2.2.2 MOX Process

The facility receives PuO_2 powder in 3013 containers from the PDCF and stores it until it can be processed in the aqueous polishing process. Purified PuO_2 from the KCA is stored in the buffer storage unit (DCE) before being batched into primary dosing (NDP), where it is blended with depleted UO_2 to form primary blend with no more than 22 wt% Pu/(U+Pu). The powder is downblended further as final blend with no more than 6 wt% Pu/(U+Pu) and is then milled and ultimately homogenized. Throughout this portion of the process, the powder is transferred in large J80 or J60 powder "jars." The powder is then pressed into pellets, sintered to a hard ceramic, ground to the proper diameter, inspected, and assembled first into rods and then into finished fuel assemblies. Additional units handle off-specification pellets, powder that must be recycled into primary dosing, and any waste generated.

[redacted] The supplier (DOE) must certify the feed material to have less than 0.5 wt% moisture content and be within assumed isotopic limits (i.e., plutonium and uranium isotopic abundances and maximum ratio of uranium to plutonium). The applicant's criticality calculations assume bounding isotopic values throughout both the aqueous polishing and the MOX processes. The vendor must comply with bounding assumptions about moisture content, isotopic composition, and density of the feedstock; the applicant will audit the vendor's quality assurance program to ensure this. (The staff will follow up on the applicant's audit methods during the verification of the PSSCs, as discussed in Section 6.1.3.2 of this SER. In addition, the applicant has stated that it will perform nondestructive assay, such as calorimetry and gamma spectroscopy, on a sampling basis to confirm the mass and isotopic content). This is not specifically credited for meeting the performance requirements and the double contingency principle, but these actions will provide defense in depth.) Compliance with the structural integrity requirements of DOE-STD-3013-2004 (Stabiilzation, Packaging, and Storage of Plutonium-Bearing Materials) (DOE, 2004) will ensure that these limits are maintained through storage and handling until the interior cans are opened in the KDA during the aqueous polishing process, so that assumptions made in the applicant's criticality evaluations will remain valid.

Limits on the moisture content of purified PuO_2 reintroduced from the aqueous polishing process are ensured by controls on temperature and residence time (screw speed) in the KCA furnace. Subsequent to this (until pelletizing and sintering), the moisture content of the powder is limited by handling the material within moderation controlled gloveboxes. These gloveboxes rely on such standard industry strategies as taking credit for the structural integrity of the gloveboxes, limiting the quantity of moderating materials used within the glovebox (such as organic additives and equipment lubricants), restricting the use of extraneous moderators, and prohibiting solution-bearing process piping. The staff reviewed these approaches to implementing moderator control and found them to be consistent with applicable guidance (e.g., ANSI/ANS-8.22-1997) and standard industry practice. The applicant stated that it would administratively control moderators through development of a Moderator Control Program, which would include moderator exclusion training as an important component. This program is yet to be developed. The staff will confirm the applicant's development of the Moderator Control Program, as well as the associated moderator exclusion training, as part of the verification of the PSSCs.

The criticality analyses conservatively assume that the powder can absorb up to 3 wt% moisture from humidity in the air, plus another 2 percent to account for organic additives introduced during blending. The staff reviewed historical records from the reference facilities concerning moisture content and determined that the values assumed in the criticality analysis bounded the worst-case historical data. For AFS material, the impurity level must be limited to less than 30 wt% to ensure that the above assumed uptake from moist air will remain bounding (as chemical impurities can change physical properties of the material). Extraneous sources of moderator are strictly limited in moderation controlled areas. Total lubrication inside gloveboxes is limited to less than 1 kilogram of water-equivalent moisture (and any single component with more than 100 grams of lubricant must use nonhydrogenous materials). The calculation deriving this limit was conservative in that it assumed that all of the fissile material in the glovebox was arranged in a spherical configuration and fully reflected by water. [redacted] The applicant has yet to complete calculations that will definitively demonstrate the bounding nature of the 1,000-gram moderation limit over all credible arrangements of individual 100-gram items. The staff will confirm the applicant's completion of criticality calculations to support operation of the MFFF as part of the verification of the PSSCs, as discussed in Section 6.1.3.4 of this SER. The applicant has also stated that its estimates of quantities of lubricants to be used in moderation controlled areas were preliminary and were based on representative vendors and equipment. The staff will verify the applicant's implementation of moderation controlled areas as part of the verification of the PSSCs. Other sources of moderator will be excluded by facility design (e.g., exclusion of liquid-bearing pipes, sill height limitations, floor drains), and by prohibiting the use of moderating (hydrogenous) fire suppression agents.

Following pelletizing, moderation control is much less important, because the moderator cannot readily be intimately mixed with the fuel at that point. Upon exit from the sintering furnace, the residence time and pellet diameter are recorded to ensure that pellets have been adequately sintered (driving off almost all of their internal moisture). The sintered pellets must then pass through a "mechanical filter" before passing the grinding wheel in units PRE and PRF. If the diameter of the pellet is too large, it will not successfully pass through the aperture of the mechanical filter. Sintered and unsintered "green" pellets are segregated by means of barcode readers and an observation station where the furnace residence time and diameter records are checked before they are placed in storage. The segregation of sintered and unsintered pellets is not significant for pellet storage, but is credited for ensuring that pellets still containing significant moisture are not scrapped and subsequently recycled as primary blend, which can lead to exceeding moderation limits in the recycled powder or new pellets. The staff examined all pathways by which areas analyzed subcritical for sintered pellets only could receive unsintered pellets by mistake to ensure that such a scenario will be highly unlikely and found the pathways acceptable.

Mass control is ensured primarily through the use of IROFS scales to limit the quantity of oxide in individual containers and by tracking the number and identity of such containers entering or leaving an enclosure with barcode readers and system programmable logic controllers (SPLCs). Scales are generally used to measure the weight of containers before and after filling to ensure that they comply with mass limits and to detect any possible spills. The mass limits were derived from conservative calculations that (1) assume all of the material is in a spherical configuration, (2) the bounding moderation is present in the most reactive heterogeneous arrangement, as described above, and (3) the powder spheres are fully reflected by water. The amount of material transported even in the large J60 or J80 jars is significantly less than the maximum safe mass thus determined. In general, multiple failures of either mass or moderator (or both) would be needed before criticality is possible. Most of these scenarios have sufficient margin to be included in the Category B criterion for being highly unlikely.

[redacted]

The staff reviewed reference facility records and experimental data associated with the powder density and determined that the assumed density values for PuO_2 and MOX powders are conservative (NRC, 2008c). Besides using bounding density values based on historical data and experiments, the applicant has stated that it will verify powder density values during startup testing, and will measure density at key points in the process thereafter. [redacted]

The applicant stated in response to RAI NCS-90 (NRC, 2009c) that it will perform calculations showing that the MOX process will be subcritical, even if full density (4.6 g/cc) scrap is introduced. The staff will verify completion of criticality calculations to support operation of the MFFF as part of the verification of the PSSCs, as discussed in Section 6.1.3.4 of this SER.

The staff determined that the applicant's approach to controlling all these parameters is consistent with standard industry practice, complies with the double contingency principle, and is an acceptable approach to meeting the regulations of 10 CFR Part 70.

The MOX process units reviewed included PuO_2 receiving (DCP), PuO_2 3013 storage (DCM), PuO_2 buffer storage (DCE), can receiving and emptying (DCE), scrap processing (NCR), powder auxiliary unit (NXR), jar storage and handling (NTM), pellet storage (PPJ), the sintering furnaces (PFE/PFF), grinding (PRE/PRF), pellet handling (PML), rod storage (STK), rod tray handling (SMK), filter dismantling (VDR), fuel assembly handling and storage (TAS), acid recovery (KPC), stripped uranium and high alpha waste (KWD), waste organic solvent (KWS), offgas treatment (KWG), oxalic mother liquor recovery (KCD), and primary dosing (NDP). The staff selected representative accident sequences and IROFS for further review and determined that they adequately met both the performance requirements and the double contingency principle.

6.2.2.3 Auxiliary Systems

Auxiliary systems are those that are not directly involved with the fuel manufacturing process but are needed to support either the aqueous polishing or the MOX process. They may or may not contain fissile material under normal conditions. The auxiliary systems reviewed included the MOX process ventilation, pneumatic transfer, and container hand-carry and nonautomated transfer (lumped together as the RCA unit); automatic sampling unit (KPG); laboratory liquid waste unit (LGF); laboratory units (LLJ, KCA/B/C); and laboratory test line unit (LCT). Also reviewed were the waste handling processes: the waste storage unit (VDQ) and the waste nuclear counting unit (VDT).[5]

The MOX process ventilation does not normally contain fissile material; mass control is instituted by means of multiple redundant high-efficiency particulate air filters. These are favorable geometry filters, equipped with differential pressure gauges and subject to periodic surveillance. These controls and management measures ensure that failure leading to a significant accumulation of process powder in ventilation ductwork will be highly unlikely. The ductwork itself is mostly favorable geometry, though not specifically credited. Besides mass, the ventilation system credits moderation control; processes connected to the ventilation system are within moderation controlled gloveboxes. These gloveboxes are under “dynamic confinement,” meaning that they are under negative pressure and have a dry atmosphere, produced by a constant flow of dry air or nitrogen, to limit the moisture available for condensation. However, some wet offgas may be carried over into process ventilation from the furnaces. To address this possibility, the applicant performed an analysis to show that ventilation piping surfaces will remain above the dewpoint, thereby preventing condensation. The applicant has since stated, in response to RAI NCS-90NCS-94, that it will revise the applicable NCSE to modify the technical justification as to why the ventilation system will not contain liquid water. The staff will confirm the applicant’s implementation of moderation control for the process ventilation system as part of the verification of the PSSCs.

The pneumatic transfer system mostly consists of favorable geometry piping, in which fissile material is transferred in sealed, mass-controlled containers. The container and the transfer shuttle constitute two layers of containment. The staff reviewed the criticality safety basis for the pneumatic transfer system and determined that a very large number of spills would be required to constitute a criticality concern. A small accumulation over time would eventually result in a mass imbalance in one of the destination gloveboxes, which would be detected. Therefore, the risk of a criticality in the pneumatic transfer system is highly unlikely.

Occasionally, fissile-bearing items may have to be hand-carried from one process area to another; the reasons for doing so may include the replacement of defective contaminated equipment or the manual transfer of unusual scrap or waste material. Items to be transported must be contained inside sealed, watertight containers, two of which must have been shown to be subcritical when touching and fully surrounded by water. In addition, containers must be separated by 12 inches from all other fissile-bearing items and the concrete floors, ceiling, and walls. The applicant will comply with the mass limits for removed contaminated equipment by one of two means: either by assuming that the entire volume is filled with fissile material or by radiation scanning. The staff will verify calibration and demonstration of the fissile mass measurements based on IROFS radiation detectors as part of the verification of the PSSCs, as discussed in Section 6.2.2.1 of this SER.

With regard to the basis for the 12-inch separation distance, MOX Services provided an analysis based on the use of the solid angle formula (Knief, 2000). Solid angles were based on spherical masses, and individual unit unreflected k_{eff} values were determined by using the reactivity formula (Lamarsh, 2001). Results showed that the solid angle subtended by the spheres was less than 1 steradian, which is the allowable limit for individual units with unreflected k_{eff} less than 0.8. During an onsite review (see summary of meeting held December 16–17, 2008 (NRC,

[5] While the applicant grouped these units in with the MOX process units, waste processes are not tied directly to the main fuel manufacturing process, so the staff will discuss them along with the other auxiliary units.

2010a)), the applicant showed the staff new calculations that confirmed the results of the solid angle method.

The waste processing units reviewed consisted of the waste storage unit (VDQ) and the waste nuclear counting unit (VDT). The waste drums in these units are expected to consist of low-level contaminated industrial waste, but the staff chose these two units for review because they rely heavily on simple administrative controls unlike other areas in the MFFF. While drums are expected to contain only small quantities of fissile material, they were modeled with very conservative assumptions, including the following: (1) each drum was filled up to the maximum allowable mass with full-density PuO_2, (2) the fissile material was arranged in a spherical configuration within the drum, and (3) the fissile material was moderated by water or polyethylene. The drums are modeled in an infinite planar array, neglecting the steel bodies of the drums so that they may be stored in any arrangement, so long as they are not stacked (except as specifically analyzed safe in the storage racks). The nature of these conservative assumptions means that, even if a very large number of drums were stacked together, or a large number of drums exceeded their mass limits, criticality will still be very unlikely (see summary of meeting held December 15–16, 2009 (NRC, 2010a)). [redacted]

The staff determined that exceeding the mass limit in a large number of drums is highly unlikely because of the margin resulting from the nature of waste streams involved and because the drums must have their mass confirmed by two different operators (and one of the determinations must be made using either the precounter or neutron counter), with the exception of used process filters, whose mass is estimated using differential pressure measurements correlated to the mass. The applicant classified these events as Category C highly unlikely events. The applicant stated in response to RAI NCS-93 (NRC, 2009c) that it will use a combination of passive neutron counting, active neutron interrogation, and gamma spectroscopy to determine plutonium masses in drums. The staff will verify the applicant's methods for determining the contents of fissile waste drums as part of the verification of the PSSCs.

[redacted]

In the VDT unit, the main concern involves the accuracy of the drum counting process (for subcriticality of drum storage), since there are many less drums in this unit than in the VDQ. The staff reviewed the scenarios by which an erroneous mass determination could lead to criticality concerns in downstream units and determined that they adequately met the double contingency principle. (Note that drum assay is initially estimated upstream in the VDQ and VDU, with the purpose of the VDT being to obtain a more accurate assay.) Based on this review, the staff determined that there was reasonable assurance of safety in the handling and storage of waste drums in the MFFF. The staff determined that the auxiliary units reviewed adequately met both the performance requirements and the double contingency principle.

6.3 Evaluation Findings

The staff reviewed the description of the applicant's NCS Program contained in Chapter 6 of the LA. Based on the foregoing review, the staff determined that there is reasonable assurance that the applicant will establish and maintain a program sufficient to ensure health and safety and compliance with all criticality safety regulatory requirements. In particular, the staff has reasonable assurance that the applicant will (1) have in place a staff qualified to develop, implement, and maintain an NCS Program in accordance with the application's description of facility organization, administration, and management measures; (2) conduct its operations based on technical practices sufficient to ensure that licensed material will be possessed, stored, and used safely according to the requirements of 10 CFR Part 70; (3) develop, implement, and maintain a CAAS in accordance with the requirements of 10 CFR 70.24; and (4) establish safety limits and controls sufficient to ensure subcriticality, including an appropriate margin of subcriticality for safety, and the baseline design criteria of 10 CFR 70.64. Based on this review, the staff has reasonable assurance that the applicant's NCS Program will meet the requirements for a license to possess and use SNM under 10 CFR Part 70 and will ensure protection of public health and safety, including workers and the environment. The staff will confirm this conclusion during the verification of the PSSCs, as required by 10 CFR 70.23(a)(8) and discussed in Chapter 1 of this SER.

The staff also reviewed selected portions of the applicant's ISA Summary and supporting onsite ISA documents. Based on the foregoing review, the staff determined that there is reasonable assurance that the applicant will implement and maintain safety limits and controls sufficient to ensure health and safety and compliance with all criticality safety regulatory requirements. In particular, the staff has reasonable assurance that the applicant will establish controls on all credible accident sequences leading to criticality sufficient to ensure that (1) credible accident sequences will be highly unlikely, (2) all processes will be subcritical under normal and credible abnormal conditions, and (3) all processes will adhere to the double contingency principle. Based on this review, the staff has reasonable assurance that the applicant's implementation of its ISA will meet the applicable requirements of 10 CFR 70.66(a) and will ensure protection of public health and safety, including workers and the environment. The staff will verify this conclusion during the verification of the PSSCs, as required by 10 CFR 70.23(a)(8) and discussed in Chapter 1 of this SER.

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7.0 FIRE PROTECTION

This chapter of the safety evaluation report (SER) contains the staff's review of fire protection described by the applicant in Chapter 7 of the license application (LA) (MOX, 2010a). The objective of this review is to verify whether the applicant's commitments and goals related to fire protection are adequate to meet or exceed the regulatory acceptance criteria referenced below. The review is necessary to verify that the Mixed Oxide Fuel Fabrication Facility (MFFF) is adequately protected against external and internal fires and that the items relied on for safety (IROFS) identified by the applicant adequately protect against natural phenomena and the consequences of potential accidents.

The staff evaluated the information provided by the applicant for fire protection by reviewing Chapter 7 of the LA, other sections of the LA, the Integrated Safety Analysis (ISA) Summary (MOX, 2010b), and supplementary information provided by the applicant. The staff coordinated its review of fire protection with the review of explosion protection aspects (see Chapter 8 of this SER), and the review of other plant systems (see Chapter 11 of this SER).

7.1 Regulatory Requirements

The staff reviewed the fire protection information in the LA with respect to compliance with the following regulations:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 70.22(a)(7) states that the applicant must describe the equipment and facilities to protect health and minimize danger to life and property

- 10 CFR 70.23(a)(3) states that, for approval of an LA, the Commission must determine that the applicant's proposed equipment and facilities are adequate to protect health and minimize danger to life or property.

- 10 CFR 70.61, "Performance Requirements," states that the applicant shall evaluate in the ISA its compliance with the performance requirements.

- 10 CFR 70.64, "Requirements for New Facilities or New Processes at Existing Facilities," requires that the design of new facilities incorporate the baseline design criteria and defense-in-depth practices. With respect to fire protection, 10 CFR 70.64(a)(3) requires that the MFFF design "provide for adequate protection against fires."

7.2 Regulatory Acceptance Criteria

Sections 7.4.3 and 7.5 of NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility" (NRC, 2000), outlines the acceptance criteria for the U.S. Nuclear Regulatory Commission (NRC) review of the applicant's fire safety program and equipment. Sections of NUREG-1718 that are no longer applicable to the review Chapter 7 of this SER, based on the current design, are: Section 7.4.3.2.I concerning design of the ventilation system (the facility ventilation system is evaluated in Sections 11.3 and 7.3.2.6 of this SER), Section 7.4.3.2.O concerning hydrogen use and storage (Hydrogen gas is not used in the current design), Section 7.4.3.2.Z concerning flammable and combustible solvents (storage and handling of materials in the facility are discussed in Section

7.3.1.2 of this SER), Section 7.4.3.2.AA concerning inert gas purge and vent systems (the systems mentioned in the SRP are not in the current design), and Section 7.4.3.2.BB concerning incinerators, boilers, and furnace (the systems mentioned in the SRP section are not in the current design). In addition, Sections 7.4.3.3.A–E concerning requirements for a site fire brigade were not considered applicable because the Savannah River Site (SRS) fire department will perform all manual firefighting operations (see Section 7.3.3 of this SER).

7.3 Staff Review and Analysis

Sections 7.3.1 through 7.3.4 of this SER provide the staff's evaluation of how the applicant addressed the fire protection acceptance criteria in NUREG-1718. SER Section 7.3.5 is the staff's evaluation of the IROFS, and SER Section 7.3.6 gives the staff's evaluation of the applicant's event sequences for the regulatory requirements on fire safety.

7.3.1 Organization and Conduct of Operations

The organization and conduct of operations are the management measures that ensure that fire safety is administered appropriately at a licensed facility. Section 7.1 of the LA (MOX, 2010a) describes the applicant's commitment to ensure that IROFS as identified in the ISA Summary (MOX, 2010b) are available and reliable; fire protection organizational responsibilities are defined; transient ignition sources and combustibles are controlled; and the facility maintains a readiness to extinguish or limit the consequences of a fire. The applicant has developed a fire protection program with administrative controls in order to meet the organizational and operational guidance of NUREG-1718 (NRC, 2000). The fire protection program and associated administrative controls meet the regulatory acceptance criteria and fire protection baseline design criteria of 10 CFR 70.64(a)(3) and are, therefore, acceptable. The following sections provide more details on the applicant's plans for a fire protection program and administrative controls.

7.3.1.1 Fire Protection Program

In Chapter 7 of the LA, the applicant described the fire protection program developed for the MFFF. The program establishes defense-in-depth practices for IROFS and the procedures, equipment, and personnel required to implement the program. The program is designed to do the following:

- Prevent fires from starting.
- Detect fires rapidly and determine their location.
- Inform MFFF workers of fires.
- Inform the SRS Operations Center of fires.
- Control and limit the spread of fires.
- Promptly extinguish fires.
- Maintain safe egress paths for plant personnel in the event of fire.

- Protect IROFS when a fire is not promptly extinguished by the fire protection features and systems, so that neither an uncontrolled release of radioactive material or a hazardous chemical nor a criticality event occurs.

The fire protection program defines organizational responsibilities, lines of communication, and personnel qualification requirements. Specific management responsibilities are as follows:

- The manager of the plant has overall responsibility for formulation, implementation, effectiveness, and assessment of the MFFF fire protection program.

- The manager of the production function is responsible for implementing periodic inspections to minimize the amount of combustibles in areas with IROFS and for determining the effectiveness of housekeeping practices. The position is also responsible for ensuring the availability and acceptable condition of fire protection systems and equipment and fire stops and fire-rated penetration seals, and for ensuring that prompt and effective corrective actions are taken to remedy conditions adverse to fire protection.

- The manager of the maintenance function is responsible for periodic inspections and testing of fire protection systems and equipment in accordance with established procedures.

- The manager of the quality assurance function is responsible for ensuring the effective implementation of the quality-affecting aspects of the fire protection program by planned inspections and scheduled audits.

- The individual responsible for the fire protection function has at least 5 years of experience as a fire protection engineer. This position is responsible for reviews and evaluations of proposed work activities to identify potential transient fire loads and to periodically assess the effectiveness of the fire protection program, including through fire drills and training. The individual responsible for fire protection reports to the manager of the regulatory function.

- The manager of the training function is responsible for providing MFFF-specific training to the SRS fire department. The position is also responsible for implementing a program for training MFFF personnel in administrative procedures that implement the fire protection program and emergency procedures relative to fire protection.

Consistent with the guidance in Section 7.4.3.1.A of NUREG-1718 (NRC, 2000), the fire protection program establishes the policy for the protection of IROFS at the plant and the procedures, equipment, and personnel required to implement the program at the plant site and is, therefore, acceptable.

7.3.1.2 Administrative Controls

Administrative controls establish procedures for the following:

- fire prevention
- surveillance procedures
- control of flammable and combustible materials

- control of ignition sources
- testing, inspection, and maintenance
- impairments
- fire response planning
- prefire plans

Fire prevention includes controls on operational activities, design features such as spark-resistant electrical components, and restrictions on the use of combustible materials.

Surveillance procedures include inspections of combustible loading, fire protection equipment and systems, general housekeeping, and transient combustibles.

Control of flammable and combustible materials includes the following:

- The applicant has limits on the bulk storage of combustible materials inside or adjacent to buildings or systems containing IROFS. This meets the guidance in Section 7.4.3.1.B.(i) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

- The applicant has controls on the storage and handling of ordinary combustible materials, combustible and flammable gases and liquids, combustible high-efficiency particulate air (HEPA) and charcoal filters, dry ion exchange resins, pyrophoric materials, and other combustible supplies in areas containing IROFS. Flammable and combustible liquids are handled in accordance with National Fire Protection Association (NFPA) 30, "Flammable and Combustible Liquids Code" (NFPA, 1996d). This meets the guidance in Section 7.4.3.1.B.(ii) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

- The applicant will ensure that the storage and handling of pyrophoric metals is in accordance with methods in the applicable codes and industry standards. Combustible loading in areas containing IROFS is in accordance with applicable guidance in NFPA 801, *Standard for Fire Protection for Facilities Handling Radioactive Materials*, 1998 edition (NFPA, 1998d). The applicant commits to storing, handling, and using flammable and combustible liquids in accordance with applicable sections of NFPA 30, *Flammable and Combustible Liquids Code*, 1996 edition (NFPA, 1996d). The applicant will store, handle, and use flammable and combustible gases in accordance with applicable portions of NFPA 50A, *Standard for Gaseous Hydrogen Systems at Consumer Sites*, 1999 edition (NFPA, 1999a) and NFPA 55, *Standard for the Storage, Use, and Handling of Compressed and Liquefied Gases in Portable Cylinders*, 1998 edition (NFPA, 1998c). Where appropriate, explosion prevention measures are implemented by the applicant in accordance with applicable sections of NFPA 69, *Standard on Explosion Prevention Systems,* (NFPA, 1997a). This meets the guidance in Section 7.4.3.1.B.(i) and (ii) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

- The applicant has procedures for handling transient fire loads, such as combustible and flammable liquids, wood and plastic products, or other combustible materials, in buildings containing IROFS during the phases of operation and especially during maintenance or modification activities. This meets the guidance in Section 7.4.3.1.B.(iii) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

- The applicant will ensure that the use of wood is permitted only when noncombustible products are not practical from a process consideration. Where used, wood is treated with a flame retardant. This meets the guidance in Section 7.4.3.1.B.(iii) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

- The applicant will ensure that transient combustible materials are unpacked outside of MFFF production areas as much as practical. When necessary, transient combustible packing materials may be unpacked inside MFFF production areas; however, the materials are removed from the area following unpacking. Loose combustible packing material, such as wood or paper excelsior or polyethylene sheeting, is placed in metal containers with tight-fitting, self-closing metal covers if the material remains in production areas. This meets the guidance in Section 7.4.3.1.B.(v) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

- The applicant will ensure that work-generated combustible waste is removed from buildings containing IROFS following completion of the activity or at the end of the shift, whichever comes first. This meets the guidance in Section 7.4.3.1.B.(v) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

The IROFS, combustible material loading controls (SER Section 7.3.5.1) and ignition source controls (SER Section 7.3.5.19), also ensure the control of flammable and combustible material. The applicant also provides for periodic housekeeping inspections. This meets the guidance in Section 7.4.3.1.B.(vii) and is, therefore, acceptable.

Ignition sources are controlled by design, such as the selection of appropriate electrical equipment in gloveboxes where combustible material is present and the absence of electrical equipment in process cells. The following national codes and standards are used for the selection of electrical equipment in gloveboxes to minimize the risk of electricity as an ignition source:

- NFPA 70, “National Electrical Code” (NFPA, 1999c)

- 29 CFR 1910, “Occupational Safety and Health Standards” (OSHA, 2004)

- Underwriters Laboratories (UL) 508, “Industrial Control Equipment” (UL, 1993)

- Institute of Electrical and Electronics Engineers (IEEE) 383, “Standard for Qualifying Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations” (IEEE, 1992)

Ignition sources are also controlled by work control procedures requiring the following:

- The applicant requires permits to control welding, grinding, flamecutting, brazing, or soldering operations; separate permits for each area where work is to be performed; and an allowable duration for the validity of permits. This meets the guidance in Section 7.4.3.1.B.(iv) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

- The applicant requires the conduct of welding and grinding in accordance with applicable portions of NFPA 51B, “Standard for Fire Prevention During Welding, Cutting, and Other Hot Work” (NFPA, 1999b). This meets the guidance in Section 7.4.3.1.B.(iv) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

- The applicant prohibits the use of open flames or combustion-generated smoke for leak testing. This meets the guidance in Section 7.4.3.1.B.(vi) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

- Smoking is restricted to designated areas outside of the MFFF buildings.

- Written procedures document testing, inspection, and maintenance, and the results and followup actions are recorded. Water-based MFFF fire protection systems and equipment are inspected, tested, and maintained in accordance with applicable portions of NFPA 25, "Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems" (NFPA, 1998a). Other MFFF fire protection systems are inspected, tested, and maintained in accordance with the applicable NFPA codes, manufacturer's guidelines, and operating experience. This meets the guidance in Section 7.4.3.2.S in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

- A test plan lists the responsible personnel positions in connection with routine tests and inspections of the fire detection and protection systems. The test plan contains the types, frequency, and identification of the testing procedures. This meets the guidance in Section 7.4.3.1.B.(ix) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

- A penetration seal tracking program records pertinent information regarding the installation and modification of fire-rated penetration seals that are IROFS and that are installed and maintained in accordance with UL 1479, "Fire Tests of Through Penetration Fire Stops" (UL, 1994). This meets the guidance in Section 7.4.3.1.B.(xiii) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

- Emergency lighting and communications systems are inspected, tested, and maintained in accordance with vendor recommendations. This meets the guidance in Section 7.4.3.1.B.(ix) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

- Onsite and offsite emergency communications systems are tested periodically in accordance with the site emergency preparedness program.

To achieve continuity in fire protection during periods when a fire protection system is impaired or being maintained, written procedures address impairment of MFFF fire protection systems. Disarming of the MFFF fire detection or fire suppression systems is controlled by a permit system that includes the following:

- identification and tracking of impaired equipment
- identification of personnel to be notified
- determination of needed compensatory fire protection and fire prevention measures

This meets the guidance in Section 7.4.3.1.B.(viii) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

If impairment of the protection system is planned, the necessary parts and personnel are assembled before removing the system from service.

Compensatory measures (e.g., fire watches, additional combustible controls) are implemented as appropriate in accordance with procedures when IROFS fire protection features and systems are not operable. This meets the guidance in Section 7.4.3.1.B.(viii) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

Testing and maintenance procedures for fire protection systems that are IROFS are contained in the management measures specified for those IROFS. Acceptable outage times are specified in work control procedures for fire protection system impairments. The applicant stated in the LA that exceeding the acceptable outage times for IROFS fire protection systems requires additional compensatory measures, including shutdown of processes in affected areas. This meets the guidance in Section 7.4.3.1.B.(ix) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

Procedures specify the actions to be taken by individuals discovering a fire, including guidance for notifying appropriate personnel, and means and methods that may be used by MFFF staff to extinguish a fire. This meets the guidance in Sections 7.4.3.1.B.(x) and (xi) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

The SRS fire department developed prefire plans. These plans define the strategies that are used at the MFFF for fighting fires in areas containing IROFS or that present hazards to IROFS. These prefire plans designate all the items specified in NUREG-1718, Section 7.4.3.1.B.(xii), items a–l and meet the guidance in Section A.4.8.1 of NFPA 801, “Standards for Facilities Handling Radioactive Material” (NFPA, 1998d), and are, therefore, acceptable.

Consistent with the guidance in Section 7.4.3.1.B of NUREG-1718 (NRC, 2000), the administrative controls program establishes the policy for the protection of IROFS at the plant and the procedures, equipment, and personnel required to implement the program at the plant site and is, therefore, acceptable.

7.3.2 Features and Systems

Plant fire protection features and systems include building construction, fire area determination, electrical installation, ventilation, detection and alarm, and suppression. Section 7.4 of NUREG-1718 (NRC, 2000) provides acceptance criteria for fire protection features and systems. The following provides an evaluation of how the applicant addressed these acceptance criteria at the MFFF site.

7.3.2.1 Construction

Buildings where radioactive materials are used, handled, or stored at the MFFF are of NFPA 220, “Standard on Types of Building Construction” (NFPA, 1995b), Type I or Type II construction (NRC, 2007a). Thus, the structural members, including walls, columns, beams, girders, trusses, arches, floors, and roofs, are of approved noncombustible or limited-combustible materials and will have fire resistance ratings as specified by NFPA 220 (NFPA, 1995b). Buildings that contain IROFS are Type I construction and have exterior bearing walls rated at least 3 hours and interior bearing walls, trusses, beams, girders and columns rated at least 2 hours. In addition, buildings are protected from exterior fires by observing the fire safety criteria recommended in NFPA 80A, “Recommended Practice for Protection of Buildings from Exterior Fire Exposures” (NFPA, 1996g).

As described in Section 1.1.2.1.3.1 of the LA (MOX, 2010a), the mixed oxide (MOX) fuel fabrication building (BMF) is a multistory, reinforced concrete structure. The BMF consists of reinforced concrete shear walls, floors, and a roof slab. [redacted] The staff finds that the preliminary construction features at the MFFF are adequate to meet the baseline design criteria of 10 CFR 70.64(a)(3) for fire safety and meet the guidance in Section 7.4.3.2.A–C in NUREG-1718 (NRC, 2000) and are, therefore, acceptable.

7.3.2.2 Interior Surface

Section 7.3.4.1 of the LA indicates that exposed interior walls or ceilings and any factory-installed facing material have a Factory Mutual Research Corporation-approved or UL-listed flame-spread rating of 25 or less, and a smoke-developed rating of 50 or less in accordance with American Society for Testing and Materials (ASTM) E84, "Standard Test Method for Surface Burning Characteristics of Building Materials" (ASTM, 1998). This meets the guidance in Section 7.4.3.2.D in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

Carpets and rugs are not currently planned for use in the BMF. If it is determined later that they will be installed, the carpets and rugs will be tested in accordance with NFPA 253, "Standard Method of Test for Critical Radiant Flux of Floor Covering Systems Using a Radiant Heat Energy Source" (NFPA, 2006). This meets the guidance in Section 7.4.3.2 E in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

Protective coatings for floors will be fire resistant with a flame-spread rating of 25 or less, and a smoke-developed rating of 50 or less in accordance with ASTM E84 (ASTM, 1998).

7.3.2.3 Storage Racks

Section 7.3.4.1 of the LA states that racks for the storage of plutonium oxide, uranium oxide, or MOX in powder, pellet, or rod form are noncombustible. The applicant provides combustible loading controls to prevent the buildup of combustibles in areas where storage racks are located. Section 7.2.3 of the LA discusses combustible loading controls. Limiting combustible materials in areas where special nuclear material is stored reduces the intensity of potential fires if they occur. This meets the guidance in Section 7.4.3.1.B.(xiii) in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

7.3.2.4 Electrical Considerations

Section 11.4 of the LA discussed the electrical systems at the MFFF. To prevent fires from initiating and adversely affecting critical systems, the electrical systems at the MFFF are designed with redundant Class 1E circuits and equipment and are located in Quality Level 1 structures. Within these structures, redundant Class 1E circuits and equipment are separated by Quality Level 1 structures, barriers, distance, or a combination, with the preference being separation by Quality Level 1 structures. IEEE 384, "Standard Criteria for Independence of Class 1E Equipment and Circuits" (IEEE, 1992b), is used as the basis for minimum separation distances except in gloveboxes, where separation is maintained to the extent practical.

- Electrical cable used in gloveboxes is IEEE-383 (IEEE, 1992a) qualified (i.e., ignition resistant and self-extinguishing).

- IROFS power cables from emergency buses are installed in conduit.

- The design of electrical distribution equipment within gloveboxes is in accordance with NFPA 70, "National Electric Code" (NFPA, 1999c).

These considerations also protect electrical systems from the effects of smoke and fire that initiate outside the electrical systems. Section 11.4 of this SER evaluates the electrical systems at the MFFF. The staff finds that electrical systems are robust and meet the performance requirements of 10 CFR 70.61. The applicant's electrical analyses provided reasonable assurance that the electrical IROFS protect against the consequences of potential accidents and natural phenomena. The applicant's electrical design meets the guidance in Section 7.4.3.2.G in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

7.3.2.5 Lightning Protection

Section 11.4 of the LA (MOX, 2010a) states that each MFFF building and structure has a grounding grid. The various grounding grids are interconnected to form the grounding grid system. The MFFF grounding system complies with the requirements of NFPA 70 (NFPA, 1999c) and certain other applicable grounding codes and standards. The portions of the MFFF grounding system that serve lightning protection functions also comply with the requirements of NFPA 780, "Lightning Protection Code" (NFPA, 1997d). This meets the guidance in Section 7.4.3.2.H in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

7.3.2.6 Ventilation System

Pressure gradients between the confinement zones ensure that leakage air flows from the zones of lowest contamination risk to zones of increasing contamination risk. During a fire, the main objective is to maintain differential pressure between the room of fire origin and the surrounding areas. Depending on whether gloveboxes or dispersible materials are present, the heating, ventilation, and air conditioning (HVAC) dampers in process rooms and process cells are operated to ensure that combustion products flow through the exhaust stacks of the gloveboxes, the process rooms, or the process cells.

The ductwork in the ventilation systems incorporates manual and automatic dampers and controls to distribute and regulate the movement of air. The ductwork is welded stainless steel or welded galvanized pipe. As discussed in Section 7.3.1 of the LA, closure devices with fire resistance ratings are provided where ventilation ductwork penetrates fire barriers. These devices have fire resistance ratings that are consistent with the designated fire resistance ratings of the fire barriers penetrated. The four different HVAC systems have the following fire damper configurations:

- In C2 ventilation areas (for example, process rooms containing rods or assemblies and corridors around C3 areas), automatic fire dampers are provided in the medium depressurization exhaust (MDE) system supply and exhaust ductwork.

- In process rooms and other C3 ventilation areas (process rooms) with dispersible radioactive material, the high depressurization exhaust (HDE) system exhaust fire dampers have manual controls (chain wheel operator or electric motor). The room supply fire dampers for these areas are automatic.

- For the process cell exhaust (POE) system, room exhaust fire dampers are manually operated (chain wheel operator). The room supply fire dampers for these areas are automatic.

- In C4 ventilation areas (gloveboxes), fire isolation valves in the very high depressurization exhaust (VHD) headers are manually controlled. This meets the guidance in Section 7.4.3.2.X in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

All chain wheel operators are accessible from the corridor outside of the affected process rooms. This meets the guidance in Section 7.4.3.2.J in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

The BMF building design incorporates airlocks that offer access to the process rooms. Fire barriers separate the airlocks from the process rooms and an independent ventilation system ventilates the airlocks. The airlocks maintain a negative pressure with respect to the areas adjacent to the process room, thereby reducing the spread of combustion products from the process room. In addition, the deliberate pressure cascade from the safe havens to the stairwells ensures that the smoke infiltration is minimized during a fire in the MOX processing or aqueous polishing (AP) areas. The force required to open all doors is within the limits of the guidance of NFPA 101, “Life Safety Code” (NFPA, 1997b).

SER Section 11.3 provides the staff’s evaluation of the MFFF HVAC and confinement systems.

7.3.2.6.1 Fire Detection in the Ventilation System

Section 7.3.2 of the LA states that smoke and heat detectors are located in the HVAC supply ventilation intake header. Heat detectors are provided upstream of the HVAC final filters. Smoke detectors are also installed in the ventilation exhaust ducts of the process cells, which are inaccessible during plant operation. Installation of smoke and heat detectors is in accordance with NFPA 72, “National Fire Alarm Code” (NFPA, 1996f). Where necessary, NFPA 72 testing and surveillance requirements will be satisfied through replacement.

7.3.2.6.2 Filter Design and Protection

At the BMF, HEPA filters are used to prevent the release of radioactive materials from the three dynamic confinement systems. HEPA filters meet the requirements of American Society of Mechanical Engineers (ASME) AG-1, “Code on Nuclear Air and Gas Treatment” (ASME, 2003). Air stream dilution and the use of spark arresters provide fire protection for the BMF final HEPA filter system. Roughing filters (spark arresters) and prefilters are located upstream of the HEPA filter exhaust plenums. Spark arresters prevent hot particles from impacting the final filters. The assemblies are designed and fabricated to the same temperature ratings as the duct materials in which they are installed.

7.3.2.6.3 Detection and Suppression Protection in the Ventilation Systems

Temperature detectors are provided in the ductwork upstream of each final filtration unit. Detectors alarm in the event of high temperatures in the ductwork.

Automatic suppression is not provided in the final filter plenums, although Appendix E to NUREG-1718 (NRC, 2000) recommends it. According to Section 11.3 of the LA, spark arresters and dilution of high temperature exhaust streams will provide fire protection to the final HEPA filter systems to prevent prolonged exposure to temperatures above 204 degrees Celsius (C) (400 degrees Fahrenheit (F)), the maximum filter service temperature. The applicant's analyses determined that mixed airflows to the filters would not exceed 204 degrees C (400 degrees F) under a maximum room temperature condition of 615 degrees C (1,140 degrees F). The applicant determined this maximum temperature condition assuming that the suppression system was inoperable. The staff reviewed the applicant's temperature calculations and found them to be acceptable (NRC, 2008a).

7.3.2.7 Means of Egress Protection

The facility layout complies with the 1997 version of NFPA 101 with exceptions. According to Section 7.6.2 of the LA, security door locks and barriers installed along the means of egress prevent the required free escape of occupants from inside the building as required by NFPA 101 (NFPA, 1997b). Security concerns for special nuclear material require that building occupants are not allowed free access to the outside of the facility, even during an emergency situation. This meets the guidance in Section 7.4.3.2.K in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

The means of egress are arranged and maintained to provide free and unobstructed lighted egress from all parts of the facility. Buildings at the MFFF are designed to provide means of egress that are adequate in number, location, and capacity for emergencies. In general, impediments to egress such as door locks are not installed. Corridors lead to strategically located 2-hour fire-rated exit stairs and exit passageways that lead to protected safe havens. Exit stairs are designed to prevent smoke infiltration during a fire.

Because of safeguards and security concerns, the BMF is equipped with five safe havens for emergency egress. Personnel leaving the BMF enter the safe havens and are not allowed to exit the safe haven buildings (BSH) until security forces monitor them. Sprinkler suppression systems, separate ventilation systems, and emergency lighting protect the safe havens. An outer security barrier with a minimum fire resistance rating of 3 hours structurally separates the safe havens from the BMF. The staff finds that the availability and protection of the means of egress will result in a safe path of escape during a fire.

Emergency lighting is provided for means of egress and for critical operations areas where manual operations must be performed during a power outage of normal alternating current power sources. Standby generators support the emergency egress lighting. There are two standby generators; each can operate continuously for 24 hours. The staff finds that the preliminary emergency lighting provisions provide the necessary illumination in the event power to normal lighting is interrupted.

7.3.2.8 Fire Areas and Barriers

For facility design and operational purposes, the BMF is subdivided into fire areas. The applicant used guidance from NFPA 801 (NFPA, 1998d) to determine fire area boundaries. Section 7.3.1 of the LA indicates that fire areas separate IROFS and areas that contain SNM, from areas that contain fire hazards. Noncombustible walls with a minimum 2-hour fire rating separate fire areas from other fire areas.

The function of fire barriers is to separate fire areas from one another and to confine fires to their area of origin. Fire barriers are constructed of noncombustible material and meet the criteria set forth by ASTM E119, “Standard Test Methods for Fire Test of Building Construction and Materials” (ASTM, 2000a). Three-hour and four-hour fire barriers separate some hazardous areas. Peak fire temperature modeling was performed for certain fire areas located within the MFFF. Areas chosen for fire modeling are areas where the fire loading exceeded 80 percent of the minimum fire loading equivalent to the fire resistance rating of the fire area barriers. The worst case simulations were found to have insignificant excursions over the ASTM E119 (ASTM, 2000a) curve for the fire duration ratings of the barriers. Evaluations performed for these worst cases identified no adverse effects on the barriers.

Fire barriers in the MFFF include the following:

- passive concrete structural barriers (walls, floors, ceilings, protected openings, etc.)
- removable panel 2-hour and 3-hour fire-rated barriers
- fire-rated removable plugs
- penetration seals
- fire propagation barriers and fire wrap
- active fire doors
- active pellet handling fire doors
- active rod process area fire doors
- [redacted]

Passive concrete structural barriers (walls, floors, ceilings, protected openings, etc.) will be constructed and designed in accordance with the applicable requirements of NFPA 220 (NFPA, 1995b) and NFPA 221, “Standard for Fire Walls and Fire Barrier Walls” (NFPA, 1997c). The concrete structural barriers are of seismic Category I design.

Removable panels are designed for the design-basis earthquake and have a 3-hour fire rating. The panels are also designed for a combined room pressure and a clean agent discharge pressure of 7 inches of water. The removable panels are also designed for the design-basis earthquake.

A few applications use fire-rated removable plugs in locations where they will ease expected maintenance during the life of the MFFF. They will have a fire resistance rating of 2 hours.

Penetration seals are fire barrier assemblies that allow the passage of system components through walls without compromising the integrity of the fire barrier rating. The penetration seal assembly designs are capable of providing a 2- or 3-hour fire rating and meet the specifications of ASTM E814, “Fire Tests of Through-Penetration Fire Stops” (ASTM, 2000b), and UL 1479 (UL, 1994). The designs are, therefore, in compliance with NFPA 801(NFPA, 1998d) and NFPA 221 (NFPA, 1997c). The panels are also designed for a combined room pressure and a clean agent discharge pressure of 7 inches of water.

Fire propagation barriers are ductwork and duct-mounted components that are credited in place of a fire damper in a specified location where ductwork penetrates a 2-hour fire barrier. Fire propagation barriers prevent the passage of flames and hot gases. Fire wrap insulation is installed on the outside of ductwork to extend the 3-hour rating between fire walls. Fire wrap insulation on the duct extends the fire area from which the duct is routed and removes it from the fire area through which it is passing. Fire propagation barriers and fire wrap are used to eliminate the need for a fire damper in specified locations where ductwork penetrates a fire barrier.

Active fire doors are used to protect openings in fire-resistive walls at the MFFF. Fire doors used in fire barriers are installed in accordance with the applicable requirements of NFPA 80, “Standard for Fire Doors and Fire Windows” (NFPA, 1999d).

The active pellet handling (PML conveyor) system transfers fuel pellets to and from different fire areas. Special automatic doors are installed in the fire area boundaries that the PML conveyor passes through. These doors are normally closed and are opened only during transfers, with their position monitored and controlled by surveillance. A fire propagation analysis performed by the applicant determined that fire will not propagate through the closed doors. The NRC reviewed the analysis (DCS01-ASI-DS-NTE-R-10353) and found it to be acceptable such that a fire will not be able to propagate through the closed doors (NRC, 2008a).

Active rod process area fire doors allow the passage of fuel rods and trays and are normally controlled by the Rod Tray Handling (SMK) Programmable Logic Controller (PLC). The control switches over to the fire safety controller upon fire detection. Mechanical sensors that are not IROFS (defense in depth) control the opening and closing motors of the fire doors and protect against closing fire doors on fuel rods. If a fire door is blocked by a rod tray, an operator will be dispatched to manually move the rod tray and close the fire door. During a 24-hour period, the doors are open only 5 to10 percent of the time (MOX, 2008).

[redacted] The automatic fire dampers in the air supply ductwork are either activated by electrical controls or resettable “fuse” link. Otherwise, operators control supply-side dampers as a defense-in-depth measure. HDE exhaust fire dampers are manually operated with a chain wheel operator, or, in some cases, they are manually operated by a remote push button. Manual operation of the fire dampers in the HDE exhaust ductwork is a defense-in-depth measure. Air-operated fire isolation valves are installed in the VHD exhaust line outside the process room fire barriers. These valves can be operated by a local hand switch, by a remote hand switch at the normal PLC monitor, or by using a manual chain wheel. Manual operation of the IROFS fire isolation valves is a defense-in-depth measure. The fire dampers and isolation valves are seismic Category I.

The applicant’s fire rated barriers meet the guidance in Section 7.4.3.2.L in NUREG-1718 (NRC, 2000) and are, therefore, acceptable.

7.3.2.9 Storage of Flammable and Combustible Liquids and Floor Drainage

The handling and use of combustible and flammable liquids are controlled by design and limited by procedures in areas containing IROFS. Flammable and combustible liquids are stored, handled, and used in accordance with applicable sections of NFPA 30 (NFPA, 1996d). Drainage in areas handling radioactive materials is sized to accommodate a spill of the largest single container of any flammable or combustible liquid in the area. Floor drainage from areas containing flammable or combustible liquids is trapped to prevent the spread of burning liquid beyond the area. The MFFF design specifies that firewater is drained and collected. The applicant's design meets the guidance in Sections 7.4.3.2.M and 7.4.3.2.DD in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

7.3.2.10 Flammable Gas Storage

Flammable and combustible liquids are stored, handled, and used in accordance with applicable portions of NFPA 50A (NFPA, 1999a) and NFPA 55 (NFPA, 1998c). Where appropriate, explosion prevention measures are implemented in accordance with applicable sections of NFPA 69 (NFPA, 1997a). This meets the guidance in Section 7.4.3.2.N in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

7.3.2.11 Fire Alarm and Detection Systems

Section 7.3.2 of the LA states that the fire alarm systems are designed according to NFPA 72 (NFPA, 1996f). Fire detection and alarm communication devices include the proprietary supervising workstation (PSW), annunciator panels, local fire alarm panels, the fire alarm panel data network, the digital alarm communications transmitter, and firefighter telephones. The fire detection and alarm system PSW is located in the polishing and utilities control room. The PSW sends fire alarm information to workstations in each process control room. The PSW also transmits fire detection system alarm signals to fire annunciator panels in the central and secondary alarm stations. Additional fire annunciator panels are provided in the emergency control rooms, the operations support center, and the utilities control room for use during emergency situations. The PSW also includes an integral digital alarm communications transmitter, which retransmits these signals to the SRS Operations Center.

Upon detection of a fire, audible and visual alarms are provided in the affected parts of the MFFF. The alarm systems are capable of annunciating and differentiating fire conditions, supervisory indicators, or trouble signals. Alarm signals are transmitted to the monitored alarm center at the SRS fire department and the aqueous polishing (AP) control room. From an in-office review of fire detection design documents (NRC, 2008a), the staff determined that initiating circuits are capable of transmitting an alarm under circuit fault conditions of single ground, open, or both. This meets the guidance in Section 7.4.3.2.P in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

Heat and smoke detectors supplement or can actuate fire extinguishing systems, fire dampers, and door closure devices. Detection systems are located throughout the MFFF in accordance with the principles of NFPA 72 (NFPA, 1996f) and based on the specific needs of the individual fire areas. Smoke or heat detectors or both are located inside gloveboxes and in the HVAC supply air ventilation intake header. Heat detectors are provided upstream of HVAC final filters. Smoke detectors are also installed in the ventilation exhaust ducts, which are inaccessible during plant operation.

Each glovebox is provided with a minimum of two detectors. Heat detectors are installed in gloveboxes prone to dusty conditions. Smoke detectors are deployed in gloveboxes where process operations generate little dust. NFPA 72 requirements will be met through redundancy and replacement (NRC, 2007a).

The primary power supply for the fire detection/alarm system is the normal power system, which has two sources of offsite alternating current power. In the event that both sources of normal power are lost, the detection/alarm system can be powered by the standby alternating current power systems, and then by the emergency power systems. The emergency power systems are IROFS. SER Section 11.5 contains the staff's evaluation of the electrical power supply systems.

7.3.2.12 Water Supply and Drainage

The MFFF design incorporates a water supply system in accordance with NFPA 801 (NFPA, 1998d) requirements. Section 7.3.4 of the LA (MOX, 2010a) describes the fire protection water supply system as consisting of an underground loop around the MFFF site, fire hydrants, and valves. [redacted] The F-Area at SRS supplies the fire water. The licensee provided curves for the three F-Area fire pumps (one electric, two diesel), showing that each pump alone could meet the MFFF fire demand.

The staff determined that the MFFF water supply system accommodates the requirements for automatic and manual suppression activities at the MFFF. This meets the guidance in Section 7.4.3.2.Q in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

7.3.2.13 Automatic Fire Suppression

A combination of automatic suppression systems, fire hose stations, exterior hydrants, and portable extinguishers provide suppression at the MFFF. Automatic suppression is provided in areas where potentially significant fire loading is present.

[redacted]

Preaction systems are used to avoid possible ingress of water into areas where fissile material is handled. Preaction systems are used in the MOX processing, shipping and receiving, and AP areas and in the emergency and standby generator buildings. Preaction systems reduce the chance of accidental discharge by requiring independent actions for water discharge such as smoke detector and sprinkler head actuation. Wet-pipe sprinkler systems (discharge water when elevated temperatures are detected) protect the administration, technical support, secured warehouse, and reagent processing buildings. Sprinklers are designed according to NFPA 13, "Standard for the Installation of Sprinkler Systems" (NFPA, 1996a). The emergency fuel storage vault (UEF) and the truck bay areas in the shipping and receiving building are equipped with automatic deluge systems. A deluge system uses open sprinklers or spray nozzles and is designed in accordance with NFPA 15, "Standard for Water Spray Fixed Systems for Fire Protection" (NFPA, 1996c). All water-based systems will be periodically inspected, tested, and maintained in accordance with NFPA 25 (NFPA, 1998a). This meets the guidance in Section 7.4.3.2.S in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

The clean agent fire suppression systems provide fire suppression in areas where water-based suppression is undesirable. Clean agent systems are divided into halogenated and nonhalogenated systems. The halogenated systems are not used in process rooms. Both types of clean agent supply are provided by high-pressure storage cylinders that are located in cylinder storage rooms and in various discrete locations near the rooms that they are protecting. A reserve quantity of each type of clean agent is provided equal to the largest demand from the largest clean agent storage location. A typical clean agent delivery system consists of agent stored in a bank of cylinders, a cylinder manifold, a distribution manifold, a pressure-reducing orifice and selector valve for each fire area, distribution piping to each fire area, and a network of distribution piping and discharge nozzles for each room or protected space within the fire area.

Following the injection of nonhalogenated clean agent, the fire detection alarm and control system automatically closes the supply ventilation dampers to the rooms. The exhaust ventilation system for the room is automatically closed after the injection of nonhalogenated clean agent, except in rooms containing dispersible radioactive material. In such rooms, the design includes a specified quantity of clean agent for extended discharge to account for the loss of suppressant through the exhaust ducting. This ensures that the concentration in the room will not drop below the design concentrations.

The design and installation of nonhalogenated clean agent systems and agent quantity requirements comply with the applicable requirements of NFPA 2001, "Standard on Clean Agent Extinguishing Systems" (NFPA, 2004), with the following exceptions:

- Extended discharges are used in rooms containing gloveboxes. The extended discharge of nonhalogenated clean agent uses dedicated clean agent cylinders.
- The minimum clean agent concentration is 20 percent more than the design concentrations provided in NFPA 2001 (NFPA, 2004).
- The soak time has been increased from 10 to 20 minutes.

To confirm the effectiveness of the minimum design concentrations provided in NFPA 2001 to extinguish a fire when the agent is released over an extended period, the clean agent was tested in accordance with Section 36 of UL 2127, "Inert Gas Clean Agent Extinguishing System Units" (UL, 2001). The results of the tests indicated that the minimum design concentrations must be 20 percent greater than those values provided in NFPA 2001 (NFPA, 2004) for the materials tested. For conservatism, the concentration of clean agent is maintained for 20 minutes (NRC, 2008b). This "soak time" of 20 minutes will cover the SRS fire department response time of 15 minutes.

Not all fire areas in the MFFF are provided with automatic suppression. Suppression is not provided in some airlocks, solvent cells, plenums, chases, and areas that are not normally occupied, have low combustible loading, or that have no ignition sources.

Based on the variety and redundancy of suppression features, the applicant does not place total reliance on a single fire suppression feature at the BMF. As described in SER Section 7.3.3, standpipe and hose systems provide backup fire suppression to the automatic suppression systems. Early fire suppression capability is also provided by the specially configured portable carbon dioxide (CO_2) extinguishers with quick disconnect fittings used for glovebox fires.

Based on information provided to date, the staff finds that the fire suppression strategy provides diversity and defense in depth. Automatic and manual fire suppression controls the spread of fire, reducing the challenge to fire barriers. The staff finds that the provisions for suppression are acceptable because they satisfy the baseline design criteria of 10 CFR 70.64(a)(3) and because they provide defense in depth. This meets the guidance in Section 7.4.3.2.J in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

7.3.2.14 Combustible and Pyrophoric Metals

When plutonium oxide is fully oxidized, it is not pyrophoric. Uranium oxide does not oxidize in an inert atmosphere, and it oxidizes very slowly in air under process temperature conditions. Section 11.1.33 of the LA (MOX, 2010a) describes the safety procedures to be used in handling zircaloy swarf. The applicant intends to use titanium for the electrolyzer circuit and associated equipment that could be exposed to silver (II) ions; the staff evaluates this practice in SER Section 11.2.1.3.4. SER Section 11.1.1.2.1 discusses the potential for uranium dioxide (UO_2) pyrophoricity and burnback. Based on the evaluations in SER Sections 11.2.1.3.4 and 11.1.1.2.1, the handling of pyrophoric material meets the guidance in Sections 7.4.3.2.U and V in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

7.3.2.15 Glovebox Protection

Gloveboxes, described in Section 11.3.9.1 of the LA, provide physical and visual access to internal equipment, processes, and material. A typical glovebox is a large, stainless steel, enclosure mounted box on a structural stainless steel stand. Glovebox windows consist of rectangular fire-resistant polycarbonate panels that fit into frames in the glovebox walls and ceilings. Lead glass panels overlay windows where radiation shielding is required to reduce operator exposures.

Light fixtures are generally installed outside of the gloveboxes. They provide illumination for the interior spaces through windows located in glovebox ceilings.

The applicant's use of polycarbonate windows is not in compliance with NFPA 801 (NFPA, 1998d) guidance. However, the applicant has demonstrated that an equivalent level of fire protection is achievable with the use of fire-resistive polycarbonate (DCS, 2000). Compared to other plastic glovebox materials, polycarbonate is relatively difficult to ignite and will not sustain combustion without an external heat flux. The applicant stated that under seismic inertia loading and seismic deflection, polycarbonate is superior to noncombustible materials that are allowed by the code, such as glass. This meets the guidance in Section 7.4.3.2.W of NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

SER Section 11.3 evaluates other aspects of glovebox fire protection as related to the MFFF ventilation system. The discussion of design features of C4 confinement systems (Table 11.3-1 of the LA) lists adequate features to demonstrate that the facility meets the guidance in Sections 7.4.3.2.X and 7.4.3.2.Y of NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

7.3.2.16 Flammable and Combustible Liquids and Gases

Section 7.2.3 of the LA states that flammable and combustible liquids are stored and handled in accordance with NFPA 30 (NFPA, 1996d). Most vessels containing flammable or combustible liquids are located in process cells. [redacted] Flammable and combustible gases are stored and handled in accordance with NFPA 50A (NFPA, 1999a) and NFPA 55 (NFPA, 1998c). The discussion of various explosion events in SER Chapter 8 contains a detailed safety evaluation of flammable and combustible liquids and gases.

7.3.2.17 Special Hazards

BMF emergency battery rooms are located in the MFFF shipping and receiving building and are separated from other areas by 3-hour fire walls and provisions for ventilation and hydrogen gas detection. The provisions limit hydrogen gas accumulation to less than 2 percent by volume (50 percent lower flammable limit). The fire protection features are adequate by the standards in IEEE 484, "Recommended Practice for Installation Design and Installation of Vented-Led Acid Batteries for Stationary Applications" (IEEE, 1996).

The BMF has several laboratories that are used for physical and chemical analyses of samples from the AP or MOX Processing (MP) processes. Fire protection for laboratories is designed to meet NFPA 45 "Standard for Fire protection for Laboratories Using Chemicals" (NFPA, 1996e). The isolation of these special hazards reduces the potential fire damage, while escape routes are safeguarded. This meets the guidance in Section 7.4.3.2.CC in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

7.3.3 Manual Firefighting Capability

The standpipe and hose systems allow manual firefighting capabilities throughout the MFFF. Because of criticality concerns, the MP, Shipping and Receiving (SR), and AP areas have a dry standpipe system instead of the normally pressurized wet standpipe system. The standpipe systems are designed for use by the SRS fire department. For normally dry standpipes, operation of the system requires the opening of an isolation valve. After the fire is extinguished, the standpipe water supply is secured and the standpipe is drained.

Portable fire extinguishers are provided throughout the BMF and inside all buildings at the MFFF so occupants can extinguish small fires. Extinguishers are selected and located according to fire hazards and to their effectiveness. A combination of multipurpose dry chemical, metal use, and CO_2 extinguishers are provided. Portable extinguishers are provided according to NFPA 10, "Standard for Portable Fire Extinguishers" (NFPA, 1998b). Specially configured portable CO_2 bottles are provided in rooms with gloveboxes. These extinguishers can be quickly disconnected and attached to the glovebox to suppress fires within the glovebox without overpressurization.

MFFF manual fire fighting capability consists of the SRS FD. The SRS FD is a full time professional fire department sufficiently trained and qualified to fight MFFF fires. Manual fire fighting needs assessments conducted by the SRS FD and Shaw AREVA MOX Services, LLC determined that minimum onsite fire fighting capabilities are met by the SRS FD. The staff agrees with this conclusion because the SRS FD meets all appropriate NFPA Standards for fire suppression activities including NFPA 1500 "Standard on Fire Department Occupational Safety and Health Program" (NFPA, 1997e). NFPA 1500 covers requirements for training; fire apparatus and equipment; protective clothing and equipment; and emergency operations. The NFPA 1500 requirements meet and/or exceed the requirements of NFPA 600 "Standard on Industrial Fire Brigades" (NFPA, 1996h) which is referred to in NUREG-1718 (NRC, 2000) as guidance for manual firefighting capability. The MFFF fire protection baseline needs assessment also provided the maximum arrival time for the SRS FD after an alarm signal, determined to be 15 minutes (Tiktinsky to Kotzalas, June 2008) (NRC, 2008a).

This meets the guidance in Section 7.4.3.3 in NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

7.3.4 Fire Hazard Analysis

The fire hazard analysis (FHA) documents the specific fire hazards, the fire protection features proposed to control those hazards, and the adequacy of MFFF fire safety. The FHA provides information for each fire area and describes operational concerns that can affect fire safety in the MFFF.

To prepare an FHA, the applicant divided the MFFF into fire areas and evaluated the fire safety of each fire area and the MFFF as a whole. Next, the applicant identified the fire barriers that surround each fire area and analyzed each fire area to determine the types of combustibles and ignition sources expected to be present. The facility includes IROFS as needed to satisfy the safety function that is specified by the ISA for each fire area. The applicant also determined the planned fire protection systems (detection, suppression, and barriers) for each fire area, as well as the codes and standards to be used in the design of the fire protection systems.

The applicant incorporated information from the FHA into the nuclear safety evaluations for fire events as reviewed by the staff in June 2007 (NRC, 2007a). As a supporting document, the FHA accomplishes the following:

- demonstrates that the multiple levels of fire protection provided ensure adequate protection of the MFFF from fires

- analyzes the potential fires at MFFF, including areas where measures have been taken to prevent fires from occurring and areas where fires can occur but measures have been taken to mitigate their effects

- identifies the fire areas throughout the MFFF for limiting fire spread, protecting personnel, and limiting consequential damage

- demonstrates the adequacy of fire barrier walls, floors, and ceilings in concert with HVAC systems to confine the design-basis fires

In Section 7.5 of the LA, the applicant committed to reviewing and periodically updating the FHA at defined intervals and as necessary following changes and modifications to the facility, processes, or inventories in accordance with MOX Services' configuration management processes.

The staff reviewed the FHA (NRC, 2007a) and concludes that the FHA meets the guidance in Appendix D to NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

7.3.5 Fire Protection Items Relied on for Safety

As described in Section 7.3.4.3 of the ISA Summary, (MOX 2010b), the applicant's safety assessment demonstrates that the MFFF IROFS provide protection against hazards in accordance with the requirements of 10 CFR 70.61. The IROFS described in this section ensure that adequate protection is provided against fires.

7.3.5.1 Combustible Material Loading Controls

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7.3.6 MOX Fire Scenarios

The MOX ISA organized fire events into 21 separate groups:

- F-01—Fire in AP Process Cells
- F-02, 04—Fire Involving Gloveboxes in the AP/MP C3 Glovebox Areas

- F-03—Fire in the AP Gloveboxes with Vessels Containing Solvents
- F-05—Fire in Titanium/Electrolyzer in the AP/MP C3 Glovebox Area
- F-06—Fire in 3013 Canister in C1 and/or C2 Areas
- F-07—Fire in 3013 Transport Cask in C1 and/or C2 Areas
- F-08—Fire in Fuel Assembly and Inspection Area
- F-09—Fire in MOX Fuel Transport Cask in C1 and/or C2 Area
- F-10—Fires in Waste Container in C1 and/or C2 Areas
- F-11—Fire in Transfer Container in C1 and/or C2 Areas
- F-12—Fire in Final C4 HEPA Filter C1 and/or C2 Areas
- F-13—Fire Outside the BMF Propagating to the Inside
- F-14—Fire in Facilitywide Systems (Pneumatic Transfer)
- F-15—Fire in Facility Propagating from One Fire Area to Another
- F-16—Fire in Secured Warehouse
- F-17—Fire Involving Vessels Containing Solvents in C2 Confinement Areas
- F-18—Fire in UO_2 Intermediate Storage Room in C2 Area
- F-19—Fire Event Involving Zircaloy Swarf
- F- 20—Fire Event Potentially Producing Excessive Soot in a Fire Area Ventilated by HDE
- F -21—Fire Event in the Hydraulic Pump Room Ventilated by HDE

7.3.6.1 Fire in the Aqueous Polishing Process Cells (F-01)

AP process cells are rooms that contain process equipment such as vessels and piping that require no routine maintenance or inspection. Process cells contain no known ignition sources and electrical equipment, wiring, or lighting. The only combustibles in a process cell are solutions contained within some process equipment that is constructed of titanium, stainless steel, or carbon steel and welded together. Process cell walls are generally [redacted] of concrete and structural steel. During maintenance activities, hazardous materials are removed from the cells. In evaluating the potential for fire in the process cell, five different scenarios were considered by the applicant:

(1) fire originating within a process vessel

(2) fire originating outside a process vessel but within a process cell
(3) fire resulting from or occurring after a process vessel leak
(4) fire originating external to a process cell
(5) fire during a maintenance activity

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The NRC staff finds that use of the designated IROFS is acceptable to comply with the single failure criterion for this event. The single failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MOX Project Qualify Assurance Program (MPQAP)), and the use of codes and standards for engineered IROFS, gives the NRC staff reasonable assurance that this high-consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61 in the event of a fire in the AP process cells.

7.3.6.2 Fire Involving Gloveboxes (F-02 and F-04)

Gloveboxes confine radioactive materials and are located within process rooms (C3 confinement zone) that are ventilated by the HDE system. The C4 confinement barriers and the VHD exhaust system maintain a negative differential pressure between the gloveboxes and the interior of the process rooms. Combustible materials within process rooms include plastic and wire insulation in instrumentation and detectors, electrical cable insulation, transient combustibles, bag port sleeves, and electric process motor wire insulation. Combustibles associated with the glovebox include window seals, window shielding, glove ring plugs, gloves, and polycarbonate windows.

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The NRC staff finds that use of the designated IROFS is acceptable to comply with the single failure criterion for this event. The single failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS, gives the NRC staff reasonable assurance that this high-consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61 in the event of a fire involving gloveboxes.

7.3.6.3 Fire in Aqueous Polishing Gloveboxes with Vessels Containing Solvents or Other Explosion Materials (F-03)

SER Section 8.1.2.4.2 evaluates solvent fires and explosions.

7.3.6.4 Fire in the Titanium Electrolyzer in the Aqueous Polishing Glovebox Areas (F-05)

SER Section 11.2.1.3.4 evaluates a fire in the titanium electrolyzer.

7.3.6.5 Fire Event Involving a 3013 Container, C1 and/or C2 Areas (F-06)

Fires that could affect 3013 containers of PuO_2 were postulated to occur in the C1 and C2 areas. Estimates of fire temperature were determined in the B-156 pallet preparation room and the B-155 density measurement room, with the density measurement room having the higher heat flux and longer fire duration.

[redacted]

[redacted]

[redacted]

7.3.6.6 Fire Event in the Plutonium Dioxide Truck Bay Receiving Area (F-07)

A fire in the PuO truck bay receiving area is postulated to begin in the cask handling room, D-102. [redacted]

[redacted]

[redacted]

[redacted]

The NRC staff finds that use of the designated IROFS is acceptable to comply with the single failure criterion for this event. The single failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS, gives the NRC staff reasonable assurance that this high-consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61 in the event of a fire in the PuO_2 truck bay receiving area.

7.3.6.7 Fire Event Involving Fuel Rods/Fuel Assembly (F-08)

A fire event may be postulated to occur in the fuel assembly and inspection area (B-174), mockup loading area (B-174a), assembly storage area (B-183), handling area (B-185), egress air lock (B-189), and plenum (B-199). The rod storage and handling area (B-186) is a second fire area of concern.

[redacted]

[redacted]

For the zircaloy swarf fire, the applicant determined the maximum temperature of the rod directly in front of the fire at a radial distance of 1.03 meters to be 654 degrees F. This temperature was also found to be insufficient to cause damage.

The staff reviewed the justification for the quantities of hydraulic fluid and zircaloy swarf assumed in the analyses in calculation DCS01-ASI-DS-CAL-R-10110-A and were found to be acceptable based on the facility design and expected quantities of these materials (NRC, 2008a).

[redacted]

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The NRC staff finds that use of the designated IROFS is acceptable to comply with the single failure criterion for this event. The single failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS, gives the NRC staff reasonable assurance that this high-consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61 in the event of a fire involving fuel rods or a fuel assembly.

7.3.6.8 Fire Event in MOX Fuel Transport Cask in C1 or C2 Areas (F-09)

A diesel fuel fire can be postulated in the fuel truck bay that could affect new fuel in a fuel transport cask. [redacted]

[redacted]

[redacted]

The NRC staff finds that use of the designated IROFS is acceptable to comply with the single failure criterion for this event. The single failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS, gives the NRC staff reasonable assurance that this high-consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61 in the event of a fire involving a MOX fuel transport cask in C1 or C2 areas.

7.3.6.9 Fire Event in a MOX Waste Container in C1, C2, or C3 Areas (F-10)

Waste containers are stored in room B-254, which is a C3 area. [redacted]

[redacted]

[redacted]

[redacted]

The NRC staff finds that use of the designated IROFS is acceptable to comply with the single failure criterion for this event. The single failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS, gives the NRC staff reasonable assurance that this high-consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61 in the event of a fire involving a MOX waste container in C1, C2, or C3 areas.

7.3.6.10 Fire Event in a Transfer Container in C2 Area (F-11)

A fire event is postulated to occur in a personnel and material corridor, to grow, and eventually to impact a transfer container. [redacted]

[redacted]

The NRC staff finds that use of the designated IROFS is acceptable to comply with the single failure criterion for this event. The single failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards, gives the NRC staff reasonable assurance that this high-consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61 in the event of a fire impacting a transfer container in a C2 area.

7.3.6.11 Fire Event Involving KWG, Process Cell Exhaust, and Very High Depressurization Exhaust Final High-Efficiency Particulate Air Filters (F-12)

A fire is postulated to occur in fire areas that contain fans and final HEPA filter assemblies (e.g., rooms B-389, B-390, and B-397a/b, C2 areas for VHD final HEPA filters, rooms C-416 and C-429, and C3b area for KWG final HEPA filters; and rooms B-213 and B-388, C2 area for POE final HEPA filters). [redacted]

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The NRC staff finds that use of the designated IROFS is acceptable to comply with the single failure criterion for this event. The single failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards, gives the NRC staff reasonable assurance that this high-consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61 in the event of a fire involving the KWG, POE, and VHD final HEPA filters.

7.3.6.12 Fire Event Outside the MOX Fuel Fabrication Building (F-13, NPH-04, and EMMH-4)

Fires are postulated to occur external to the BMF, the BEG, and associated UEF and the waste transfer line. An external fire affecting the BMF has the potential to result in radiological and chemical consequences to all potential receptors. [redacted]

[redacted]

[redacted]

The NRC staff finds that use of the designated IROFS is acceptable to comply with the single failure criterion for this event. The single failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS, gives the NRC staff reasonable assurance that this high-consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61 in the event of a fire outside the BMF.

7.3.6.13 Fire Event Involving Facilitywide Systems (F-14)

Facilitywide systems that can be affected by fire include the HVAC systems, the electrical systems, and the pneumatic transfer systems. Other sections of this SER evaluate fire events that could affect the HVAC system. SER Section 7.3.6.1 addresses a fire event in the AP process cell, SER Section 7.3.6.2 discusses a fire event involving gloveboxes, and SER Section 7.3.6.12 considers a fire event outside the BMF.

The electrical systems are facilitywide systems that can be affected by fire. SER Section 7.3.2.4 discusses protection of the electrical systems from the effects of fires.

The pneumatic transfer system consists of the NTP PuO_2 (133 millimeter) can pneumatic transfer system, the LTP (76 millimeter) sample pneumatic transfer systems, and the LLP (33 millimeter) laboratory pneumatic transfer system. [redacted]

[redacted]

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The NRC staff finds that use of the designated IROFS is acceptable to comply with the single failure criterion for this event. The single failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS, gives the NRC staff reasonable assurance that this high-consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61 in the event of a fire involving facilitywide systems.

7.3.6.14 Fire Propagation from One Fire Area to Another Fire Area (F-15)

The event postulated is a fire propagating from one fire area to another fire area within the facility through an opening in a fire barrier. The second fire area is assumed to be a C3b glovebox room containing radioactive material.

[redacted]

[redacted]

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The NRC staff finds that use of the designated IROFS is acceptable to comply with the single failure criterion for this event. The single failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS, gives the NRC staff reasonable assurance that this high-consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61 in the event of fire propagation from one fire area to another fire area.

7.3.6.15 Fire Event in Secured Warehouse Building (F-16)

A fire has been postulated to occur in the secured warehouse building (BSW) involving UO_2 powder stored in drums. Potential ignition sources for this fire are electrical circuit failures, mechanical failures of equipment, or maintenance activities. The fire is assumed to result in overpressurization and rupture of UO_2 powder storage drums.

[redacted]

[redacted]

[redacted]

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[redacted]

The NRC staff finds that use of the designated IROFS is acceptable to comply with the single failure criterion for this event. The single failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS, gives the NRC staff reasonable assurance that this high-consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61 in the event of a fire in the secured warehouse.

7.3.6.16 Fire Event Involving Vessels Containing Solvents in C2 Confinement Areas (F-17)

SER Chapter 8 includes an evaluation of solvent fires and explosions in the discussion of explosion events.

7.3.6.17 Fire in UO_2 Intermediate Storage Room in C2 Area (F-18)

A fire event was postulated to occur in the UO_2 intermediate storage room, which contains drums of UO as a staging area before the addition of dosing units. [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

The NRC staff finds that use of the designated IROFS is acceptable to comply with the single failure criterion for this event. The single failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS, gives the NRC staff reasonable assurance that this high-consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61 in the event of a fire in a C2 confinement area involving drums of depleted uranium oxide powder.

7.3.6.18 Fire Event Involving Zircaloy Swarf (F-19)

Section 5.3.4.2.18 of the ISA Summary evaluated a fire event involving zircaloy filings or swarf or both as a generic fire event. Section 5.3.4.2.7 of the ISA Summary evaluated a fire caused by zircaloy chips as a possible fire source for fuel rods and fuel assemblies. [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

The NRC staff finds that use of the designated IROFS is acceptable to comply with the single failure criterion for this event. The single failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS, gives the NRC staff reasonable assurance that this high-consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61 in the event of a fire involving zircaloy swarf.

7.3.6.19 Fire Event Potentially Producing Excessive Soot in a Fire Area Ventilated by HDE (F-20)

This fire event is postulated to involve all fire areas ventilated by the HDE system, including those that do not contain dispersible radioactive material. [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

The NRC staff finds that use of the designated IROFS is acceptable to comply with the single failure criterion for this event. The single failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS, gives the NRC staff reasonable assurance that this high-consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61 in the event of a fire potentially producing excessive soot in a fire area ventilated by HDE.

7.3.6.20 Fire Event in the Hydraulic Pump Room Ventilated by HDE (F-21)

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

The NRC staff finds that use of the designated IROFS is acceptable to comply with the single failure criterion for this event. The single failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS, gives the NRC staff reasonable assurance that this high-consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61 in the event of a fire in the hydraulic pump room ventilated by the HDE system.

7.4 <u>Evaluation Findings</u>

7.4.1 Compliance with 10 CFR 70.22 and 10 CFR 70.23

The staff reviewed the LA for a license to possess and use special nuclear material for the MFFF according to Chapter 7 of NUREG-1718. The staff evaluated the organization and conduct of operations, facility fire protection features and systems, manual firefighting capability, and the FHA and concludes that the applicant's proposed equipment and facilities are adequately described and will protect health and minimize danger to life or property.

7.4.2 Compliance with the 10 CFR 70.61 Performance Requirements

The staff reviewed the applicant's fire accident analyses, including the reliability and applicability of selected IROFS to the postulated initiators and fire area hazards. The staff concludes that the applicant's proposed equipment, facilities, and procedures provide a reasonable level of assurance that adequate fire protection will be provided and maintained for those IROFS needed to meet the safety performance requirements and the baseline design criteria of 10 CFR 70.61.

7.4.3 Baseline Design Criteria

The staff reviewed the design bases for fire protection systems, fire-related administrative controls, and buildings as described in the LA and ISA Summary. The staff concludes that fire protection-related IROFS and defense-in-depth controls will be designed, constructed, and used consistent with good engineering practice, which dictates that certain minimum requirements be applied as design and safety considerations for any new nuclear process or facility. The applicant meets these minimum requirements through its use of NFPA 801 (1998d) and other applicable, nationally accepted fire protection codes and standards. The staff concludes that the facility meets the requirements of 10 CFR 70.64(a)(3) with respect to fire protection.

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8.0 CHEMICAL SAFETY

8.1 Conduct of Review

This chapter of the safety evaluation report (SER) contains the U.S. Nuclear Regulatory Commission (NRC) staff's review of chemical and process safety described by the applicant, Shaw AREVA MOX Services (MOX Services), in Chapter 8 of the license application (LA) (MOX, 2009a), with supporting process safety information from Chapters 5 and 11 of the LA. The purpose of the NRC's review of the MOX Services chemical safety program and the design of the facility is to evaluate whether the applicant will adequately protect workers, the public, and the environment during normal operations against chemical hazards of licensed material and its byproducts. The chemical safety program and the facility's design must also protect against facility conditions or operator actions or both that can affect the safety of licensed materials and thus present an increased chemical risk.

Regulatory Requirements

The regulatory bases for the review are the general and additional contents of an application that address chemical process safety, as required by Title 10 of the *Code of Federal Regulations* (10 CFR) 70.22, "Contents of Applications," and 10 CFR 70.65, "Additional Content of Applications." In addition, the staff reviewed chemical process safety information in the LA and the Integrated Safety Analysis (ISA) Summary (MOX, 2009b) and supporting information to provide reasonable assurance of compliance with 10 CFR 70.61, "Performance Requirements," 10 CFR 70.62, "Safety Program and Integrated Safety Analysis," and 10 CFR 70.64, "Requirements for New Facilities or New Processes at Existing Facilities."

Regulatory Guidance and Acceptance Criteria

The review of the LA (MOX, 2009a) and ISA Summary (MOX, 2009b) focused on the design bases of chemical process safety systems and components. For each chemical process safety system, the staff reviewed information provided by the applicant concerning the safety function, system description, and safety analysis. The review also included design-basis considerations, such as redundancy, independence, reliability, and quality. Chapter 8 of NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility" (NRC, 2000), contains the guidance applicable to the NRC's review of chemical process safety for the proposed facility. This chapter is applicable in its entirety. The staff also used NUREG-1601, "Chemical Process Safety at Fuel Cycle Facilities" (NRC, 1997), and NUREG-1513, "Integrated Safety Analysis Guidance Document" (NRC, 2001), as guidance documents for this review. Section 8.4.3 of NUREG-1718 (NRC, 2000) identifies the acceptance criteria applicable to this review.

8.1.1 Background

As stated in the memorandum of understanding (MOU) between the NRC and the Occupational Safety and Health Administration (OSHA) entitled, "Worker Protection at NRC-licensed Facilities" (Volume 53, Number 210, of the *Federal Register*, dated October 31, 1998, pages 43950–43951), the NRC oversees chemical safety issues related to (1) radiation risk produced by radioactive materials, (2) chemical risk produced by radioactive materials, and (3) plant conditions that affect the safety and safe handling of radioactive materials. These types of chemical safety issues represent an increased radiation risk to the workers. However, the NRC does not oversee facility conditions that result in an occupational risk but do not affect the safe use of licensed material. The NRC has codified the MOU provisions applicable to the Mixed Oxide Fuel Fabrication Facility (MFFF) in 10 CFR 70.64(a)(5).

The NRC staff reviewed the LA (MOX, 2009a) and the ISA Summary (MOX, 2009b) submitted by the applicant and considered the following areas:

- chemical process description
- list of hazardous chemicals affecting licensed materials
- chemical accident sequences
- chemical accident consequences
- chemical process items relied on for safety (IROFS)
- management measures
- chemical process safety interfaces
- baseline design criteria (BDC)

The staff reviewed the applicant's responses to requests for additional information and ISA documents, as necessary, to better understand the process and safety requirements. The staff evaluated the information to determine whether the facility's design complied with the BDC and defense-in-depth requirements specified in 10 CFR 70.64(a) and 10 CFR 70.64(b), respectively. Chapter 5 of this SER discusses compliance with these regulations in more detail. The following sections summarize general information about the MFFF processes and the NRC staff's evaluation.

8.1.2 Areas of Review and Evaluation Findings

8.1.2.1 Chemical Processes

8.1.2.1.1 Aqueous Polishing Process

The overall function of the aqueous polishing (AP) process is to remove impurities from the feed plutonium for use in the mixed oxide (MOX) processing (MP) area. The MFFF will receive plutonium feedstock as plutonium dioxide (PuO_2). The U.S. Department of Energy's (DOE) Pit Disassembly and Conversion Facility (PDCF) located nearby will disassemble plutonium pits from weapons and convert the material to PuO_2 for use as MFFF feedstock, utilizing advanced recovery and integrated extraction system (ARIES) technology. A smaller amount of plutonium from other DOE sources will also be utilized as MFFF feedstock (referred to as alternate feedstock (AFS)). The PuO_2 received at the MFFF contains small amounts of impurities that have to be removed before the MOX fuel can be used in reactors. For PDCF/ARIES feeds, these impurities are primarily gallium, americium, and highly enriched uranium. For AFS feeds, the number of impurities and the impurity concentrations are higher. The AP process consists of the following 17 process units or systems (unit designators are indicated in parentheses):

(1) decanning (KDA) unit
(2) milling (KDM) unit
(3) recanning (KDR) unit
(4) dissolution (KDB) unit
(5) dechlorination and dissolution (KDD) unit
(6) purification cycle (KPA) unit
(7) solvent recovery (KPB) unit

(8) oxalic precipitation, filtration, and oxidation (KCA) unit
(9) homogenization (KCB) unit
(10) canning (KCC) unit
(11) oxalic mother liquor recovery (KCD) unit
(12) acid recovery (KPC) unit
(13) process off-gas treatment (KWG) unit
(14) aqueous waste reception (KWD) unit
(15) solvent waste reception (KWS) unit
(16) automatic sampling (KPG) unit
(17) laboratory liquid waste receipt (LGF) unit

The AP process involves three major steps: dissolution, purification, and conversion. These steps may be preceded by an optional pretreatment step, depending upon the impurities of the feedstock. Figure 8.1-1 is a block diagram of the AP process.

[redacted]

8.1.2.1.2 MOX Process

The MP area receives polished PuO_2 from the AP process, uranium dioxide (UO_2) depleted in the uranium-235 (^{235}U) isotope, and the required components for assembling light-water reactor MOX fuel assemblies. The process mixes the plutonium and uranium oxides to form MOX fuel pellets. The pellets are loaded into fuel rods, which are then assembled into MOX fuel assemblies for use in commercial reactors. The MP area is designed to process up to 87 metric tons of heavy metal (uranium and plutonium) annually.

The MFFF uses the advanced MIcronized Master Blend (A-MIMAS) process for the manufacture of MOX fuel assemblies. A-MIMAS represents the latest evolution of the successive MIMAS fabrication processes, adopted by BELGONUCLEAIRE and COGEMA, to produce MOX fuel pellets. A-MIMAS uses a two-step mixing process. In the first step, the PuO_2

powder is mixed with depleted UO_2 and recycled scrap powder to form a primary blend (master blend), with a nominal PuO_2 content of 20 percent of the total mass. This mix is then micronized. In the second step, the primary blend is forced through a sieve and poured into a jar and mixed with depleted UO_2 and scrap powder to obtain the final blend, which will have the specified plutonium content. The maximum PuO_2 content in the final blend is nominally 6 percent of the total mass. The two-step mixing process is used to ensure a consistent product.

8.1.2.1.3 Laboratory

The MFFF laboratory is located in the MP area. The main portion of the laboratory is located on the third floor of the MP building (BMP). The other portion is located on the intermediate level of the BMP. The three major sections of the laboratory comprise the following:

(1) the MFFF laboratory
(2) the test line (LCT)
(3) the 33-millimeter (mm) pneumatic transfer system (LLP)

The MFFF laboratory is primarily used to perform chemical and physical analyses of samples coming from the MP production units, AP production units, and the test line (LCT). Analyses are performed in the laboratory for the following purposes:

- manufacturing control (process control)
- material control & accountability
- product quality control (specification analyses)
- criticality safety
- process safety
- subsequent waste disposition at the Savannah River Site (SRS)

The following operations are also performed in the laboratory:

- laboratory liquid and solid waste management
- preparation of reagents used in the MFFF laboratory
- analysis of depleted uranyl nitrate samples
- temporary storage of scrap materials from the MFFF laboratory
- dissolution tests of AFS PuO_2 powders
- dissolution operations occasionally performed for dissolution (KDB) and dissolution of chlorination feed (KDD) bag prefilter residues (less than 40 grams of plutonium)
- calibration
- document storage

Samples are transferred from the MP and AP areas to the MFFF laboratory in vials. Samples are transferred between the different analytical units of the laboratory in aliquot containers, where an aliquot is a measured portion of a sample taken for analysis. Vials and aliquot containers specific to liquid and solid samples are not reusable. Transfers are either manual or pneumatic. However, most transfers are pneumatic via the LLP or the 76-mm pneumatic transfer system (LTP). Additional discussion of the laboratory can be found in Section 11.11 of this SER.

8.1.2.1.4 Chemical Reagents

The AP and MP processes at the MFFF use a wide variety of chemicals, a significant number of which are hazardous. Table 8.1-1 lists the hazardous characteristics and incompatibilities associated with these chemicals. Of the chemicals used in the AP and MP processes, at least 20 exhibit one of the following hazardous characteristics:

- corrosivity
- flammability
- explosivity
- chemical burn
- toxicity

Tables 8.1-1 and 8.1-2 list the process chemicals used in the AP and MP processes, including the chemical formula, chemical state, and central abstract system registry number (CASRN). Tables 8.1-1 and 8.1-2 also list two chemicals (uranyl nitrate and UO_2), respectively, which are stored in the secured warehouse building (BSW).

The AP process uses numerous reagents. The applicant has designed these reagent systems to maintain segregation or separation of vessels and components from incompatible chemicals in order to prevent chemical explosions under normal, off-normal, and accident conditions, including earthquakes. Control of the chemical makeup of the reagents introduced into the cells or AP reagent rooms prevents explosions caused by chemical reactions. The applicant has committed to labeling chemicals, piping, tanks, and other components to prevent reagent preparation errors. The applicant has also indicated that it will perform reagent handling in accordance with material safety data sheet requirements (Section 11.8 of this SER discusses these systems in greater detail).

The following reagents support the AP process functions:

- nitric acid
- tributyl phosphate (TBP)
- hydroxylamine nitrate (HAN)
- hydrazine
- sodium hydroxide
- oxalic acid
- hydrogenated polypropylene tetramer (diluent)
- sodium carbonate
- hydrogen peroxide
- manganese nitrate
- aluminum nitrate
- zirconium nitrate
- silver nitrate
- sodium sulfite
- sodium nitrite

- uranyl nitrate

Table 8.1-3 provides a list of these reagents (some are listed more than once if they used in differing concentrations), along with the downstream transfer unit and the normal operating range.

In addition, the following reagents will be used in either the MP process or as oxygen scavengers in the steam and condensate system (SPS):

- zinc stearate
- azodicarbonamide
- carbohydrazide
- morpholine

Zinc stearate is a lubricant used in the MP process. It is packed in small, ready-to-use plastic bags that will be manually introduced into the relevant powder process glovebox to be mixed with powders in process. The bags will be introduced into the gloveboxes via a glove port using a "bag-in bag-out" procedure.

Azodicarbonamide is a pore-former used in the MP process. It will also be packed in small, ready-to-use plastic bags that are manually introduced into the relevant powder process glovebox to be mixed with powders in process. The bags are introduced into the gloveboxes via a glove port using a bag-in bag-out procedure.

Carbohydrazide and morpholine are used as oxygen scavengers in the steam and condensate (SPS) system. Carbohydrazide will be purchased as a solid, while morpholine will be delivered as a liquid.

Table 8.1-1 Summary of Process Chemicals in the Aqueous Polishing Building

CHEMICAL			
Name	**Formula**	**CASRN**	**State**
Aluminum Nitrate	$Al(NO_3)_3*9H_2O$	13473-90-0	Liquid
Butanol (Note 3)	$C_4H_{10}O$	71-36-3	Liquid
Butyl Nitrate (Note 3)	$C_4H_9\ NO_3$	928-45-0	Liquid
Chlorine	Cl_2	7782-50-5	Gas
Dibutyl Phosphate (Note 3)	$C_8H_{19}\ PO_4$	107-66-4	Liquid
Dilutent, HPT (C10-C13 Isoalkanes)	$C_{12}H_{26}$ (mixture)	68551-17-7	Liquid
Hydrazine (0.2 N)	N_2H_4	302-01-2	Liquid
Hydrazine Nitrate	$N_2H_4\ NO_3$	13464-97-6	Liquid
Hydrazoic Acid	HN_3	7782-79-8	Liquid
Hydrogen Peroxide	H_2O_2	7722-84-1	Liquid
Hydroxylamine Nitrate	$NH_2OH\text{-}NO_3$	13465-08-2	Liquid
Manganese Nitrate	$Mn(NO_3)_2$	10377-66-9	Solid/Liquid
Nitric Acid	HNO_3	7697-37-2	Liquid
Nitric Oxide (Note 1)	NO	10102-43-9	Gas
Nitrogen	N_2	7727-37-2	Gas
Nitrogen Dioxide	NO_2	10102-44-0	Gas
Nitrogen Oxides (Note 1)	NO_X	N/A	Gas
Oxalic Acid	$H_2C_2O_4$	144-62-7	Liquid
Oxygen	O_2	N/A	Gas

Plutonium Dioxide	PuO_2	N/A	Solid
Plutonium Oxalate (Note 2)	$Pu(C_2O_4)_2$	N/A	Solid/Liquid
Plutonium Nitrate (Note 2)	$Pu(NO_3)_4$	N/A	Liquid
Silver Nitrate	$AgNO_3$	7761-88-8	Solid/Liquid
Sodium Carbonate	Na_2CO_3	497-19-8	Liquid
Sodium Hydroxide	NaOH	1310-73-2	Liquid
Sodium Nitrate	$NaNO_3$	7632-00-0	Liquid
Sodium Sulfite	Na_2SO_3	7757-83-7	Liquid
Tributyl Phosphate	$(C_4H_9)_3PO_4$	126-73-8	Liquid
Uranyl Nitrate	$UO_2(NO_3)_2$	36478-76-9	Liquid
Zirconium Nitrate	$Zr(NO_3)_2$*$5H_2O$	13746-89-9	Liquid

Notes:

1. Chlorine and nitrogen oxides are byproducts of AP processing.
2. Plutonium oxalate and plutonium nitrate are intermediate products of AP processing.
3. Butanol, dibutyl phosphate, monobutyl phosphate, and butyl nitrate are byproducts of TBP degradation.

Table 8.1-2 Summary of Process Chemicals in the MOX Processing Building

CHEMICAL			
Name	**Formula**	**CASRN**	**State**
Argon-Hydrogen	95% Ar, 5% H	N/A	Gas
Azodicarbonamide (pore-former)	$H_2NCONNCONH_2$	123-77-3	Solid
Carbohydrazide	CH_6N_4O	497-18-7	Solid
Helium	He	7440-59-7	Gas
Isopropanol	C_3H_7OH	67-63-0	Liquid
Morphaline Borane	$C_4H_{12}BNO$	4856-95-5	Solid
Nitrogen	N_2	7727-37-9	Gas
Plutonium Dioxide	PuO_2	N/A	Solid
Uranium Dioxide	UO_2	1344-57-6	Solid
Zinc Stearate	$Zn(C_{18}H_{35}O_2)_2$	557-05-1	Solid

8.1.2.1.5 Chemical Process Inventories

The applicant provided chemical inventory information in Chapters 8 and 11 of the LA (MOX, 2009a). Table 8.1-3 summarizes the maximum chemical inventories per vessel used by the applicant for chemical consequence analyses at the MFFF.

Table 8.1-3 Summary of Maximum Chemical Inventories per Vessel Used for Chemical Consequence Analyses at MFFF

[redacted]

[redacted]

[redacted]

Notes:
1. S (Solid); L (Liquid)
2. Evaluated for operating (elevated) temperature releases. These are the only process chemicals that were exposed to temperatures higher than room temperature and were deemed capable of exceeding the temporary emergency exposure limits (TEELs) because of their relatively low vapor pressures compared to the other process chemicals handled in the facility.

The applicant provided unit-specific chemical inventory information for each unit within the AP and MP processing areas within the individual unit descriptions provided in Chapter 11 of the LA (MOX, 2009a) and Chapter 4 of the ISA Summary (MOX, 2009b). The applicant also provided Unit- and vessel-specific expected radionuclide information in Chapter 11 of the LA (MOX, 2009a) and Chapter 4 of the ISA Summary (MOX, 2009b).

The staff reviewed the list of chemicals, radionuclides, and the associated inventory information provided by the applicant. The chemical listing appears sufficiently complete, and the quantities appear consistent with the proposed activities and the amount necessary for safety assessments. The staff finds that this information is consistent with the guidance provided in Acceptance Criterion 8.4.3.1-E of NUREG-1718 (NRC, 2000); therefore, the information is adequate and acceptable.

8.1.2.1.6 Chemical Process Ranges

According to guidance provided in NUREG-1718 (NRC, 2000), the description of the range of chemicals should include the approximate input, in-process, and output ranges of chemical and radioisotope concentrations, mass flow rates, and other properties.

The applicant provided unit-specific chemical process information for each unit within the AP and MP processing areas within the individual unit descriptions provided in Chapter 11 of the LA

(MOX, 2009a) and Chapter 4 of the ISA Summary (MOX, 2009b). These descriptions provided information such as maximum equipment volumes in a particular unit under PDCF/ARIES and AFS feed conditions, organic-to-aqueous (O/A) phase flow ratios, and nominal temperatures. This information, together with the information on chemical inventories and chemical process limits described in Sections 8.1.2.1.5 and 8.1.2.1.7 of this SER, provide an overview of the ranges of process operations expected in the MFFF.

The NRC staff has reviewed this information and finds that it appears sufficiently complete, and the quantities appear consistent with the proposed activities and the amounts necessary for safety assessments. The staff finds that this information is consistent with the guidance provided in Acceptance Criterion 8.4.3.1-F of NUREG-1718 (NRC, 2000); therefore, the information is adequate and acceptable.

8.1.2.1.7 Chemical Process Limits

The applicant provided information on chemical process limits in Section 5.2.5.4 of the LA (MOX, 2009a) and Section 5.1.2.7.3 of the ISA Summary (MOX, 2009b).

The applicant performed a determination of setpoints for IROFS in accordance with the provisions of International Society of Automation (ISA) Standard S67.04.01-2000 (ISA, 2000) and NRC Regulatory Guide 1.105, “Setpoints for Safety Related Instrumentation,” issued December 1999 (NRC, 1999).

The applicant indicated that safety limits for engineered and administrative IROFS were established in safety documents, such as nuclear safety evaluations (NSEs). From these safety limits, analytical limits were established to account for process system dynamics and transient behaviors. The analytical limits provide for a margin between the safety limits and the process response following activation of a protective response. From the analytical limits, the setpoints were established by analysis to account for effects of the measurement and response systems. The setpoints provide for an additional margin between the analytical limits and the protective response and include consideration of instrumentation drift and uncertainty.

The applicant established operating limits to provide for a sufficient margin between the established setpoints and normal process conditions. The applicant also developed operating procedures to implement operating limits and control operations to ensure that safety limits are not exceeded.

The applicant’s definitions of the related setpoint terms follow. Figure 8.1.2.1.7-1 illustrates the relationship of these terms to one another.

Safety Limit: Safety limits were chosen to maintain the integrity of the physical barriers that protect against events that could cause the performance criteria of 10 CFR 70.61 to be exceeded, which include criteria for radiological and chemical releases and protection against criticality events. These limits can be defined in terms of directly measured process variables or in terms of a calculated variable involving two or more measured process variables. The applicant lists safety limits in Section 5 of the ISA Summary (MOX, 2009b). The safety limits were established in safety documents, such as NSEs.

Analytical Limit: The analytical limit is a limit of a measured or calculated variable established by the safety analysis to ensure that a safety limit is not exceeded. The safety analysis established an analytical limit in terms of a measured or calculated variable and a specific time after that value is reached to begin protective action. The analysis accounted for the dynamic or transient nature of certain process variables and ensures that these variables do not exceed the safety limit as a result of this transient behavior.

Trip Setpoint: The setpoint is a predetermined value for actuation of the final setpoint device to initiate a protective action. An allowance is provided between the setpoint and the analytical limit to ensure an actuation before the analytical limit is reached. This allowance was used to account for instrument uncertainties and response times not accounted for in the analytical limit. For its determination of setpoints for IROFS, MOX Services committed to complying with the provisions of ISA Standard S67.04.01-2000 (ISA, 2000) and Regulatory Guide 1.105 (NRC, 1999).

Operating Limits: The operating limit is a limiting value (or range of values) for a process parameter within which plant operators normally operate the facility. This value was established to minimize challenges to IROFS. The operating limit determines the threshold for operability for a system of the facility. Operating limits were based on experience at the reference plants (i.e., the La Hague and MELOX facilities in France) and were produced in collaboration with system engineers, safety engineers, instrument and control engineers, and plant operations staff.

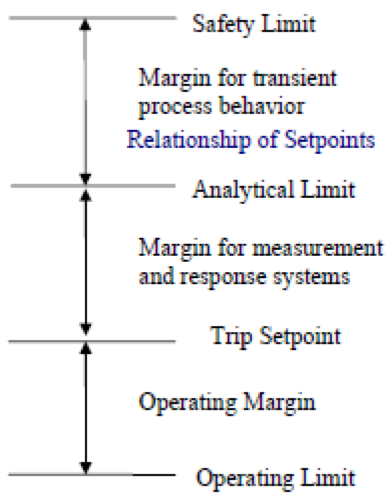

Figure 8.1.2.1.7-1 Relationships of process limits and setpoints

The applicant provided safety and operating limits for controlled process parameters in Table 5.3.6-9 of the ISA Summary (MOX, 2009b). The staff reviewed these limits and found them to be consistent with the methodology described above. The staff finds the applicant's description to be consistent with guidance provided in Acceptance Criterion 8.4.3.1-G in NUREG-1718 (NRC, 2000); thus, the description is acceptable.

8.1.2.2 Hazardous Chemicals and Potential Interactions

This section discusses the evaluation of potential chemical interactions to identify those chemicals that cannot be mixed under specified conditions and those mixtures that could create a safety hazard (e.g., a fire or explosion) and a release of radioactive material. Potential adverse reactions between the reagents used in the AP and MP processes were examined. In addition, interactions between the reagents and plutonium and uranium were examined to identify possible hazards related to colloids formation, polymerization of plutonium, precipitate formation, or explosion. Furthermore, interaction between the reagents and water was assessed. Finally, interactions between the reagents used in the AP process and those used in the MP process or as oxygen scavengers was investigated for possible hazards.

8.1.2.2.1 Chemicals

Sections 11.1 and 11.2 of this SER describe the chemical processes that take place as a part of normal operations at the MFFF.

The applicant generated chemical interaction matrices for AP and MP reagents (MOX Services, 2009a) to evaluate possible chemical interactions and the appropriate controls required on process parameters to ensure that hazardous interactions are prevented or mitigated. The applicant postulated that these chemicals are mixed either by failure of operations or equipment within the AP process itself or result from an inadvertent mixing by a technician in the reagent processing building (BRP) or MP area laboratories. The next section of this SER describes chemical interactions in more detail.

8.1.2.2.2 Chemical Interactions

Human error or equipment malfunction may result in inadvertent chemical interactions and initiate hazardous reactions. The applicant identified hazardous chemical characteristics and incompatibilities with the associated materials or process conditions, which are summarized in Tables 8.2-1 through 8.2-4 of the LA (MOX, 2009a) and Tables 5.1.3-4 through 5.1.3-7 of the ISA Summary (MOX, 2009b). The staff concluded that this list was consistent with standard interaction tables used by AICHE (AICHE, 2008). The applicant conducted hazard and operability analyses as part of the ISA during the detailed design phase and has submitted a complete chemical interaction matrix as part of the LA. The applicant has committed to controlling chemical preparation in accordance with operating procedures by using trained personnel to minimize the potential for unexpected interactions. To minimize the risk associated with inadvertent chemical interactions, the applicant will prepare most chemical reagents for the AP process in the BRP (a nonradiological building), with subsequent distribution to the AP area.

Sections 8.4.3.2(A), (B), and (C) of NUREG-1718 (NRC, 2000) list the acceptance criteria for potential chemical interactions. The staff notes that the applicant (1) identified hazardous chemicals expected to be used or produced at the facility, (2) identified general precautions and multiple specific hazards associated with the interaction of chemicals, and (3) considered radiolysis effects. Specific hazards include red oil (solvent) reactions, HAN/hydrazine autocatalytic reactions, azides decomposition, Pu (VI) oxalate explosions, and hydrogen deflagrations and explosions. The staff finds that the overall description of hazardous chemicals and potential interactions is consistent with guidance provided in Sections 8.4.3.2(A), (B), and (C) of NUREG-1718 (NRC, 2000), and is generally adequate and acceptable. SER Sections 8.1.2.4, 11.1, and 11.2 discuss further the significant specific chemical-related risks.

The applicant's ISA included an analysis of the potential for explosions and the IROFS that are required to prevent these events. In addition, the applicant identified events involving chemical releases, alone or in combination with radioactive releases. The applicant also identified IROFS to protect against these chemical risks at the MFFF.

Normal process conditions do allow for interactions of some chemicals identified as incompatible, provided that process parameters are controlled to allow safe operating conditions. For chemicals identified as incompatible that are mixed under normal process conditions, the applicant identified the following conditions that are controlled as necessary to maintain safe operating conditions:

- Aluminum nitrate—The low concentrations of aluminum nitrate (1 gram per liter (g/L)) used in the AP process are compatible with nitric acid.

- Diluent—For the chemical incompatibility between the diluent and oxygen, Section 8.3.3.6 of the LA (MOX, 2009a) discusses IROFS controls to ensure that either (1) all vapor compositions in the AP process are maintained at or below 60 percent of the lower flammability limit (LFL) or (2) by application of purge gas, effluent gas remains out of the flammability range throughout the system being protected.

- Dinitrogen tetroxide (equivalent entry for nitrogen dioxide)—Concentration and nitreous (NOx) flow controls in KPA*CLMN6000 allow controlled (i.e., intentional) HAN and hydrazine destruction by NOx, with adequate venting capacity provided for off-gases that are generated.

- Hydrazine monohydrate—Concentration and temperature controls ([N_2H_4] < 0.14 moles per liter (M); [HNO_3] < 1.75 normal (N); temperature < 55 degrees Celsius (C) or 50.6 degrees C ensure that hydrazine and nitric acid can coexist with minimal reactivity such that safe operating conditions are maintained.

- Hydrogen peroxide—Concentration controls ([H_2O_2] < 10 weight percent (wt%); [HNO_3] < 8 N) in KDD/KDB*TK3000 allow for the safe use of hydrogen peroxide to reduce Ag (II) to Ag (I) and Pu (VI) to Pu (IV) in the presence of nitric acid.

- HAN—Concentration and temperature controls ([HAN] < 0.8 M; [HNO_3] < 1.75 N; temperature < 55 degrees C or 50.6 degrees C) are employed to limit the depletion of HAN by nitric acid to prevent a HAN autocatalytic reaction.

- Nitric acid—The concentration of degradation products from reactions of organic compounds such as TBP and hydrogenated polypropylene tetramer (HPT or diluent) by nitric acid is limited to safe levels during normal process conditions by processing spent solvent through the solvent recovery unit, which removes solvent degradation compounds from the AP process.

- Oxalic acid—Manganese nitrate is used to catalyze the destruction of oxalic acid by nitric acid in the oxalic mother liquor recovery (KCD) unit.

- Silver nitrate—Silver nitrate is stable in acidic solutions in the absence of strong reducing agents, which are not used in AP process vessels that contain silver nitrate. The reduction of silver nitrate to silver metal by hydrogen peroxide is extremely slow and poses no safety hazard in KDB*TK3000, KDD*TK3000, and KDD*TK4000.

- Sodium carbonate—Adequate vent capacity for vessels KPB*MIXS1000 and KWD*TK4015 allow for safe neutralization reactions with nitric acid.

- Sodium hydroxide—Adequate vent capacity for vessels KPB*MIXS1000 and KWD*TK4015 allow for safe neutralization reactions with nitric acid.

- Sodium nitrite—Flow controls of nitric acid into KWD*TK4015, combined with adequate vent capacity for this vessel, allows for safe acidification of sodium nitrite into nitrous acid for the purpose of azide destruction.

- Sodium sulfite—Flow control of sodium sulfite allows for controlled reduction of chlorine (Cl_2) in KDD*CLMN7000/8000.

- TBP—The hydrolytic degradation of TBP in alkaline solutions used for solvent washing in the solvent recovery (KPB) unit is mitigated by the separation of TBP from the alkaline stream (i.e., contact between TBP and sodium hydroxide is limited by solvent separation).

In Section 8.2 of the LA (MOX, 2009a) and Section 5.3 of the ISA Summary (MOX, 2009b), the applicant provided chemical interaction matrices describing potential interactions between chemical reagents present in the AP and MP processes, interactions between chemical reagents present in the AP process and actinides and water, chemical interactions between species (mainly oxidizing) that may be generated in situ within the AP process, and chemical interactions between chemicals used in the MP process and steam supply unit.

The staff has reviewed this information and finds that it appears sufficiently complete and is consistent with the proposed activities and adequate for safety assessments. The staff finds that the information provided is consistent with guidance in Section 8.4.3.2(B) of NUREG-1718 (NRC, 2000) and follows AICHE standards and; therefore, the staff finds that this information is adequate and acceptable.

8.1.2.2.3 Unusual and Unexpected Reactions

The applicant stated that, in the chemical conditions encountered in the plutonium-uranium extraction (PUREX)-based process at the reference La Hague facility, chemical incompatibilities between the reagents have been mitigated or prevented through the control of process parameters. Tables 8.2-1 through 8.2-4 of the LA (MOX, 2009a) and Tables 5.1.3-4 through 5.1.3-7 of the ISA Summary (MOX, 2009b) present chemical interaction matrices created as a part of the applicant’s ISA to assess the chemical compatibility or incompatibility of the reagents that are postulated to be mixed either by failure of operations or equipment within the AP process itself or as a result of an inadvertent mixing by a technician in the BRP or BMP laboratories.

Some AP operations produce additional chemical compounds. The applicant stated that the behavior of these mixtures is well understood from experience at the La Hague facility in France and is included in the chemical process safety evaluation.

In general, for vapor and gaseous species, AP chemical interactions produce nitrogen oxides, carbon dioxide, carbon monoxide, and hydrogen, as well as possible plutonium, americium, and uranium mixtures entrained in nitric acid vapors. These chemicals will be generated in tanks and equipment and will be collected by the off-gas treatment system (KWG). Solvent-diluent vapors will also be collected in a separate stream and treated by the off-gas system before being released to the stack.

The staff has reviewed the information provided for unusual and unexpected reactions and finds that it appears sufficiently complete and is consistent with the proposed activities and adequate for safety assessments. The staff finds that the information provided is consistent with guidance in Section 8.4.3.2(C) of NUREG-1718 (NRC, 2000); therefore, the staff finds that this information is adequate and acceptable.

SER Section 8.1.2.4 discusses process safety information and specific concerns in more detail, and SER Sections 11.1 and 11.2 discuss specific chemical concerns by process unit. The staff findings are presented in those sections.

8.1.2.3 Chemical Accident Sequences

The applicant provided information on chemical-related events in Sections 5.5 and 8.3 of the LA (MOX, 2009a) and Section 5.3 of the ISA Summary (MOX, 2009b).

This section presents the staff's assessment of accident sequence bases, unmitigated sequences, estimated concentrations, and concentration limits for chemical safety. This section also provides the methodology and results for the evaluation of chemical consequences that may be associated with a release of radiochemical materials.

As reflected in 10 CFR 70.64(a)(5), the design must provide adequate protection from the following:

- chemical risks produced from licensed radioactive material
- facility conditions which affect the safety of licensed material
- hazardous chemicals produced from licensed materials

According to guidance provided in NUREG-1718 (NRC, 2000), the description of the foregoing chemical accident sequences should identify unmitigated accident sequences that could result in high or intermediate consequences to an individual. The applicant should also identify the standards used to establish the concentration limits and the methods used to assess the severity level of the accident sequence.

Subsequent sections present staff evaluations and conclusions.

8.1.2.3.1 Chemical Consequence Limits

<u>Concentrations Limits</u>

As required by 10 CFR 70.65, the applicant must provide a description of the proposed quantitative standards used to assess the consequences to an individual from acute chemical exposure to licensed material or chemicals produced from license materials. The applicant has adopted Revision 18 of the Temporary Emergency Exposure Limits (TEELs) to categorize exposures in accordance with the qualitative criteria established in 10 CFR 70.61. The following are the TEEL definitions:

- TEEL-1 is the maximum concentration in air below which it is believed nearly all individuals could be exposed without experiencing other than mild transient adverse health effects or perceiving a clearly defined objectionable odor.

- TEEL-2 is the maximum concentration in air below which it is believed nearly all individuals could be exposed without experiencing or developing irreversible or other serious health effects or symptoms that could impair their abilities to take protective action.

- TEEL-3 is the maximum concentration in air below which it is believed nearly all individuals could be exposed without experiencing or developing life-threatening health effects.

The DOE Subcommittee on Consequence Assessment and Protective Action (SCAPA) developed the TEELs because acute exposure guideline levels (AEGLs) and emergency response planning guidelines (ERPGs) exist only for a limited number of chemicals. SCAPA created TEELs so that DOE facilities, the U.S. Department of Defense, and some other Government agencies could conduct appropriate emergency preparedness hazard analysis and perform consequence assessments for the thousands of chemicals lacking AEGLs and ERPGs. TEELs are considered temporary; they are approximations of potential values and are subject to change whenever new or better information becomes available.

AEGLs and ERPGs are available for chlorine, hydrazine, nitric acid, and nitrogen dioxide, which are chemicals of concern in the AP process.[6] Since TEELs are considered temporary and NUREG-1718 (NRC, 2000) recommends the use of AEGLs or ERPGs to establish the chemical concentration limits,[7] the staff compared AEGLs and ERPGs values to the corresponding TEELs for the chemicals of concern in the AP process. The staff notes that TEEL values for those chemicals are comparable to corresponding AEGL and ERPG values.

The NRC does not promulgate its own chemical consequence limits, but relies on published standards developed by other Government agencies (e.g., OSHA, National Institute for Occupational Safety & Health (NIOSH)). Since the TEELs are standards developed by DOE for emergency preparedness, and given that they are comparable to AEGLs and ERPGs, the staff finds the use of TEELs to assess chemical consequences acceptable. The information provided is consistent with guidance in Section 8.4.3.3(D) of NUREG-1718 (NRC, 2000); therefore, the staff finds that this information is adequate and acceptable.

Consequences Categorization

Based on the provision of 10 CFR 70.61, three severity levels, High (H), Intermediate (I), and Low (L), are used to define the chemical concentration limits. Table 8.1.2.3.1-1 identifies the chemical consequence categories used to define the level of risk. Note that for uranium accidents, intakes are used, instead of concentration-based TEELs, to establish consequence categories. In most cases, events involving an airborne release of plutonium or americium are considered to have high consequences to the facility worker and IROFS are applied by the applicant to reduce their likelihood.

Table 8.1.2.3.1-1 Application of Chemical Limits to Qualitative Chemical Consequence Categories

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6 The applicant chose the chemicals of concern in the AP process based on maximum inventory, location, TEEL/ERPG values, and chemical properties.

7 The NRC does not promulgate its own chemical consequences limits but relies on values from other Government agencies and organizations that have a clear toxicological and regulatory basis.

Consequence Category	Worker (100 m)	IOC (160 m)
High	Concentration ≥ TEEL-3	Concentration ≥ TEEL-2 Soluble uranium intake ≥ 30 mg Insoluble uranium respirable intake ≥ 30 mg
Intermediate	TEEL-3 ≥ Concentration ≥ TEEL-2 Soluble uranium intake ≥ 30 mg Insoluble uranium respirable intake ≥ 30 mg	TEEL-2 ≥ Concentration ≥ TEEL-1 30 mg > Soluble uranium intake > 1 mg 30 mg > insoluble uranium respirable intake > 10 mg
Low	TEEL-2 > Concentration 30 mg > Soluble uranium intake > 1 mg 30 mg > insoluble uranium respirable intake > 10 mg	TEEL-1 > Concentration Soluble uranium intake < 10 mg Insoluble uranium respirable intake < 10 mg

The staff concludes that the use of TEELs to establish chemical consequence categories is acceptable for addressing the applicable 10 CFR 70.61 performance requirements for chemical protection. Furthermore, the information provided is consistent with guidance in Section 5.4.3.2(vi) of NUREG-1718 (NRC, 2000); therefore, the staff finds that this information is adequate and acceptable.

<u>Techniques, Assumptions, and Models</u>

The applicant's hazard analysis considered many different types of events that could cause an adverse human health or environmental effect as a result of accidental exposure to chemical sources. These types of accidents are categorized into the major events of fires, explosions, loss of confinement, load drops, and nuclear criticality. Chapters 5 and 8 of this SER provide the staff's evaluation of the applicant's hazard assessment.

The applicant calculated a bounding consequence for each identified event in the hazard evaluation. For the analysis, the facility worker is considered to be inside the MFFF, close to the potential accident, whereas the site worker is considered to be 100 meters from the release point. The individual outside the control area (IOC), as defined in the ISA, is the maximally exposed individual outside the controlled area boundary, either 68 meters (for BSW releases) or 160 meters (for MFFF building stack releases) from the release point. The applicant determined the consequences for the IOC and the facility worker based on the material released, the release mechanism, and the location of the worker relative to the release. MOX Services considered a range of initial conditions, as well as the failure modes of storage containers and associated systems. The following release mechanisms are stated in the reviewed LA (MOX, 2009a):

- leaks and ruptures involving equipment vessels and piping
- evaporating pools formed by spills and tank failures
- flashing and evaporating liquefied gases from pressurized storage

For the chemical consequences analysis, the applicant assumed the largest credible unmitigated spill or loss of containment accident. That is, the applicant modeled releases using the total material at risk from the largest single tank or container. No credit was taken for process equipment installed to scrub and remove gases and vapors of the potentially released chemicals before release from the MFFF.

The applicant used the largest evaporation rate calculated from two models as input to the Atmospheric Relative Concentrations in Building Wakes (ARCON) 96 computer code. The models are the following:

$$E = A \cdot K_M \cdot \frac{MW_M \cdot P_V}{R \cdot T} \qquad \text{(8.1.2.3.1-1)}^{8}$$

Where:

E = evaporation rate (kilogram per second (kg/s))
A = area of the evaporation puddle (square meter (m^2))
K_M = mass transfer coefficient (meter per second (m/s))= $0.0048 \cdot U^{7/9} \cdot Z^{-1/9} \cdot Sc^{-2/3}$
U = wind speed (m/s)
Z = pool diameter in the along-wind direction (meter (m))
Sc = laminar Schmidt number (dimensionless)
MW_M = molecular weight of the material of interest (kilogram per kilomole (kg/kmol))
P_V = vapor pressure (pascal (Pa))
R = gas constant (8.314 joule per mole kelvin (J/mol-K))
T = ambient temperature (kelvin (K))

$$Q_0 = A_P \cdot K_g \cdot \frac{MW_M \cdot P_V}{R \cdot T_P} \qquad \text{(8.1.2.3.1-2) (NRC, 1998)}$$

Where:

Q_0 = evaporation rate (kg/s)
A_P = area of the pool (m^2)
K_g = mass transfer coefficient (m/s)= $D_m \cdot N_{sh}/d$
D_m = molecular diffusivity of the vapor in air (m^2/s)
d = effective diameter of the pool (m)
N_{sh} = Sherwood number (dimensionless)
T_P = temperature of the pool (K)

For the unmitigated releases, the applicant assumed to occur outdoors, with an air speed of 2.2 m/s. This is consistent with 95-percent "worst case" meteorological conditions at SRS and are conservative assumptions.

To validate the methods used by the applicant, the staff performed independent calculations to estimate chemical exposures to the site workers and IOC in case of a release. Using the evaporation rate model in NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," issued March 1998 (NRC, 1998), the staff calculated the evaporation rate for some of the chemicals used in the AP process. The calculated evaporation rate was input into the Areal Locations of Hazardous Atmospheres (ALOHA) code to model chemical releases in the AP process. ALOHA is an atmospheric dispersion model maintained by the Hazardous Material

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8 Kawamura, P.I., and D. Mackay, "The Evaporation of Volatile Liquids," *Journal of Hazardous Materials,* 15:343–364, 1987.

Division of the National Oceanic and Atmospheric Administration and the U.S. Environmental Protection Agency. This model is used primarily for the evaluation of the consequences of atmospheric releases of chemical species. ALOHA is widely used by the public, Government agencies (e.g., DOE), and the private sector. The staff assumed release of the entire inventory for the calculation. Since the applicant's assumptions are conservative, the staff used the same assumptions for the analysis. For the chemicals of concern within NRCs jurisdiction, the staff noticed that concentrations exceeding the TEEL-1 at 160 meters or the TEEL-2 at 100 meters will not result given the aforementioned assumptions. Based on its confirmatory analysis, the staff accepts the applicant's methods.

Health effects on the facility worker from chemical exposure resulting from releases of licensed materials containing plutonium and americium are assumed to be dominated by the radioactive dose; thus, prevention or mitigation of the radioactive consequences would also prevent or mitigate the chemical exposure.

The staff found that the techniques, assumptions, and models utilized by the applicant to estimate hazardous chemical concentrations are generally consistent with industry practice and generally follow the guidance on atmospheric and consequence modeling found in NUREG/CR-6410 (NRC, 1998). Consequently, the staff finds them to be acceptable.

8.1.2.3.2 Chemical Release Events

This section discusses chemical releases that could result in events that exceed low consequence criteria under 10 CFR 70.61. The two broad categories of these events include the following:

- CRE-1—Chemical releases that result in acute chemical exposures to an individual from licensed materials or hazardous chemicals produced from licensed materials.
- CRE-2—Chemical releases that may interfere with operator performance of a NRC required safety function or serve as an initiator to an event by disabling an operator.

8.1.2.3.2.1 <u>CRE-1—Chemical Releases That Result in Acute Chemical Exposures to an Individual from Licensed Materials or Hazardous Chemicals Produced from Licensed Materials</u>

The applicant evaluated chemical consequences resulting from a release of hazardous chemicals directly produced from the processing of licensed material and releases of licensed material under 10 CFR 70.61. Events identified as potentially resulting in intermediate or high chemical consequences are discussed as part of the event descriptions related to NPH-1 (SER Section 11.11.6.1), NPH-2 (SER Section 11.11.6.2), F-16 (SER Section 7.1.6.15), LH-15 (SER Section 11.6.5.14), LOC-3 (SER Section 11.7.5.1), LOC-4 (SER Section 11.7.5.2), LOC-11 (SER Section 11.7.5.4), LOC-13 (SER Section 11.7.5.5), and explosion events (SER Chapters 8 and 11).

8.1.2.3.2.2 <u>CRE-2—Chemical Releases That May Interfere with Operator Performance of a NRC Required Safety Function or Serve as an Initiator to an Event by Disabling an Operator</u>

The applicant evaluated those events that could result in a release of hazardous chemicals not subject to 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," that may interfere with operator performance of a NRC required safety function. The applicant identified no events in which operator performance of a NRC required safety function to preclude a release of licensed material could be impaired by an exposure to a chemical not subject to 10 CFR Part 70 or in which disabling of an operator by a chemical exposure could initiate a release of licensed material. Therefore, no IROFS are necessary for a release of hazardous chemicals not subject to 10 CFR Part 70.

Based on the staff's review of the LA (MOX, 2009a) and ISA Summary (MOX, 2009b), the staff agrees with this finding and concludes that chemical releases that may interfere with operator performance of a required safety function or serve as an initiator to an event by disabling an operator cannot cause consequences in excess of low as defined by 10 CFR 70.61. Furthermore, the staff finds that the provisions of 10 CFR 70.61 are met for this event.

8.1.2.4 Process Safety Controls

8.1.2.4.1 Safety Strategies for Events Involving Hydrogen (EXP01 and EXP03)

8.1.2.4.1.1 <u>Argon-Hydrogen Mixture in Sintering Furnace and Hydrogen Storage (EXP01)</u>

Sintering furnace hydrogen explosions (EXP01) are considered credible in the pellets processing area where hydrogen explosions may occur in the high temperatures of the furnaces that sinter MOX fuel pellets. The applicant identified five explosion scenarios involving hydrogen in the sintering furnaces which could lead to a release of radioactive material. The applicant's safety strategy to prevent hydrogen explosions in the sintering furnaces is the application of IROFS to meet the performance criteria of 10 CFR 70.61.

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Staff Evaluation and Findings

The NRC staff finds that the proposed IROFS are adequate to comply with the single-failure criterion because each active engineered IROFS is redundant and fail-safe. Also, the applicant committed to provide adequate IROFS separation and to design the IROFS necessary to safely shut down the sintering furnace to be available during and after a seismic event. The single-failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (described in the MPQAP), and the use of codes and standards for engineered IROFS give the NRC staff reasonable assurance that these high consequence scenarios are highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61.

8.1.2.4.1.2 Radiolytic Hydrogen Production (EXP03)

Event Description

Radiolysis is the dissociation of molecules that can lead to gas generation. It occurs when organic and aqueous fluids are irradiated, as in the case of the MFFF, by plutonium and americium. Since the organic and aqueous fluids are hydrogenous substances, the generated gas of concern is hydrogen. Hydrogen gas can build up in the vapor spaces of tanks and vessels. If an overpressurization occurs or if the concentration of the flammable gas exceeds the LFL, there is a risk for a radiolysis-induced explosion which can result in the release of licensed material. There is also a risk of radiolysis in the waste-handling system because of the confinement of radioactive material.

For the analysis of this explosion event, the applicant assumed hydrogen to be the only gas generated from radiolysis. The applicant has relied on National Fire Protection Association (NFPA) 69, "Standard on Explosion Prevention Systems" (NFPA, 2008), to set the bounding values for hydrogen concentrations. According to NFPA 69, in the absence of automatic controls, the maximum hydrogen concentration allowed must not exceed 25 percent of its LFL, which corresponds to a value of 1 percent for this analyzed event. The hydrogen LFL is 4 percent. The 1-percent maximum hydrogen concentration provides an adequate safety margin between the hydrogen explosive limit and the maximum amount of hydrogen allowed in any tank or vessel since it meets NFPA 69.

Safety Strategy

The safety strategy for this event is prevention in both the AP process, and the waste-handling system. The applicant's approach for prevention in the AP process equipment involves the dilution or inerting of the vapor space, the monitoring and control of material at risk (MAR), the control of the vapor space, and the detection and recovery from a loss of instrument air before reaching an explosive limit. The prevention safety strategy in the waste-handling system requires the containers to be designed such that hydrogen buildup in excess of the explosion limits does not occur, while providing appropriate confinement of radioactive material.

Within the AP process, the applicant identified two types of vessels for this explosion event. In the first type of vessel, dilution (or scavenging) air is continuously provided; in the other type of vessel, dilution air is not provided. Dilution air is not provided to a vessel if the hydrogen concentration will not reach 1 percent in the vapor space within 11 days without instrument air being supplied. Otherwise, all other tanks and vessels, in which a hydrogen concentration of 1 percent or more could be reached within 11 days, are supplied continuously with dilution air.

Items Relied on for Safety

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Staff Evaluation and Findings

The NRC staff evaluated the applicant's calculation for determining the minimum and actual dilution airflows to the vessels in which the hydrogen concentration will reach 1 percent or greater within 11 days. The applicant's calculation is based on the type of solution present (i.e., organic or aqueous) and the maximum credible MAR amount in the vessel of interest. After determining the minimum dilution airflow, the applicant rounded this value to the nearest 50 NL/h and then multiplied the resulting value by a factor of 1.25. The NRC staff finds that this calculation provides reasonable assurance of safety based on the added safety margins the applicant put in place to determine the actual dilution airflow.

The credited IROFS for each of the types of vessels discussed above provide a diverse method to control each of the parameters that have an effect in the generation and accumulation of hydrogen. In addition, the applicant has committed to have fail-safe and redundant active engineered IROFS (i.e., radiation detectors and isolation valves) and use them in combination with passive engineered IROFS (i.e., vessel overflows and material of waste containers) and administrative IROFS (i.e., sampling and detection of and recovery from loss of instrument air).

The NRC staff finds that this is an acceptable approach to complying with the single-failure criterion. The single-failure criterion, in combination with the experience in La Hague, management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS, give the NRC staff reasonable assurance that this high consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61.

The applicant also uses features to reduce the challenge to IROFS where practical. [redacted] The NRC staff finds that these features comply with the defense-in-depth requirements of 10 CFR 70.64(b).

[redacted]

[redacted]

8.1.2.4.1.3 Electrolysis-Related Explosions (EXP13 and EXP17)

Electrolytically Generated Hydrogen Explosions (EXP13)

Electrolysis-related explosions are postulated to occur from hydrogen (H_2) that may be generated electrochemically at the cathode of the electrolyzer. The applicant's safety strategy for electrolysis-related explosions involves the application of IROFS to meet the performance criteria of 10 CFR 70.61.

[redacted]

[redacted] [redacted]

Event Description

[redacted]

[redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted]

[redacted]

[redacted]

The applicant's safety strategy for this event is to control the normality of the catholyte side of the electrolyzer during electrolysis and to verify flow through the electrolysis cell to ensure that the normality determined is representative of the solution in the cell. [redacted]

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[redacted]

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[redacted]

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[redacted]

[redacted]

[redacted]

[redacted]

Staff Evaluation and Findings

The credited IROFS discussed above provide a diverse method to control each of the parameters that have an effect on the generation and potential ignition of hydrogen in the electrolyzers. The applicant has also committed to have fail-safe and redundant active engineered IROFS (e.g., normality controllers, process-level controls, and catholyte flow monitoring) and use them in combination with passive engineered IROFS (e.g., process vessels (membrane) and pipes).

The NRC staff finds that this is an acceptable approach to complying with the single-failure criterion. The single-failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (described in the MPQAP), and the use of codes and standards for engineered IROFS give the NRC staff reasonable assurance that these high consequence scenarios are highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61.

The applicant also uses defense-in-depth features to reduce the challenge to IROFS, where practical. [redacted] The NRC staff finds that these features comply with the defense-in-depth requirements of 10 CFR 70.64(b).

Perchlorate Explosions (EXP16h and EXP17)

Explosion in KDD Scrubbing Columns Caused by Higher Oxides of Chlorine (EXP16h)

Chlorine removal from the electrolyzer is provided by the oxidation of chloride ions (Cl^-) at the anode to form molecular chlorine (Cl_2, a gas which is swept through a series of filters and injected through the Venturi injector scrubber into the recirculating tank. At this stage, Cl_2 is reverted to chloride ion using a 0.5 M sodium sulfite (Na_2SO_3) scrubbing solution that is continuously recirculated through the packed scrubbing column to ensure maximum contact with the off-gases, and maintained at a high pH with sodium hydroxide (NaOH) to prevent corrosion and optimize sulfite reduction of chlorine. An inline sulfite analyzer indicates when scrubbing is complete; the spent solution is then transferred to the first KDD chlorine waste tank and on to the second KDD chlorine waste tank for sampling and dilution before transfer to the low-level liquid waste unit (KWD). Residual unreacted chlorine is scavenged through a demister before being discharged to the VHD unit.

However, alkaline conditions (pH 12) in the scrubber are also potentially favorable for chlorine to oxidize to higher oxidized species. These higher oxides of chlorine are strong oxidants that can react exothermically or explosively. Although these reactions do not generate gaseous products, and organic solvents are not present in the scrubber, energy releases from uncontrolled reactions could lead to pressurization of chlorine gas. This explosion would not directly involve any radioactive material that could exceed the performance requirements of 10 CFR 70.61. However, the explosion event could damage nearby IROFS components in the vicinity of the electrolyser and may indirectly create radiological and chemical consequences to the facility, radiological consequences to site workers and the IOC, a release to the environment that would exceed the performance requirements of 10 CFR 70.61, or some combination of these consequences based on the damage from an explosion.

Event Description

In the scrubber, chlorine gas is reduced by sulfite ion according to the oxidation-reduction (redox) reaction:

$$2\,OH^- + SO_3^{2-} + Cl_2 \rightarrow SO_4^{2-} + 2\,Cl^- + H_2O \qquad (8.1.2.4.1.3\text{-}5)$$

However, in the alkaline conditions (pH 12) of the scrubber, chlorine can undergo a competing disproportionation redox reaction (that does not involve sulfite), through direct hydrolysis with hydroxide ion:

$$2\ OH^- + Cl_2 \rightarrow OCl^- + Cl^- + H_2O \qquad (8.1.2.4.1.3\text{-}6)$$

The hypochlorite ion (OCl^-) formed in this reaction is an unstable species that is prone to subsequent disproportionation redox reactions leading to the formation of higher oxides of chlorine, namely chlorate (ClO_3^-) and perchlorate (ClO_4^-):

$$3\ OCl^- \rightarrow 2\ Cl^- + ClO_3^- \qquad (8.1.2.4.1.3\text{-}7)$$

$$4\ ClO_3^- \rightarrow 3\ Cl^- + ClO_4^- \qquad (8.1.2.4.1.3\text{-}8)$$

However, the hypochlorite ion is also readily and quickly reduced by sulfite (Fogelman, et al., 1989):

$$OCl^- + SO_3^{2-} \rightarrow Cl^- + SO_4^{2-} \qquad (8.1.2.4.1.3\text{-}9)$$

Thus, the fate of any hypochlorite ion formed from equation (8.1.2.4.1.3-6) is determined by competing equations (8.1.2.4.1.3-7) and (8.1.2.4.1.3-9). In the absence of the sulfite ion, equation (8.1.2.4.1.3-7) eventually leads to the formation of chlorate and perchlorate ions. These higher oxidized species are strong oxidizing agents that undergo exothermic reactions with reducing agents such as sulfite.

[redacted]

To minimize the potential for explosive conditions in the scrubber, the applicant's safety strategy is to prevent the formation of higher oxides of chlorine in this equipment. [redacted]

<u>Items Relied on for Safety</u>

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Staff Evaluation and Findings

The staff finds that the applicant sufficiently demonstrated that no IROFS are necessary for this event scenario and that the defense-in-depth measures to be implemented by the applicant provide additional assurance that interactions with IROFS during this accident event are reasonably prevented. .

The NRC staff finds that this approach is reasonable and acceptable. Furthermore, the staff finds that these features comply with the defense-in-depth requirements of 10 CFR 70.64(b).

Electrolyzer-Related Perchlorate Explosion (EXP17)

Perchlorate explosions are postulated to occur from the oxidation of chlorine in the electrolyzer which is an IROFS. The applicant's safety strategy for explosions from perchlorate or other higher oxides of chlorine involves the application of IROFS to meet the performance criteria of 10 CFR 70.61.

[redacted]

[redacted]

[redacted]

Event Description

The expected chloride reaction on the anode during dechlorination is as follows:

$$2Cl^{-} \rightarrow Cl_2 + 2e^{-} \qquad (8.1.2.4.1.3\text{-}10)$$

However, other oxidation-reduction (redox) couples are possible at the anode. In the literature, standard oxidation potentials for various redox couples for higher oxides of chlorine under acidic or basic conditions can be found. Higher oxides of chlorine are a safety concern for two reasons: (1) material compatibility—various chlorates are highly corrosive and (2) explosion hazard—various forms of perchlorates are explosion hazards (e.g., potassium perchlorate, ammonium perchlorate). The couple in the above reaction has the lowest potential and will be the thermodynamically preferred reaction. However, the difference between the redox potential for this reaction and those of other reactions (e.g., perchlorate) is rather small, and, in the absence of kinetic considerations, some of the other redox reactions could occur in the electrolyzer.

Considerable kinetic barriers are associated with these other redox reactions which require a large overpotential that precludes the formation of higher oxides of chlorine at the anode. As a result, unless a large current density is applied, conditions favorable to formation of higher oxides of chlorine are restricted. Under acidic conditions (like those found in the electrolyzer), the first step involves the hydrolysis of chlorine (Cl_2) leading to a disproportionation redox reaction and subsequent formation of hypochlorous acid (HOCl):

$$H_2O + Cl_2 \rightarrow HOCl + Cl^- + H^+ \qquad (8.1.2.4.1.3\text{-}11)$$

Higher oxides of chlorine such as chlorate (ClO_3^-) and perchlorate (ClO_4^-) can be generated in solution from HOCl, and its conjugate base hypochlorite (OCl^-), through a series of disproportionation redox reactions:

$$2\,HOCl + OCl^- \rightarrow 2\,Cl^- + 2\,H^+ + ClO_3^- \qquad (8.1.2.4.1.3\text{-}12)$$

$$4\,ClO_3^- \rightarrow 3\,Cl^- + ClO_4^- \qquad (8.1.2.4.1.3\text{-}13)$$

One process upset condition involves the addition of silver nitrate ($AgNO_3$) before the removal of chloride from the process solution. This could occur if the dechlorination step is incomplete or if AFS material is introduced to the KDB unit (where a dechlorination step is not performed). The addition of Ag^+ to a solution containing Cl^- promotes silver chloride (AgCl) precipitation, but this is counteracted by an accompanying increase in ionic strength with silver nitrate addition that increases the solubility of silver chloride. Overall, this upset condition would result in a small amount of soluble chloride that could complicate the subsequent electrolysis of Ag^+ but would have little effect on the dynamics with respect to higher oxides of chlorine.

If higher oxides of chlorine were to reach the KPA unit, exothermic reactions could occur with the organic solvent in the extraction pulsed column. These reactions could generate potentially explosive gases and raise the solution temperature such that the LFL for the solvent would be exceeded. In addition, higher oxides of chlorine continuing downstream to the raffinate tanks and to the KPC evaporators could cause an explosion event.

[redacted]

The safety strategy for this event group is to control the quantity of higher oxides of chlorine that can be produced in an electrolyzer. [redacted]

[redacted]

[redacted]

Risk Discussion

The applicant indicated that a number of thermodynamic and kinetic factors constrain the formation of higher oxides of chlorine under the conditions encountered in the electrolyzer. First, reaction (8.1.2.4.1.3-10) becomes progressively disfavored thermodynamically as the acidity is increased, and little HOCl is produced under the highly acidic conditions that are nominally present in the electrolyzer. The low solubility of Cl_2 in acidic aqueous solutions, which is diminished further by the relatively warm operating temperature within the electrolyzer and non-IROFS pressure controls that maintain a low pressure in the vapor space above the anolyte surface, also limit HOCl production. Additionally, formation of higher oxides of chlorine is largely prohibited at this pH by (1) the negligible concentration of hypochlorite (very little of the weak acid HOCl is dissociated to its conjugate base OCl^-, where $[HOCl]/[OCl^-] \sim 10^8$), which is required for chlorate formation according to reaction (8.1.2.4.1.3-12) and (2) thermodynamic factors that favor the reverse reaction in equation (8.1.2.4.1.3-12). The applicant indicated that alternative electrochemical pathways for chlorine, other than that shown in equation (8.1.2.4.1.3-10), are suppressed by the operating current density in the electrolyzer. Furthermore, a competing anodic reaction reverts HOCl back to chloride with the evolution of oxygen gas. Finally, any chlorate that may be generated is highly susceptible to rereduction by residual chloride in solution to give (via a chlorine dioxide intermediate) chlorine gas and, although further disproportionation of any remaining chlorate towards the formation of perchlorate (equation (8.1.2.4.1.3-13)) is thermodynamically favored, a large kinetic barrier keeps this reaction sluggish up to 100 degrees C.

Thus, under normal operating conditions, in which acidic conditions are maintained and the pressure at the surface of the anolyte is controlled (thus eliminating chlorine availability in solution), the formation of higher oxides of chlorine in a KDD or KDB electrolyzer is negligible. Additionally, the addition of hydrogen peroxide would reduce any higher oxides of chlorine into chloride and chlorine.

Items Relied on for Safety

[redacted]

[redacted] [redacted]

[redacted]

General Normality Controls

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

CEA Flowsheet Validation Testing

To validate the dechlorination and dissolution flowsheet in the MFFF AP process, the French Atomic Energy Commission, known as Commissariat a l’Energie Atomique (CEA), performed laboratory scale radioactive tests with weapons-grade plutonium oxide contaminated with chloride and other metallic impurities at the CEA Atalante Facility in Marcoule, France (Brossard, et al., 2003).

The CEA conducted this experimental program to determine any effect on process operations caused by the differences between the relatively pure weapons-grade plutonium dioxide (from the PDCF) and the lower grade plutonium oxide from the AFS.

The process used to purify the AFS PuO_2 is based on a Ag (II) catalyzed electrolytic dissolution followed by plutonium purification by liquid-liquid extraction.

The Ag (II) dissolution process for plutonium oxide has reached industrial maturity in France, which has led to its implementation for the dissolution of plutonium powders and for the treatment of various solid waste at an industrial scale in French facilities.

The CEA successfully tested a derivative process in the 1980s for the treatment of chlorinated ashes and in the 1990s for hydrometallurgical treatment of plutonium salt-bearing salt baths. That process included a milling step, a direct electrolytic dechlorination, and an electrolytic plutonium oxide dissolution (Brossard, et al., 2003).

The purification process is based on the purification flowsheets developed for the R4 facility in France (UP2 plant at La Hague). The CEA tested the following process operations:

- milling
- electrolytic dechlorination
- silver (II) catalyzed electrolytic dissolution
- plutonium purification by liquid-liquid extraction.

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

These CEA tests demonstrated the feasibility of quantitatively dissolving, by electrogenerated Ag (II), plutonium powders obtained from molten chloride salts processing operations, milled and calcined for 2 hours at 950 degrees C under air. For all of the tests performed, more than 99.6 percent of the plutonium which had been previously dechlorinated, with an efficiency greater than 99 percent, was dissolved in the nitric acid solution.

The liquid-liquid extraction tests were successfully conducted leading to a plutonium recovery close to 100 percent, with an impurity content below the limits set for the process.

These tests validated the process flowsheet for the MFFF plant to polish low-grade plutonium oxide from AFS.

Staff Evaluation and Findings

The credited IROFS discussed above provide a method to control each of the parameters that have an effect in the generation and accumulation of higher oxides of chlorine. The applicant committed to have fail-safe and redundant active engineered IROFS [redacted]

The NRC staff finds that this is an acceptable approach for complying with the single-failure criterion. The single-failure criterion, in combination with the experience in La Hague, experimental results at CEA facilities, management measures (as described in the Chapter15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS, give the NRC staff reasonable assurance that this high consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61.

The applicant also uses defense-in-depth features to reduce the challenge to IROFS, where practical. [redacted] The NRC staff finds that these features comply with the defense-in-depth requirements of 10 CFR 70.64(b).

[redacted]

- [redacted]

- [redacted]

- [redacted]

8.1.2.4.2 Solvent Explosions (EXP06)

The purification cycle and the solvent recovery cycle are two units in which solvent and diluent are used in processing. The purification cycle uses the solvent-diluent mixture for the extraction of plutonium. The solvent recovery cycle regenerates the solvent-diluent mixture by removing the degradation products and adjusting the TBP content and stores the treated solvent at a slightly acidic condition to prevent degradation by hydrolysis. The aqueous stream leaving the pulsed columns and mixer-settlers is washed with the diluent to remove traces of entrained solvent.

[redacted]

[redacted]

Event Description

The applicant indicated that this event is a process-related chemical explosion involving solvent in AP vessels, tanks, and piping in AP process cells or AP gloveboxes resulting in an breach of the AP vessels, tanks, and piping and the potentially a dispersal of radioactive material outside of the plutonium confinement zone.

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

The applicant's safety strategy for this event is preventive and involves inhibiting the creation of flammable or explosive vapors within the vessel headspaces and vessel venting systems as a result of HVAC temperature deviations by limiting HVAC supply air temperatures, controlling process solution temperatures, and inerting specified vessels under certain conditions.

<u>Fire Events Affecting TBP or HTP or Both (EXP06c)</u>

Fires external to process equipment have the potential to increase process temperatures such that a flammable or explosive mixture could be formed in the gaseous phase. [redacted]

<u>Events Involving Separate Phase Solvent (TBP and HTP) (EXP06d)</u>

The separate phase solvent is a layer of solvent over an aqueous solution, separated because of a density difference. Since TBP is slightly soluble in the aqueous solution and HTP is essentially immiscible in the aqueous solution, the aqueous solution will contain TBP at or near its solubility limit when a separate layer of solvent exists. [redacted]

<u>Events Involving Separate Phase HTP (EXP06e)</u>

In this event type, separate phase HTP is postulated as entering process vessels operating at temperatures in excess of the safety limit. [redacted]

[redacted]

<u>Events Involving Soluble TBP (EXP06f)</u>

During the solvent extraction of plutonium in the KPA unit, TBP extracts the plutonium from the aqueous phase into the organic phase, necessitating the commingling of the two phases. While HTP is essentially insoluble in the dilute aqueous nitric acid phase, TBP is partially soluble. [redacted]

[redacted]

[redacted]

Events Involving Solvent Degradation Products (EXP06g)

The solvent safety-basis temperature limit, which is established to prevent flammable vapors in the headspace of vessels containing solvent, depends upon the effective removal of degradation products, which could otherwise require the reduction of the temperature limit. [redacted]

Safety Strategy

The applicant's safety strategy for solvent explosions involves the application of IROFS to meet the performance criteria of 10 CFR 70.61. [redacted]

[redacted]

[redacted]

The applicant's safety strategy for this event is preventive and involves utilization of administrative controls to limit the accumulation of solvent degradation products that could otherwise cause the reduction of the flashpoint of solvent that has been permitted to remain in situ in the presence of plutonium for an extended period of time.

To ensure that the lower flammability limit is not reached, the applicant has committed to follow the guidelines of NFPA 69 (NFPA, 2008) for the processing of flammable liquids. Specifically, NFPA 69 requires that the combustible concentration be maintained at or below 25 percent of the LFL, except for cases in which automatic instrumentation with safety interlocks is provided. In these cases, the combustible concentration is permitted to be maintained at or below 60 percent of the LFL.

In addition, the applicant has stated that, in those cases in which automatic instrumentation cannot prevent the concentration from exceeding 60 percent of the flammable limit, purge gas at the point of use will be applied. NFPA 69 gives the following guidance for point-of-use purge gas:

- Purge gas shall be introduced and exhausted so that the distribution is ensured and the desired reduction in oxidant concentration is maintained throughout the system being protected.
- Instrumentation shall be provided to monitor the purge gas supplied to the distribution system.
- The oxygen concentration shall be checked on a regularly scheduled basis.

[redacted]

[redacted]

The equipment and systems which are covered by this event group include those portions of the AP process that either normally contain solvent (i.e., KPA, KPB, KWS, and LGF units) or abnormally contain solvent (i.e., KWG, KPC, KCA, KCD, and KWD units).

[redacted]

[redacted]

<u>Process Temperature Events Involving TBP or HTP or Both (EXP06a)</u>

[redacted]

[redacted] [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Room Temperature Events Involving TBP or HTP or Both (EXP06b)

[redacted]

[redacted]

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Events Involving Separate Phase HTP (EXP06e)

[redacted]

[redacted] [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Events Involving Soluble TBP (EXP06f)

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Events Involving Solvent Degradation Products (EXP06g)

[redacted]

[redacted]

[redacted]

[redacted]

Staff Evaluation and Findings

The credited IROFS discussed above provide a method to control each of the parameters that can contribute to solvent explosions in the AP process. The applicant has also committed to have fail-safe and redundant active engineered IROFS (e.g., density controllers and temperature controllers) and use them in combination with passive engineered IROFS (e.g., slab settler and process vessels and pipes) and administrative IROFS (e.g., stationing of firewatch and administrative controls to maintain sufficient buffer volume in the KWD HAW unit).

The NRC staff finds this to be an acceptable approach to complying with the single-failure criterion. The single-failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (described in the MPQAP), and the use of codes and standards for engineered IROFS give the NRC staff reasonable assurance that these high consequence scenarios are highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61.

The applicant also committed to use defense-in-depth features to reduce the challenge to IROFS, where practical. [redacted]

[redacted] The NRC staff finds that these features comply with the defense-in-depth requirements of 10 CFR 70.64(b).

The staff finds reasonable assurance that the applicant has identified the hazards and accident sequences associated with solvent explosions which could result in a release of radioactive material and credited IROFS are sufficient to meet the performance requirements of 10 CFR 70.61, consistent with the acceptance criteria of Sections 7.4 and 8.4 in NUREG-1718.

8.1.2.4.3 HAN and Hydrazine in Nitrate Media (EXPs 04, 10, and 11)

DOE has used HAN as a reductant in nuclear materials processing and for decontamination of equipment. British and French reprocessing plants have also used HAN as a reductant. In addition, the U.S. Army has used HAN as an oxidizer in propellant mixtures (DOE, 1998).

HAN was incorporated into many nuclear fuel reprocessing plants in the early 1970s for the reduction of Pu (IV) to Pu (III) in the plutonium-uranium extraction (PUREX) process. HAN generally replaced ferrous sulfamate and hydroxylamine sulfate (HAS) for plutonium reduction because it has the proper reduction potential for the Pu (IV) to Pu (III) reduction and its reaction products (molecular nitrogen (N_2), nitrous oxide (N_2O), and water (H_2O)) do not contribute to the volume of solid waste produced during reprocessing (DOE, 1998).

In the United States, HAN was the plutonium reductant of choice used at SRS and Hanford in various PUREX-type processes to recover plutonium. The British have employed HAN safely in the Thermal Oxide Reprocessing Plant (THORP) at Sellafield, where it is used in the main plutonium feed to the plutonium purification cycle and two scrub feeds to remove traces of plutonium from the uranium purification cycle. HAN was also used safely for over 10 years in counter-current flowsheet trials during process development for THORP (DOE, 1998). The French reprocessing plant at La Hague has used HAN for reductive stripping of plutonium for many years. The chemical is received at 1.9 M and subsequently diluted and mixed with nitric acid and hydrazine for use in the process (DOE, 1998).

<u>Accidents Involving HAN</u>

Numerous incidents involving uncontrolled HAN reactions have occurred at DOE's Hanford and Savannah River sites. For example, at the Hanford Site, an explosion occurred at the Plutonium Reclamation Facility (PRF) on May 14, 1997. A dilute solution of HAN and nitric acid was created in a 400-gallon stainless steel tank, in preparation for restart of the facility in 1993. However, the restart of the facility was cancelled in December of that year, and the tank was not subsequently drained. Over the following 4 years, the contents of the vented tank concentrated by a factor of approximately 25 as a result of evaporation (DOE, 1998). The higher concentration solution, the effect of iron (Fe) from the metal surfaces of the tank, and an increased ambient room temperature contributed to an autocatalytic reaction that resulted in an explosion. The explosion destroyed the tank and the chemical makeup room where it was located. It breached the facility room and created a toxic chemical release. A fire system pipe was ruptured, flooding the PRF and spreading low levels of plutonium contamination to the ground outside the facility. No one was injured by the explosion, but some personnel were reported to be exposed to the toxic fumes (DOE, 1998).

Also at Hanford, an exothermic chemical reaction involving HAN, nitric acid, and hydrazine occurred in a section of pipe containing 2BX solution at the PUREX plant on December 3, 1989. The overpressurization of the piping resulted in the failure of a flange gasket. There was also an event at the PRF in the 1970s in which Tank A-109 overpressurized when strong nitric acid was added to the tank. It was possible that the tank contained a heel of HAN.

At SRS, on December 28, 1996, high temperatures in a tank in F-Canyon containing a HAN-nitric acid solution caused an autocatalytic reaction resulting in the eructation of approximately 250 gallons of solution (DOE, 1998). A similar eructation event occurred on September 26, 1972, when improper startup temperatures in an evaporator caused the overconcentration of approximately 6,000 pounds of HAN and nitric acid.

Other HAN-related incidents prompted DOE to launch a study of the properties of solutions containing HAN and nitric acid. The study yielded a technical report entitled, “Technical Report on Hydroxylamine Nitrate,” which contains recommendations on the storage and use of HAN (DOE, 1998).

This report (DOE, 1998) states that the margin of safety for the use of HAN is defined by the control of chemical concentration and ratio of each reactant, temperature, pressure, and presence of catalysts (e.g., iron and plutonium). The report’s recommendations include the following:

- Incorporate conservative safety envelope limits into appropriate safety documents, standards, and procedures recognizing the uncertainties in available data.
- Passivate the surfaces of HAN-nitric acid solution tanks and piping.
- Store unused HAN in the original, sealed manufacturer’s shipping container. If only portions are used, avoid contamination of the material and reseal the container to preclude evaporation.
- Control the chemical makeup and addition system by (1) defining mixing sequences and controls, (2) making up only the amount required, (3) eliminating direct addition of concentrated acid, (4) maintaining chemicals within specification, (5) controlling heating conditions to process specifications, (6) draining and flushing the system to a neutral pH and refilling with water for extended downtimes, (7) confirming that tanks assumed operationally empty contain no heel and then draining and flushing, and (8) disposing of unneeded chemicals.
- Ensure an effective system pressure relief.
- Establish and maintain surveillance programs to ensure that the necessary controls continue to be in place.
- Train engineering and operating personnel on the potential hazards along with possible off-normal conditions and controls necessary to remain within safety limits.
- Evaluate the use of alternate reductants.

8.1.2.4.3.1 <u>HAN Decomposition and Explosions (EXP04)</u>

In the ISA Summary (MOX, 2009b), the applicant postulated a HAN explosion at the MFFF. This event is identified as EXP04. This explosion event is caused by an autocatalytic reaction between HAN and nitric acid (HNO_3) under certain conditions. The autocatalytic reaction can quickly produce an excessive amount of gases. If the vessel vents cannot accommodate the gas production, it can cause an overpressurization leading to the explosion.

HAN is used in the AP process to reduce Pu^{+4} to Pu^{+3}. The plutonium transfer from the organic phase to the aqueous phase results from this reduction in the plutonium valence. Additionally, HAN reacts with nitrous acid (HNO_2) (nitrous acid is always present in HNO_3 solutions), which is the key intermediate chemical to the initiation of the autocatalytic reaction. The following equations characterize the reactions of HAN:

$$2NH_3OH^+ + 2Pu^{+4} \rightarrow 2Pu^{+3} + N_{2\,(g)} + 2H_2O + 4H^+ \qquad (8.1.2.4.3.1\text{-}1)$$

$$NH_3OH^+ + 2HNO_3 \rightarrow 3HNO_2 + H^+ + H_2O \qquad (8.1.2.4.3.1\text{-}2)$$

$$NH_3OH^+ + HNO_2 \rightarrow N_2O_{\,(g)} + 2H_2O + H^+ \qquad (8.1.2.4.3.1\text{-}3)$$

equation (8.1.2.4.3.1-1) is the Pu^{+4} reduction reaction with HAN. By multiplying equation (8.1.2.4.3.1-3) by 3 and adding the result to equation (8.1.2.4.3.1-2), the following overall autocatalytic reaction between HAN and HNO_3 is obtained:

$$4NH_3OH^+ + 2HNO_3 \rightarrow 3N_2O_{(g)} + 4H^+ + 7H_2O \qquad (8.1.2.4.3.1\text{-}4)$$

The applicant uses hydrazine (N_2H_4) as a defense-in-depth strategy to minimize the reaction between HAN and HNO_2 (equation (8.1.2.4.3.1-3)), thereby, increasing the HAN available for plutonium reduction. Hydrazine can also reduce Pu^{+4} to Pu^{+3}. The hydrazine concentration has to be limited (less than 0.14 M) because of other safety issues (see Section 8.1.2.4.3.2 of this SER for more details).

The staff notes that energetic HAN-HNO_3 reactions can occur under the right conditions, as evidenced by a DOE investigation of an accident at Hanford (DOE, 1998). As a result of this explosion, DOE investigated the situation and concluded that the HAN phenomena involved the interdependence between at least the following five parameters (NRC, 2005):

1. chemical concentration of each reactant
2. molar ratio of nitric acid to HAN $\left(\frac{[HNO_3]}{[HAN]}\right)$
3. temperature of the mixture
4. concentration of metal ions (as catalysts)
5. pressure of the system (appears to influence the severity of the reactions, but not the initial autocatalytic initiation)

<u>Safety Strategy</u>

During the construction authorization request (CAR) stage, the applicant proposed a HAN safety strategy divided into two categories: (1) process vessels containing HAN and hydrazine nitrate without nitrous oxide addition and (2) process vessels containing HAN and hydrazine nitrate with nitrous oxide addition. Table 8.1.2.4.3.1-1 identifies the principal structures, systems, and components (PSSCs) for the first category; Table 8.1.2.4.3.1-2 identifies those for the second category.

Table 8.1.2.4.3.1-1 PSSCs for Process Units with HAN and without Nitrous Oxide Addition

PSSC	Safety Function	Controlled Parameter	Design Basis
Process Safety Control Subsystem	Maintain temperature below safe limits	Temperature	< 50 °C (122 °F)
Chemical Safety Controls	Maintain maximum nitric acid concentration	[HNO_3]	< 6 M
	Maintain minimum hydrazine concentration	[N_2H_4]	≥ 0.1 M
	Maintain maximum HAN concentration	[HAN]	< 2.5 M
	Limiting residence time of nitric acid, HAN, hydrazine with nitric acid, and plutonium-bearing solution	Time	as low as reasonable (probably several months)

Table 8.1.2.4.3.1-2 PSSCs for Process Units with HAN and Nitrous Oxide Addition

PSSC	Safety Function	Controlled Parameter	Design Basis
Off-Gas Treatment System	Exhaust path for removal of off-gases, which provides a means for heat transfer/pressure relief for affected process vessels	Heat transfer, pressure relief	Revised CAR Table 11.8-2
Chemical Safety Controls	Limit nitric acid concentration	[HNO_3]	Revised CAR Table 11.8-2
	Limit hydrazine concentration	[N_2H_4]	Revised CAR Table 11.8-2
	Limit HAN concentration	[HAN]	Revised CAR Table 11.8-2
	Limit hydrazoic acid concentration	[HN_3]	Revised CAR Table 11.8-2

However, a shift from the safety strategy established in the CAR stage was necessary for the following two reasons:

(1) It is difficult to demonstrate the presence of a minimum concentration of hydrazine in the process (slow decrease because of radiation producing nitrous acid from nitric acid, potential process upset conditions leading to more nitrous acid to the extraction column, or loss of stripping solutions).

(2) There is a small margin between the minimum hydrazine concentration (0.1 M for the HAN safety strategy) and the maximum hydrazine concentration (0.14 M for the hydrazoic acid safety strategy).

Therefore, the applicant removed hydrazine from the HAN safety strategy, but it is used as a defense-in-depth strategy (as mentioned above) to prevent a HAN explosion event.

The applicant has subsequently changed the HAN safety strategy to a low acid philosophy. According to the DOE report on HAN explosions (DOE, 1998), HAN-HNO_3 solutions are more stable when the molar ratio of nitric acid to HAN is low (this can be achieved by either low nitric acid concentration or high HAN concentration) and when the temperature is low.

The applicant's general safety strategy is to minimize the threat of HAN explosions by monitoring and controlling the parameters that affect HAN reactivity (i.e., HAN concentration, nitric acid concentration, plutonium concentration, and temperature) within limits that permit applicable vessels within the system to safely vent any gases produced by HAN reactions to prevent overpressurizations (EXP-04a). In cases in which solvent may be present, the applicant's safety strategy includes the requirement to maintain process temperatures resulting from any exothermic HAN reactions below those established for the bounding event (solvent explosion (EXP-06)). When the controlling parameters cannot be assured because of the nature of the process (e.g., temperature or HNO_3 concentration), HAN will be segregated from those parts of the system through the application of transfer protocols based on IROFS sampling. Alternatively, HAN destruction will be assured through the application of IROFS controls at the KPA recycling tank (EXP-04b).

The applicant derived the technical bases for process conditions limits from the results of an updated kinetic model. This model was developed to determine the behavior of HAN-HNO_3 solutions based on existing published kinetic data and rate reactions governing the respective reactions. The applicant developed a separate thermal model specifically to analyze the thermal response of the stripping column (KPA*PULS3000) to various process upsets that could lead to overtemperature conditions. The results of these two models serve as the applicant's basis for establishing limits on the operational parameters that were identified to ensure that overpressurization events or explosions from autocatalytic HAN-nitric acid reactions are highly unlikely within the AP process. Table 8.1.2.4.3.1-3 lists the limits on the process conditions.

[redacted]

The updated kinetic model includes the effects of dissolved iron and the reoxidation of Pu^{+3} in the process. These two effects increase the concentration of HNO_2 in the process, which reduces the decomposition temperature of solutions containing HAN. The model also includes additional reaction mechanisms for the autocatalytic decomposition of HAN. Additionally, the applicant did not include in the model the reactions of hydrazine (N_2H_4) and hydrazoic acid (HN_3) to determine the worst conditions. This is a conservative assumption.

The applicant proposed the following safety strategies to prevent a HAN explosion in the AP process:

(1) controls applicable where HAN is present to control HAN reactions (EXP04a)
(2) controls to prevent the introduction of HAN into certain equipment (EXP04b)

The staff notes that both the Pu^{+4} reduction reaction with HAN and the reaction between HAN and HNO_3 are exothermic. The heat released by these reactions can increase the solution temperature up to the point (LFL safety limit) of becoming a solvent explosion concern before becoming a HAN explosion issue. Therefore, the temperature limit in EXP04a is bounded by the solvent explosion event (EXP06). These reactions also produce off-gases that must be adequately vented to avoid an overpressurization explosion.

Controlling HAN Reactions (EXP04a)

In this case, the applicant's general safety strategy to minimize the threat of HAN explosions is to maintain the parameters that affect HAN reactivity (i.e., HAN concentration, nitric acid concentration, plutonium concentration, and temperature) within limits that permit the applicable vessels within the system to safely vent any ensuing off-gases resulting from HAN reactions to prevent overpressurizations. Where solvent may also be present, the applicant's safety strategy includes a requirement to maintain resulting process temperatures ensuing from any exothermic HAN reactions below those established for the bounding event (i.e., solvent explosions).

HAN reactions are controlled in the following equipment and vessels:

[redacted]

Confining HAN in the Process (EXP04b)

When the controlling parameters cannot be assured out of operational necessity (such as nitric acid concentration or process temperature), HAN is either segregated from those parts of the system through the application of transfer protocols [redacted]

[redacted]

[redacted]

Items Relied on for Safety

[redacted]

[redacted]

- [redacted]
- [redacted]

KPA Plutonium Stripping Column, KPA Uranium Scrubbing Column, Second KPA Diluent Washing Column, and KPA Slab Settler

[redacted]

[redacted]

- [redacted]
- [redacted]
- [redacted]
- [redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

KPA Oxidation Column and Associated Bottom Tank, KPA Air Stripping Column and Associated Bottom Tank, KPA Plutonium Reception Tank, and KCA Batch Constitution Tanks

HAN remaining in the aqueous stream from the reduction of Pu (IV) to Pu (III) in the plutonium stripping column flows downstream to the oxidation column, where it is eliminated through reaction with NO_x. [redacted]

[redacted]

- [redacted]
- [redacted]
- [redacted]
- [redacted]

[redacted]

[redacted]

KPA Recycling Tank

HAN that may have inadvertently reached the KPA uranium vessel (KPA*TK5300) is routed to the recycling tank (KPA*TK9500) for destruction by NO_x [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

KPA Plutonium Barrier Mixer-Settler

HAN is introduced to the plutonium barrier mixer-settler (KPA*MIXS4000) from a HAN reagent unit (RHN) tank. [redacted]

[redacted]

KPA Plutonium Rework Tank

The KPA plutonium rework tank (KPA*TK8500) receives HAN-bearing solutions from draining vessels that contain HAN. Because of the number of vessels that drain into this tank, numerous combinations of chemical makeup could affect HAN safety. [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Experience at La Hague Facility

Discussions with the applicant during in-office reviews indicated that the French reprocessing plant at La Hague (the facility that serves as the basis for the MFFF AP flowsheet) has used HAN for reductive stripping of plutonium for many years without incident. In actuality, the facility uses both HAN and uranous (U (IV)) nitrate as reducing agents, depending upon the nature of the feed solutions and the separation desired.

Use of U (IV) nitrate is desirable when uranium concentrations are high relative to the concentration of plutonium and high purity plutonium separation is not needed. For example, in spent fuel reprocessing at La Hague, the plutonium concentration [redacted] is only about [redacted] of the uranium concentration [redacted] in the feed to the first extraction cycle. In this cycle, U (IV) nitrate is selected as the reductant because the plutonium will not be recovered at high purity until the second and third extraction cycles (known as the plutonium purification cycles). In those cycles, when most of the uranium has already been separated out and it is not desirable to add more uranium to the process, HAN is selected as the reductant.

Reduction of Pu (IV) with U (IV) nitrate is fast at ambient temperatures. Reaction is possible in both the aqueous and organic phases, but the reduction is slower in the organic phase. Also, parasitic reoxidation of the U (IV) is possible in the organic phase. For proper partitioning, the ratio of U (IV) to plutonium must be greater than 3. Thus, excess reductant is usually required. This excess uranium must later be recovered. Therefore, the use of U (IV) as a reductant is undesirable if a high purity plutonium separation is the objective.

On the other hand, reduction of Pu (IV) to Pu (III) by the use of HAN is desirable when a high purity separation of plutonium is the objective. The reduction does occur at ambient temperature, but the kinetics of the reaction are relatively slow. As a result, the temperature is usually increased. HAN is not extractable into the organic phase, and reduction by HAN will not occur in high acid concentrations.

Excess HAN is also needed to achieve the plutonium reduction, but unlike U (IV) nitrate, the HAN does not have to subsequently be recovered, as it can easily be destroyed by NO_x later in the process. HAN is also more stable, with respect to reoxidation, than U (IV).
Table 8.1.2.4.3.1-4 summarizes the relative advantages and disadvantages to using U (IV) nitrate and HAN.

Table 8.1.2.4.3.1-4 Relative Advantages and Disadvantages in the Use of U (IV) Nitrate and HAN in the Reduction of Pu (IV) to Pu (III)

	Advantage	**Disadvantage**
U (IV) Nitrate	Efficiency (fast reaction at ambient temperature)	Increased uranium concentration. Uranium must subsequently be recovered, if high purity plutonium product is desired.
HAN	Excess reductant is easily destroyed and thus does not need to be recovered. HAN is also more stable with respect to parasitic reoxidation of the reductant.	Low acid concentration must be maintained.

The staff notes that, in both cases at the La Hague facility, hydrazine (in the form of hydrazine nitrate, $N_2H_4 \bullet HNO_3$) is employed as a scavenger to prevent reoxidation of Pu (III) by nitrous acid (HNO_2).

Staff Evaluation and Findings

Based on the above discussion, the staff finds that HAN (with the associated use hydrazine) is an appropriate choice of reductant for use in the KPA stripping column (KPA*PULS3000) of the AP process in the MFFF. The objective of the stripping column is to separate plutonium from uranium into a relatively pure Pu (III) nitrate stream. The Pu (III) is subsequently reoxidized to Pu (IV) in the NO_x column (KPA*CLMN6000), where the residual (excess) HAN and hydrazine are also destroyed.

The NRC staff found that the applicant’s safety strategy for the use of HAN is consistent with DOE guidance document entitled “Technical Report on Hydroxylamine Nitrate” (DOE, 1998), practices employed at DOE facilities, and the practices employed at the La Hague facility in France.

The staff evaluated the MOX Services safety strategy for the use of HAN at the MFFF and finds that the applicant has described the facility, equipment, and processes in sufficient detail to meet the requirements of 10 CFR 70.22 and 10 CFR 70.65, consistent with the acceptance criteria of NUREG-1718, Section 8.4.3.

The credited IROFS discussed above provide a method to control the parameters that have an effect on HAN/nitric acid autocatalytic reactions. The applicant has also committed to have fail-safe and redundant active engineered IROFS (e.g., temperature controllers and isolation valves) and use them in combination with passive engineered IROFS (e.g., slab settler, flow restricting orifices, and process vessels and pipes) and administrative IROFS (i.e., process and reagent sampling).

The NRC staff finds that this is an acceptable approach for complying with the single-failure criterion. The single-failure criterion, in combination with experience from DOE facilities, the La Hague facility, management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS give the NRC staff reasonable assurance that this high consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61.

The applicant uses features to reduce the challenge to IROFS, where practical. [redacted] he NRC staff finds that these features comply with the defense-in-depth requirements of 10 CFR 70.64(b).

The staff also finds reasonable assurance that the applicant has identified the hazards and accident sequences associated with the use of HAN and has credited IROFS sufficient to meet the performance requirements of 10 CFR 70.61, consistent with the acceptance criteria of NUREG-1718, Sections 7.4 and 8.4.

8.1.2.4.3.2 Hydrazine and Hydrazoic Acid Explosions (EXP10)

Event Description

Hydrazoic acid (HN_3) is formed in the AP process when hydrazine (N_2H_4) is oxidized by nitrous acid (HNO_2) according to the following equation:

$$N_2H_5^+ + HNO_2 \rightarrow HN_3 + 2\,H_2O + H^+ \qquad (8.1.2.4.3.2\text{-}1)$$

Nitrous acid is always present in nitric acid (HNO_3) solutions. Hydrazine is used in the AP process to protect HAN from reacting with nitrous acid. The use of hydrazine reduces the potential for a runaway exothermic reaction of HAN with plutonium solutions in nitric acid.

When hydrazoic acid is formed it partitions mostly to the organic phase with a smaller amount going to the aqueous phase. The hydrazoic acid in the aqueous phase reaches a chemical equilibrium typical of a weak acid. Undissociated hydrazoic acid vapor can evolve from either the organic or aqueous phase according to Henry's law.

The hydrazoic acid in the organic phase is treated at the solvent recovery (KPB) unit in the solvent washing mixer-settler (KPB*MIXS1000) with sodium carbonate (Na_2CO_3) and sodium hydroxide (NaOH). The treatment removes the hydrazoic acid from the organic phase to the aqueous (alkaline) phase as sodium azide (NaN_3), according to equations (8.1.2.4.3.2-2) and (8.1.2.4.3.2-3). The sodium azide is eventually transferred to the high alpha waste (KWD) unit for final destruction of the azide ion (N_3^-). The destruction of azide ions is accomplished by adding an excess of sodium nitrite ($NaNO_2$) at the KWD alkaline waste tank (KWD*TK4010) followed by acidification of the solution at the KWD neutralization tank (KWD*TK4015). Before the solution in KWD*TK4010 is transferred to KWD*TK4015, a process sample is taken to ensure that there is an excess of the nitrite ion (NO^-) relative to the azide ion. [redacted] Equations (8.1.2.4.3.2-4) and (8.1.2.4.3.2-5) show the formation of nitrous acid and the reformation of hydrazoic acid, respectively. Nitrous acid and hydrazoic acid react according to equation (8.1.2.4.3.2-6) to finally destroy the azide ion which liberates nitrous oxide (N_2O) and nitrogen (N_2) gases.

$$2HN_3 + Na_2CO_3 \rightarrow 2NaN_3 + H_2CO_3 \qquad (8.1.2.4.3.2\text{-}2)$$

$$HN_3 + NaOH \rightarrow NaN_3 + H_2O \qquad (8.1.2.4.3.2\text{-}3)$$

$$NaNO_2 + HNO_3 \rightarrow HNO_2 + NaNO_3 \qquad (8.1.2.4.3.2\text{-}4)$$

$$NaN_3 + HNO_3 \rightarrow HN_3 + NaNO_3 \qquad (8.1.2.4.3.2\text{-}5)$$

$$HNO_2 + HN_3 \rightarrow N_2O + N_2 + H_2O \qquad (8.1.2.4.3.2\text{-}6)$$

[redacted]

[redacted]

Safety Strategy

The risk of a hydrazoic acid explosion comes from both the liquid and vapor phases. The hydrazoic acid in solution can be explosive if it is excessively concentrated or heated. In addition metal azides can be formed if the process solutions containing hydrazoic acid are mixed with solutions containing dissolved metals (e.g., aqueous raffinates solutions). Metal azides are shock-sensitive explosives if they precipitate and dry out. The applicant treated the metal azide explosion as a separate event; see Section 8.1.2.4.3.3 of this SER for the description and evaluation of the metal azide explosion event. [redacted]

The applicant's safety strategy to prevent an explosion event related to HN_3 comprises the following six elements:

(1) Limit the amount of hydrazoic acid that can form.

(2) Confine the hydrazoic acid in the process by sampling.

(3) Confine the hydrazoic acid in the process by density controls.

(4) Limit the process temperature in order not to exceed the explosive partial pressure of hydrazoic acid, which corresponds to a temperature of approximately 55 degrees C.

(5) Ensure hydrazoic acid destruction with NO_x gases at the KPA recycling tank (KPA*TK9500) and adequate ventilation of this vessel.

(6) Ensure that azides in alkaline solution do not come into contact with concentrated nitric acid.

Items Relied on for Safety

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Staff Evaluation and Findings

The NRC staff reviewed previous operating experience used to prevent a hydrazoic acid explosion in nuclear facilities at DOE (WSRC, 2001). The NRC staff found that the applicant's safety strategy is consistent with the practices used at DOE facilities. DOE, as well as the applicant, controls the maximum hydrazine concentration around the same value (0.14 M) to limit the amount of hydrazoic acid produced. This ensures that the hydrazoic acid explosive concentrations in the liquid phase (approximately 8.4 M) and vapor phase (partial pressure approximately 65 torr) are not exceeded.

For the IROFS that are AEC, the applicant proposed to use a redundant set of the particular IROFS. The AEC IROFS are the active portions of the reagent and process sampling controls, density controls, process temperature controls, process flow control which ensures the minimum NO_x flow to KPA*TK9500, the active portion of the process vessel off-gas venting, and process-level controls. In some cases, the applicant also uses a diverse set of IROFS. [redacted] The process flow control is a combination of AEC and PEC. The AEC portion of the process flow control ensures that the minimum NO_x flow exists. If the flow decreases below the minimum, it automatically isolates the flow of solutions potentially containing hydrazoic acid to KPA*TK9500. The PEC portion of the process flow control uses restricting orifices to ensure the maximum flow of solutions potentially containing hydrazoic acid that can be safely treated with the NO_x flow in KPA*TK9500. In addition, before initiating the NO_x flow an administrative

[redacted]

[redacted]

The NRC staff finds that the redundancy and fail-safe condition of the AEC IROFS, the diversity of IROFS, the dual sampling and additional activities of the EAC IROFS, and the use of PEC IROFS are adequate to comply with the single-failure criterion. In addition, the use of management measures (as described in Chapter 15 of the LA) and quality assurance requirements (as described in the MPQAP), the use of codes and standards for AEC and PEC IROFS, and the consistency of the safety strategy with previous operating experience at DOE facilities provide the NRC staff reasonable assurance that this high consequence accident scenario is highly unlikely. Therefore, the proposed safety strategy and the credited IROFS comply with the performance requirements of 10 CFR 70.61.

Additionally, the applicant proposed to use features to reduce the challenge to IROFS, where practical. For this particular event, these features include [redacted] The use of these features satisfy the defense-in-depth requirements in 10 CFR 70.64(b).

[redacted]

8.1.2.4.3.3 Hydrazine and Metal Azides Explosions (EXP11)

Event Description

Metal azides are formed in the AP process when hydrazoic acid (HN_3) comes into contact with certain metals in nitric acid solution. If metal azides precipitate and dry in a tank or drip tray, they may become unstable and explosive. Metals exist in the system as impurities or as part of the process. The general reaction for the formation of a metal azide is given by equation (8.1.2.4.3.3-1):

$$HN_3 + [\text{metal}^+] \rightarrow [\text{metal}]N_3 + H^+ \qquad (8.1.2.4.3.3\text{-}1)$$

The majority of the metal impurities come from the feed. Most of them do not affect azide precipitation because they either do not form insoluble metal azide precipitates (i.e., their ability to complex the azide ion is not significant) or their concentrations are low.

The metals that can complex the azide ion and exist as part of the process are sodium (Na) and silver (Ag). Sodium is introduced at KPB*MIXS1000, as sodium carbonate and sodium hydroxide, to remove the hydrazoic acid from the solvent. Equations (8.1.2.4.3.2-2) and (8.1.2.4.3.2-3) of the previous section of this SER show the removal of hydrazoic acid from the solvent into the alkaline (aqueous) solution as sodium azide. Sodium azide solubility increases as pH increases. Therefore, sodium azide will remain in solution in an alkaline medium until sent to the KWD unit for final destruction. Silver is added to the dissolution (KDB), and dechlorination and dissolution (KDD) units as silver nitrate ($AgNO_3$), and it is used to generate silver (II) (Ag^{+2}) ions by electrolysis. Silver (II) ions catalyze the dissolution of plutonium oxide (PuO) in nitric acid solution. [redacted]

Safety Strategy

The applicant postulated metal azide explosions in the AP process because of the presence of hydrazoic acid and metals. The applicant's safety strategy for the prevention of an explosion related to a metal azide is to limit the formation and potential precipitation of metal azides or promote downstream destruction of hydrazoic acid. Azide explosions could cause a release of radioactive material

Items Relied on for Safety

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Staff Evaluation and Findings

The applicant proposed to use a diverse set of IROFS to prevent explosions related to the formation and precipitation of metal azides. Examples of this diversity include [redacted] The strategy for the leaked solution is described above, Section 8.1.2.4.3.2 of this SER describes the strategy for the destruction of hydrazoic acid. In addition, all of the AEC IROFS are designed to be redundant and to fail safe, and the AP process vessels and pipes (a PEC) are qualified to withstand an earthquake. [redacted]

[redacted]

The NRC staff finds that this is an acceptable approach for complying with the single-failure criterion. The single-failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (described in the MPQAP), and the use of codes and standards for engineered IROFS give the NRC staff reasonable assurance that these high consequence scenarios are highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61.

The applicant also uses features to reduce the challenge to IROFS, where practical. [redacted] The NRC staff finds that these features comply with the defense-in-depth requirements of 10 CFR 70.64(b).

[redacted]

Furthermore, in the highly unlikely event that a metal azide explosion occurs, the applicant has the following features available: [redacted]

8.1.2.4.4 Hydrogen Peroxide Explosions (EXP05)

Hydrogen peroxide (H_2O_2) is used in the AP process for plutonium and silver reduction in the dissolution (KDB) and dechlorination/dissolution (KDD) units, introducing the possibility of H_2O_2 explosion events. The applicant determined that the following H_2O_2 explosion events are possible in the KDB and KDD units because H_2O_2 may be present with substrates that can be oxidized or reduced (including electrochemical reactions in the electrolyzer). Explosions related to H_2O_2 could cause a release of radioactive material:

[redacted]

[redacted]

[redacted]

- [redacted]

[redacted]

- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]

[redacted]

[redacted]

- [redacted]

[redacted]

The NRC staff finds that these features comply with the defense-in-depth requirements of 10 CFR 70.64(b). The staff also finds that the applicant has described the facility, equipment, and processes in sufficient detail to meet the requirements of 10 CFR 70.22 and 10 CFR 70.65, consistent with the acceptance criteria of NUREG-1718, Section 8.4.3. The staff evaluation concludes that the applicant's proposal to use multiple independent administrative and engineered controls to prevent the above-mentioned events is acceptable.

The staff finds reasonable assurance that the applicant has identified the hazards and accident sequences associated with hydrogen peroxide explosions and credited IROFS sufficient to meet the performance requirements of 10 CFR 70.61, consistent with the acceptance criteria of NUREG-1718, Section 8.4.

8.1.2.4.5 TBP-Nitrate (Red Oil) Explosions (EXP07)

Event Description

The acid catalyzed hydrolysis of TBP and subsequent oxidations of byproducts introduce the risk of thermal "runaway reactions," leading to an overpressurization event that could cause a release of radioactive material with consequences for the facility worker, site worker, and IOC, as well as environmental release.

The risk of red oil events affects the following processing units in the MFFF (red oil events in any of these units in the MFFF could cause a release of radioactive material):

- KPA—purification cycle
- KPB—solvent recovery
- KPC—nitric acid recovery
- KCA—oxalic precipitation, filtration, and oxidation
- KCD—oxalic mother liquor recovery
- KWD—stripped uranium
- KWS—waste solvent
- LGF—laboratory liquid waste receipt

Energetic red oil reactions can involve TBP, HNO_3, Pu–nitrate–TBP adducts, and TBP degradation products from chemical reactions and radiolysis. The organic phase in red oil reactions can have varying compositions of the aforementioned species; but, at a minimum, TBP and HNO_3, must be present before a red oil reaction can occur.

TBP degradation reactions may proceed by the following three mechanisms:

(1) Acid Catalyzed Hydrolysis: TBP hydrolyzes to form dibutyl phosphoric acid and monobutyl phosphoric acid, and ultimately phosphoric acid. Butanol, which is volatile and flammable, is produced as a byproduct in these reactions.

(2) Dealkylation: TBP undergoes dealkylation with nitric acid to butyl nitrate and dibutyl phosphoric acid.

(3) Pyrolysis: At high temperatures, TBP can undergo pyrolysis to butene, which is volatile and flammable. Butene generation begins with water at approximately 160 degrees C

and continues up to approximately 260 degrees C, where the last few percent decompose rapidly to butane. This rapid decomposition is not well understood.

A number of byproduct reactions can also occur. The degree of byproduct oxidation is a function of temperature, acidity, and organic phase metal ion concentration. Studies have identified almost 100 degradation products (out of more than 150 possible), including 17 nitrated solvents.

8.1.2.4.5.1 Applicant's Proposed Approach for Red Oil

Safety Strategy

The applicant's safety strategy for TBP-nitrate (red oil) explosions involves the application of IROFS to meet the performance criteria of 10 CFR 70.61. Red oil explosions are postulated in the AP because of the possible commingling of TBP and nitric acid. The applicant will employ one of the following three strategies, depending upon the location and operation associated with the equipment and vessels concerned:

- heat transfer strategy
- evaporative cooling strategy
- TBP prevention strategy

[redacted]

Items Relied on for Safety

Tables 8.1.2.4.5-1 and 8.1.2.4.5-2 (BNL, 2009b) summarize the applicant's engineered and administrative IROFS controls for this event.

[redacted]

[redacted]

[redacted]

The applicant maintains that the current strategy for preventing red oil explosions in the MFFF is consistent with the Defense Nuclear Facilities Safety Board (DNFSB) recommendations stated in "Control of Red Oil Explosions in Defense Nuclear Facilities" (DNFSB/TECH-33) (DNFSB, 2003). The DNFSB document lists the following simultaneous conditions as being necessary for a red oil runaway reaction to occur:

- presence of TBP in the organic phase
- organic phase in contact with nitric acid greater than 10 M (with diluent present) or 14.5 M (with no diluent present)
- solution temperature greater than 130 degrees C
- insufficient venting area

It should also be noted that, even with sufficient venting, detonation of gases can occur if other control conditions are exceeded. In addition, the presence of diluent or metal ions can exacerbate runaway reactions.

The DNFSB report recommends the following controls, to be used in combination, to prevent a red oil runaway reaction:

- Temperature Control—Maintain temperatures at less than 130 degrees C.
- Pressure Control—Provide sufficient venting to limit pressure excursions.
- Mass Control—Remove organics from the process (i.e., prevention).
- Concentration—Maintain nitric acid concentration to at or below 10 M.

The applicant's red oil strategy implements the recommended controls as follows:

(1) Temperature Control—Maintain temperatures at less than 130 degrees C.

- The only AP equipment that operates near or above this temperature is [redacted]

- A prevention strategy will be employed in [redacted] to augment the temperature controls. In [redacted] temperature control will be maintained by evaporative cooling.

- Additional temperature controls consist of [redacted] as well as normal process temperature controllers.

- Other equipment capable of heating solutions containing organic phase [redacted] and normal process temperature controllers.

(2) Pressure Control—Provide sufficient venting to limit pressure excursions.

- For the AP process, all vessels are vented. Vents are sized to satisfy the Fauske relationship for credible quantities of organic material. Additional IROFS limit organic materials to credible quantities.

- There are no valves in individual vent lines.

[redacted]

(3) Mass Control—Remove organics from the process (i.e., prevention).

[redacted]

[redacted]

[redacted]

[redacted]

(4) Concentration—Maintain nitric acid concentration at or below 10 M.

[redacted]

[redacted]

[redacted]

[redacted]

Additional controls include agitation of vessels and supply of “scavenging air.” Also, controls are placed on chemical properties of the diluent (e.g., avoid cyclic hydrocarbons in diluent).

[redacted]

The applicant calculated vent sizes by taking into account all incoming gaseous flows, all outgoing gaseous and vapor flows, and gas-producing chemical reactions that could occur within the vessel. For vessels employing the red oil heat transfer strategy, where controlled conditions significantly limit the heat generated from red-oil-related reactions, this entails venting off-gases from TBP-nitric acid and related reactions to prevent overpressurization, as well as venting of normal flows from air lifts and other sources. Heat removal via venting is not credited for red oil safety of vessels that fall under the heat transfer strategy. The same reaction rates that were used for quantifying the heat generated from TBP-nitric acid reactions were also applied to develop off-gas rates. These rates are based on the applicant’s experimental data. Vents that are intended to provide overpressure protection against red oil events for vessels employing the evaporative cooling strategy [redacted] were sized in accordance with a correlation between TBP mass and vent size developed by Fauske (Fauske, 1994).

In the heat transfer case, the applicant calculated the required vent size by applying a large safety factor (greater than a factor of 10) to the off-gas generation rate and including all credible, concurrent flows postulated for each vessel. This value was then rounded to the next highest standard pipe size and compared to the actual vent size for each vessel. In all cases, the existing vent size was at least as large as the required vent size; in most cases, it was at least one standard pipe size larger. In the evaporative cooling case, a required vent size was calculated (and rounded to the next highest standard pipe size) based on all credible, concurrent flows postulated for each vessel minus the requirement for red oil venting. This value was then compared to the actual vent size for each vessel, with the excess available venting area equated to a mass of TBP that could be safely vented according to the aforementioned correlation. In all cases, the existing vent size was adequate for more than the maximum credible amount of TBP postulated in each evaporative cooling vessel. In all but one case, the vent size was adequate for more than twice the maximum credible amount of TBP.

The applicant indicated that vent piping is designed to maintain a 5-percent slope and can be washed with decontamination fluid. Vent piping will be frequently used, and the applicant committed to continuously monitor for flow-related parameters associated with the KWG unit to ensure that it is reliable and unlikely to clog.

The applicant indicated that the process vessel venting control, including the KWG unit, is designed to be available 100 percent of the time. To have the required availability, the KWG unit design includes three fans. Each fan is designed to exhaust the maximum flow rate. Two filters in parallel are implemented. In addition, a bypass completes the KWG exhaust system.

The applicant stated that the design and technology associated with process vessel venting are known to be reliable because they are based on existing proven technologies used for numerous years in the La Hague and MELOX facilities. The La Hague and MELOX facilities implement the systems with modifications regularly deduced from operating experience. The present MFFF design incorporates these modifications.

The MFFF uses design features and management measures that protect against plugging of vent piping and equipment systems. Demisters are designed to be cleaned online and are washable. Process equipment is designed to be washed with decontamination fluid at frequent intervals or during annual outages, as determined by experience with the La Hague and MELOX facilities.

The applicant expects radiolytic degradation within the AP process to be limited because the thermal power of the plutonium to be processed is expected to be relatively low [redacted] and the residence time of organic materials in contact with the plutonium will also be limited.

The applicant described the following three potential mechanisms for transfer of TBP to the KCD and KPC units (MOX, 2005):

[redacted]

[redacted]

[redacted]

[redacted]

[redacted] The applicant has committed to test and optimize the pulsed columns after construction of the MFFF. In these tests, the retention rate, which accounts for diluent washing efficiency, will be measured as a function of pulse pressure and frequency. These parameters will subsequently be optimized.

8.1.2.4.5.2 NRC Review of Red Oil

The staff has reviewed the proposed use of evaporators at the MFFF. Contact between TBP with aqueous solutions of nitric acid and acidic solutions of metal nitrates is intrinsic to the PUREX and similar solvent extraction processes. Evaporator treatment of PUREX streams has been performed safely since the 1950s. However, several events have occurred involving a range of conditions.

Significant red oil explosions have occurred at the Hanford Site (United States) in 1953; the SRS (United States) in 1953 and 1975; Oak Ridge (United States) in 1959; the Uranium Trioxide Plant (Ontario, Canada) in 1980; and the former Soviet Union (Tomsk-7) in 1993 (Thompson, 2000; IAEA, 1998). Other, less serious incidents have also occurred, such as an explosion at the Nuclear Fuel Complex in Hyderabad, India, on November 7, 2002. An evaporator used to concentrate a purified uranyl nitrate stream before precipitation experienced a red-oil-type explosion that blew the top off of the evaporator vessel.

The staff notes from its review that these events occurred because of the unanticipated presence (either carryover or accumulation) of the TBP, solvent, and degradation products.

8.1.2.4.5.3 NRC Review of Controls, Limits, and Safety Design Bases Proposed in the Literature

Experiences at DOE and Related Facilities and DNFSB Recommendations

In 2003, the DNFSB issued a report (DNFSB/TECH-33), which assessed the potential for a red oil explosion in the DOE defense nuclear facilities complex (the complex) for the year 2003. In that report, red oil is defined as a substance of varying composition formed when an organic solution, typically TBP and its diluent, comes in contact with concentrated nitric acid at a temperature above 120 degrees C. Red oil is relatively stable below 130 degrees C, but it can decompose explosively when its temperature is raised above 130 degrees C.

The report describes three red oil events. Two have occurred in the United States—at the Hanford Site in 1953 and at SRS in 1953 and 1975. The report also describes a third red oil explosion, which occurred in 1993 at the Tomsk-7 site at Seversk, Russia.

The report states that generic types of equipment capable of producing red oil in the complex are categorized as evaporators, acid concentrators, and denitrators. The chemicals necessary to produce red oil are, at a minimum, TBP and nitric acid; other, contributory chemicals can include diluent (kerosene-like liquid used to dilute TBP) or aqueous phase metal nitrates or both.

The report generally categorizes controls for the prevention or mitigation of a red oil explosion as controls for temperature, pressure, mass, and concentration. The report states that maintaining a temperature of less than 130 degrees C is generally accepted as a means to prevent red oil explosions.

The report also stated that sufficient venting serves to keep pressure from destroying the process vessel, while also providing the means for evaporative cooling to keep red oil from reaching the runaway temperature.

The report described mass controls in the context of utilization of decanters or hydrocyclones to remove organics from feedstreams entering process equipment capable of producing red oil. In addition, limiting the total available TBP is considered another mass control that mitigates the consequence of a red oil explosion by limiting its maximum available explosive energy.

Finally, the report stated that concentration control can be utilized to keep the nitric acid below 10 M.

A significant conclusion of this study is that none of the controls should be used alone; rather, they should be used together to provide effective defense in depth for prevention of a red oil explosion.

Three facilities in the DOE complex (operating at the time the DNFSB report was issued) were identified as capable of producing a red oil explosion: H-Canyon at SRS, F-Canyon at SRS (F-Canyon is currently shut down), and Building 9212 at the Y-12 National Security Complex. These facilities contained the necessary process equipment and chemicals to form red oil and bring it to the runaway temperature. The MFFF was also identified as having the capability to produce red oil, but the report did not discuss this facility further, as it is to be regulated by the NRC.

<u>Major Red Oil Explosion Events</u>

Several, well-documented, explosive incidents have occurred in the United States (SRS in 1953 and 1975, Hanford Site in 1953). The Hanford and the first Savannah River (1953) event involved evaporators undergoing heating with low-pressure steam; temperature control was not precise but ranged from 135–140 degrees C. The second Savannah River incident (1975) involved a denitrator operating at a temperature higher than 150 degrees C. The pretreatment step for the denitrator is an evaporator operated with controls and a design-basis temperature that does not exceed 135 degrees C. Thus, the TBP and solvent mixture safely traveled through the evaporator before the event in the denitrator.

The incident at Tomsk did not involve heated equipment. The reactions started in a tank at a nominal temperature of 50 degrees C, with an organic layer estimated at 90 degrees C (IAEA, 1998). Overpressure occurred in a tank containing uranium nitrate solution and caused gases to burst through the top of the tank, displacing the cover of the containment cell and leading to a forceful explosion. Release of radioactive materials to the local environment took place through the large holes in the side walls and roof of the room and through the side wall of the galley. There was also a release via a ventilation system through a 150-meter-high stack. The initial release of radioactive materials caused contamination near the building over an area of 1500 m^2. The total beta/gamma activity of material released was said to be 1.5 terabecquerels (IAEA, 1998). The cause of the accident was most likely a lack of the compressed air needed for thorough mixing of the solutions (IAEA, 1998). Compressed air was being used to mix the solutions; however, sensor measurements indicated that it was likely that none of this compressed air was introduced into the vessel.

It is not clear whether the absence of compressed air was the result of operator error or plant failure, but investigators considered the former to be the more likely. However, it was apparent that, at the time the nitric acid was being introduced to adjust the acidity, there was not enough air to provide the necessary mixing of the solutions. Under these circumstances, the solutions could have settled out into different layers, allowing the oxidation and nitration of the organic layer by the nitric acid. It is likely that the reactions occurred with the more concentrated nitric acid solution in the upper layer. This assumption is supported to some degree by the lack of noticeable pressure rise until approximately 1.5 hours after the nitric acid solution had been introduced into the vessel. As the oxidation of organic substances by nitric acid is autocatalytic, the rate of gas release would have increased and, because the reaction is also exothermic, would have been accompanied by a rise in temperature. Eventually, a point was reached when the amount of gas generated was more than could be vented through the stack. Attempts were made to depressurize the vessel. These attempts were unsuccessful, and the pressure continued to rise until it reached about 5 atmosphere (atm). Within a few minutes, the pressure rose very rapidly to about 18 atm and the vessel ruptured. The resulting shock wave was sufficient to raise and displace the concrete cell covers, as well as cause damage to the equipment room above. Some of the organic material was released in the form of steam and small droplets, and some was probably oxidized to form gaseous products. A localized explosion of the resulting flammable cloud then occurred either as a result of a spark or because of the prevailing high temperature of about 450 degrees C (IAEA, 1998).

An estimation of the amount of organic solution in the vessel and whether enough of it was available to produce a sufficient amount of gas to rupture the vessel proved difficult. Experimental data on two-phase systems utilizing the irradiated extractant, uranyl nitrate, in organic phase and nitric acid (concentration 10–12 M) in aqueous phase indicate that gas would have been released at the temperature prevailing in the vessel. However, without knowledge of the quantity of organic material present, no reliable predictions can be made of the resultant pressure rise. Calculations have shown that to generate the pressure of 18.0–20.0 atm needed to rupture the vessel, the oxidation of 35–40 liters of the organic phase by nitric acid would have been required had the vessel been closed, and 3–5 times this amount would have been required had the gas been vented (IAEA, 1998).

The staff notes from its review that these events occurred because of the unanticipated presence (either carryover or accumulation) of the TBP, solvent, or degradation products or some combination of these materials. The applicant indicated that solvent carryover can be considered as an anticipated event in the facility.

According to the DNFSB report (2003), the following conditions are necessary for a runaway red oil reaction to occur:

- the presence of TBP in organic phase
- organic phase in contact with nitric acid greater than 10 M
- solution temperature greater than 130 degrees C
- insufficient venting area

All of these conditions must exist simultaneously in order for a runaway red oil reaction to occur. Even with sufficient venting, detonation of gases can occur if other above conditions are exceeded.

The DNFSB report (2003) also states that the following controls can be used to prevent a red oil event:

- Temperature: Maintain at less than 130 degrees C.
- Pressure: Provide a sufficient vent for the process to limit pressure excursions.
- Mass: Remove organics from the process before reaching an evaporator, acid concentrator, or denitrator.
- Concentration: Maintain nitric acid less than 10 N.

<u>Independent Review of Risk of a Red Oil Excursion by Brookhaven National Laboratory</u>

The NRC staff tasked Brookhaven National Laboratory (BNL) with undertaking an independent analysis of issues related to the risk of a red oil event in the design proposed for the MFFF (BNL, 2009). This task was part of a larger program of providing technical assistance to NRC staff on risk-informed decisionmaking for fuel cycle facilities regulated by the NRC. The BNL study contained insights useful in staff reviews of the MFFF LA (MOX, 2009a). However, the BNL study was meant only to offer an independent perspective on risk.

The BNL study (2009) developed a probabilistic risk assessment (PRA) model that evaluated the failure of some of the red oil event safety strategies resulting from internally initiated process deviations. In particular, the PRA model focused on (1) the failure of evaporative cooling in selected process vessels and (2) the failure of the TBP prevention strategy through events, such as emulsification and the formation of a third phase, or a rag layer, which would eventually result in a violation of the success criteria for evaporative cooling. BNL considered the PRA to be a limited-scope risk assessment for several reasons:

- The analysis excluded generic risks associated with external hazards, such as seismic events, internal fires, or loss of offsite power events, including station blackout. These initiating events potentially could lead to other high consequence outcomes, similar to red oil explosions. Including these other initiating events would have greatly enlarged the scope of the study, which was limited to red oil explosions. [redacted]

[redacted]

- In analyzing the red oil reaction, the applicant's characterization of the phenomenon was accepted broadly by focusing only on the thermal decomposition reactions. As noted above, the impact of radiolytic dissociation on this reaction was not considered because it was felt that radiolysis would have a minor impact in the MFFF because the concentrations and decay rates of the radionuclides involved are low.

- The analysis did not consider failures of the heat transfer strategy. This strategy applies to the adequacy of passive heat transfer to the room environment from process vessels containing solutions at lower temperatures (about 55 degrees C and below); its success depends on the proper operation of room cooling (i.e., the facility's HVAC system). However, including failures of the HVAC system would have greatly enlarged the scope of the analysis, which was limited to red oil explosions.

- The semi-empirical model for the TBP-nitrate reactions developed by the applicant to set the success criteria for evaporative cooling safety was accepted as the basis to further evaluate the phenomenon. The applicant considered this model to be conservative because it is based on the heat generated in a pure TBP-nitric acid reaction, rather than on the 30% TBP-70% HPT mixture that the MFFF will use.

The BNL study (2009) initially made a qualitative assessment of the factors that could contribute to a possible red oil explosion in the various process units comprising the AP process. There are eight process units in the AP process wherein organics and nitric acid could or may come into contact during normal operation. These units include the following: [redacted] Red oil explosions could occur in these eight units. The BNL study focused on units (1) through (5); the process conditions in these units place them at a higher risk of a red oil explosion as compared to units (6) through (8). The evaluation considered the likelihood of a red oil explosion for each of these five process units in terms of the equipment employed, the sequence of operations, and the conditions (e.g., temperature, pressure) during operations.

BNL selected four vessels in two process units for more detailed evaluation based on the heat sources present, the heat balance, and the potential for TBP transfer, which could potentially violate any of the red oil explosion safety strategies outlined earlier. These vessels included: [redacted]

Quantitative evaluation was accomplished by delineating accident sequences, presented in the form of event trees and fault trees, to gain further insights into possible combinations of failures that could lead to a red oil explosion in the process vessels selected after the qualitative assessment. Quantification, using the SAPHIRE code, gave the point frequency of a red oil explosion and a 5th-percentile and 95th-percentile frequency to show the range of uncertainty.

The red oil explosion scenario in the first-stage evaporator was modeled under two conditions of TBP accumulation: (1) normal accumulation of TBP (i.e., the accumulation of a small amount by mechanical entrainment with the aqueous phase) and (2) the upset accumulation of TBP, resulting from a severe process malfunction, such as the formation of an emulsion that transfers large quantities of solvent.

Under the first condition, a high solution temperature and failure of the evaporative cooling strategy is necessary for a red oil explosion to occur. The initiating event for this scenario is the increase in solution temperature that can lead to a red oil explosion should the evaporative cooling strategy fail. This initiating event may result from a loss of temperature control or a rupture of the heat exchanger tube. The following events in the event tree model the different ways in which the various success criteria for evaporative cooling (i.e., maintaining the required aqueous to TBP mass ratio and the TBP layer thickness) are violated. The first is the operator's failure to flush the vessel at the end of 6 months, a period assumed conservatively to cause the unavailability of evaporative cooling for 6 months until the next scheduled flushing. The second can result from a number of failures of equipment needed to maintain control of the TBP level. The last event in the tree represents the success or failure of venting to ensure the maintenance of the solution's temperature below a safe level. Venting is provided by a two-train system consisting of fans and HEPA filters with an additional standby fan. There are two red oil event (ROE) sequences for this scenario; in the first, the level control is successful but venting fails, while in the second, sufficient TBP accumulates to violate the criteria for evaporative cooling. The dominant contributor in the first sequence is common-cause failure of plugging of the two sets of HEPA filters. In the second sequence, the dominant contributor is human error, that is, the failure of the operator to carry out the vessel's 6-monthly flush.

Under the second condition, multiple failures of the barriers that prevent excessive TBP transfer must occur before the violation of the criteria for evaporative cooling. The transfer is assumed to begin with a severe process malfunction, such as the formation of an emulsion in the initial pulse extraction column in the KPA unit. Following this, the diluent washing pulse columns that remove the TBP also could fail to break up the organics entrained in the aqueous phase or in inducing a manual termination of TBP transfer. Very limited data formed the basis of assigning failure probabilities for these events. Further barriers to the transfer of organics are afforded by sampling controls that detect TBP and density controls that detect HPT. Failure of these controls was modeled using standard fault tree modeling. The initiating event for this scenario again is a loss of temperature control or rupture of a heat exchanger tube engendering a rise in solution temperature. The top events in the event trees relate to the success or failure of the various pulse columns in breaking up entrained organic material, followed by the success or failure of the sampling and density controls. Venting is not modeled, as the amount of TBP assumed to be transferred in the upset accumulation condition would violate the criteria for the success of evaporative cooling. Dominant contributors to the red oil explosion in this case include the ineffectiveness of density controls, common-cause failure of the density transmitter, failure of sampling analysis, failures in the diluent wash column, and malfunctions of the pulse extraction column.

The BNL PRA model for a red oil explosion in the concentrates collection tank assumed the following:

(1) Failure to provide cooling flow to the tank's heat exchanger could heat up the tank and initiate evaporative cooling. (Failures of the HVAC system that could also do so were not modeled; it was assumed that the facility's response to HVAC failure would be to shutdown the KPC unit).

(2) Failure of spray mixing inside the tank could create hot spots eventually initiating evaporative cooling.

(3) Should the amount of TBP in the tank increase from an inadvertent transfer, then loss of cooling or mixing would lead to a red oil explosion because the criteria for evaporative cooling would be violated.

The initiating event is the loss of cooling or mixing. The event related to the transfer of separate phase TBP was estimated using the approach developed earlier for the failure of the first-stage evaporator caused by the common pathways for transporting separate phase TBP to the process vessels in the acid recovery unit. Maintenance of level control addresses the operator's actions needed to provide aqueous makeup to maintain the criteria for success of evaporative cooling on the appropriate branches under conditions (1) and (2) above. The last event in the tree represents the success or failure of venting to maintain the solution temperature at a safe level to prevent a red oil explosion. There are four red oil explosion sequences. Two involve the transfer of large amounts of TBP to the tank after malfunctions in the pulsed extraction columns and subsequent failures of the sampling and density controls. The sequences are very similar to the scenarios under upset accumulation in the first-stage evaporator, and the dominant contributors are similar. The dominant contributor in the venting failure sequence is common-cause failure of plugging of the HEPA filters. In the remaining sequence, it is the failure of the operator to recognize the level alarm and take proper action.

The PRA model for red oil explosion in the evaporator in the oxalic mother liquor recovery (KCD) unit is based on assessing the various pathways by which organics are transferred to this vessel. Two scenarios with their respective event trees are modeled. In the first one, the initiating event is solvent transfer by mechanical entrainment; in the second one, the initiating event is a severe process malfunction leading to the transfer of a large amount of solvent. Both event trees consider the following events in sequence: the success of the wash column in breaking up and separating the entrained organics, the slab settler's effectiveness in preventing the transfer of any separate phase organics in excess of their solubility limit, and sampling for organics in the KCA batch constitution tanks. The second scenario has another top event—sampling in drip trays—in which samples of leakage are analyzed for their organic content before transfer to the KCD unit. Slab settler failures involve failures of the density controls, which were modeled by fault trees; operational failures that were taken from a supporting document on the slab settler's operation; density monitor failures, which were analyzed by fault trees; and loss of the integrity of the settler's baffle, estimated from data on corrosion rate. The other top events, except failures in the wash column, were modeled by fault tree methods. Three red oil explosion sequences resulted; in all, the dominant contributors included operational failures of the slab settler, failure of the diluent wash column, and failure of the air lift to stop the transfer of process solution to the KCD unit.

The results of the quantitative assessments showed that the point estimate frequencies of a red oil explosion in various process units are low. These low values reflect the robustness and defense-in-depth character of the multiple strategies employed in the facility to avert them. However, the quantitative estimates must be considered preliminary, as only a limited-scope PRA was carried out for the several reasons discussed earlier.

Overall, however, the BNL study (2009) found that the design proposed for the MFFF appears to have incorporated the lessons learned from previous red oil events by including multiple safety strategies in different temperature regimes to deal with the risk of red oil explosions. Each strategy is implemented through a set of IROFS. The IROFS consist of a combination of active engineered systems or controls, PECs, EACs (human action combined with a physical device that serves as an alarm to alert the operator), and administrative controls (required or prohibited operator actions). Each process or system also encompasses items and features of defense in depth. The application of industry codes and standards instills confidence in the reliability of the equipment selected as IROFS, along with the project's quality assurance program, which will be implemented in compliance with the requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." Within the qualitative definitions of event likelihood set out in 10 CFR Part 70 and NUREG-1718 (NRC, 2000), a red oil explosion can be considered to be highly unlikely at the proposed MFFF.

Experience at French Facilities

The modified PUREX process that will be employed at the MFFF is based on the processes used at spent fuel reprocessing facilities in France, particularly those currently in operation at the spent fuel reprocessing facility at La Hague. The French have extensive experience in studying the risks associated with red oil explosions. In a review of the safety analyses of spent fuel reprocessing facilities in France, the Institut de Radioprotection et de Surete Nucleaire (IRSN) (the French Nuclear Safety Institute) found that the risks of red oil explosions are mainly associated with evaporators in the following units (IRSN, 2008):

- for intercycle concentration which concentrates reextraction solutions of uranium from the first extraction cycle,* before their transfer to the second uranium extraction cycle
- for concentration of fission product solutions from the first extraction cycle
- for recovery of tritiated and nontritiated acid where aqueous acid solutions of low and medium activity are concentrated
- for treatment of oxalic mother liquors which concentrate mother liquors before recycling upstream of the phase of oxalic precipitation of plutonium
- for treatment of organic effluents by distillation

* The staff notes that the first extraction cycle at La Hague consists of the coextraction of uranium and plutonium, followed by sequential partitioning (stripping) of plutonium and then uranium, before the treatment of used solvent. This first cycle is, essentially, the entire solvent extraction portion of the AP process in the MFFF.

As a result, the IRSN recognized the following conditions for the increased risk TBP-nitrate (red oil) reactions (IRSN, 2008):

- a temperature higher than 135 degrees C in the evaporators
- the presence of significant quantities of TBP (and its degradation products) in the aqueous phase to be concentrated

The main safety measures that have been adopted at French reprocessing facilities, as sanctioned by the IRSN include the following (IRSN, 2008):

- To limit the TBP content of aqueous solutions supplying the evaporators, the supply vessels of the units concerned are fitted with solvent-flushing devices to separate, by decantation, any TBP drawn into these solutions. The various aqueous flows from the extraction cycles are washed with diluent in batteries of mixer-decanters or in a centrifugal extractor. Feedback from operation of the installations shows that these measures allow the TBP content in the aqueous phase to be limited to a maximum of a few dozen milligrams per liter.

- Systematic monitoring is conducted to confirm the absence of significant quantities of TBP in the different flows arriving in the evaporators of the units referred to, especially for the few flows which are not subject to washing with diluent.

- When aqueous solutions are transferred to the evaporators, the supply vessels are not completely emptied in order to avoid transferring any TBP which may be floating on top of the aqueous solutions.

- Before being received in distillation treatment units, organic effluents are systematically treated with carbonate and soda and washed with water in the “solvent treatments” of the different extraction cycles, which ensures a relatively low nitrate content in these effluents (a maximum of a few dozen milligrams per liter).

- To limit the temperature well below 135 degrees C in normal operation inside “thermosyphon” evaporators (evaporators in intercycle concentration units, acid recovery units, and oxalic mother liquor treatment units), the temperature of the coolant fluid is kept below that value (the temperatures of heating steam circulating in the boiler and the thermosyphon are around 130 degrees C and 110 degrees C, respectively), and the heating loop is fitted both with a pressure regulator with a high pressure warning and a temperature regulation device with a high temperature warning which, when reached, cuts off the units steam supply. This temperature threshold (generally set at between 145 degrees C and 150 degrees C) allows a temperature below 135 degrees C to be guaranteed anywhere in the evaporator. Finally, the vaporiser is protected by two valves whose loading limits any pressure increase in the cooling circuit and consequently limits the steam temperature.

  For evaporators used in fission product concentration units (boiler type evaporators), the design of the cooling circuit in these units (double heating jacket) ensures that in normal operation the temperature throughout the liquid content remains below the temperature required to trigger the reaction. A complementary study confirmed, through simulation of the temperature of the internal wall of this type of equipment, that the maximum temperature reached at the bottom of the boiler remained well below 135 degrees C.

  Finally, for evaporators in organic effluent treatment units (liquid film evaporators), the cooling circuit is fitted with pressure and temperature regulators with high pressure and temperature warning devices. In addition, the operator is warned of any excessively high temperature of the “residue” in the evaporator. The solvent temperature is also monitored when it leaves the evaporator, with a high temperature warning. These measures guarantee that a temperature of 135 degrees C is not exceeded in these units.

The IRSN assessed these safety measures, specifically in degraded situations, when the various French spent fuel reprocessing plants were commissioned.

To date, after more than 20 years of operating experience, there have been no incidents related to TBP-nitrate (red oil) reactions in the French facilities.

8.1.2.4.5.4 NRC Staff Findings and Conclusions on Red Oil

The NRC staff has considered the input from the independent review conducted by BNL personnel, as well as reviewing operating experience at DOE facilities and the La Hague facility in France, upon which the MFFF AP flowsheet is based. The staff finds that the applicant's approach to render the occurrence of a red oil explosion event highly unlikely is consistent with the experience and guidance available in the literature.

The applicant's safety strategy for red oil, which is discussed in the following paragraphs, is consistent with recommendations set forth in DNFSB-TECH33 (DNFSB, 2003).

The only AP process equipment operating near the temperature limit (greater than 130 degrees C) includes the following:

[redacted]

The safety strategy for [redacted] is to prevent TBP from reaching them (DNFSB recommended control 3). This is implemented through IROFS sampling and through the use of an IROFS slab settler to separate organics from the process stream.

The safety strategy for [redacted] is to maintain a temperature less than 122 degrees C (DNFSB recommended control 1), provide adequate venting (DNFSB recommended control 2), and to limit the quantity of TBP that can enter the evaporator (DNFSB recommended control 3). KPC*EV2000 has IROFS temperature controls on the heating solution to protect against a red oil event, IROFS sampling controls upstream to limit the accumulation of TBP in the evaporator, and venting sized to prevent pressure buildup (in accordance with the TBP mass versus overpressure relationship developed by Fauske and Associates (FAI/94-68)). The nominal nitric acid concentration in the evaporator is maintained at 9.4 N (DNFSB recommended control 4), the nominal process solution temperature is maintained at 65 degrees C, and the evaporator is maintained below atmospheric pressure.

For other equipment (e.g., heat exchangers) capable of heating a solution containing TBP, [redacted] (DNFSB recommended control 1) are put into place to protect against a lower safety temperature threshold (solvent flammability) than that for red oil. The applicant has also performed an analysis of all vessels operating at or below 50 degrees C that may contain TBP to ensure that self-heating from chemical reactions is not credible. This analysis was done to address potential self-heating conditions that may have contributed to the red oil explosion at Tomsk-7 in 1993.

All of the AP vessels that are postulated to contain TBP are vented (DNFSB recommended control 2) to protect against runaway red oil reactions. Two separate criteria are used. Vents on vessels in which the process solution temperatures may exceed 50 degrees C are sized to satisfy the Fauske relationship for credible quantities of organic that could be present. Vents on vessels in which the process solution temperatures do not exceed 50 degrees C are sized to prevent overpressurizations considering off-gassing from chemical reactions and other gaseous flows into the vessel. Valves are prohibited from individual vent lines to ensure that the ventilation paths are not impeded. [redacted] Lastly, any equipment with the potential to have pressure transients (e.g., evaporators) is equipped with pressure monitors and controllers.

[redacted]

Relative to the nitric acid concentration limit (DNFSB recommended control 4), concentrated nitric acid (approximately 13.6 N) exists in only three areas. The first is in the nitric acid reagent unit, where neither TBP nor nuclear material is present. The second is in the acid recovery unit rectification column and concentrated acid distribution tanks, where TBP is prevented via IROFS sampling upstream. The third area is in the oxalic mother liquor recovery unit evaporator and concentrates tanks, where TBP is also prevented via IROFS sampling upstream. Normal acid concentrations in all other areas of the AP process are less than 10 N; and the use of concentrated nitric acid is limited to vessels that operate at temperatures less than 60 degrees C.

The independent review performed by BNL personnel found that the design proposed for the MFFF appears to have incorporated the lessons learned from previous red oil events by including multiple safety strategies in different temperature regimes to deal with the risk of red oil explosions. Each strategy is implemented through a set of IROFS. The IROFS consist of a combination of active engineered systems or controls, PECs, EACs, and administrative controls. Each process or system also encompasses items and features of defense in depth. The application of industry codes and standards instills confidence in the reliability of the equipment selected as IROFS, along with the project’s quality assurance program that will be implemented in compliance with the requirements of Appendix B to 10 CFR Part 50. Within the qualitative definitions of event likelihood set out in 10 CFR Part 70 and NUREG-1718, a red oil explosion can be considered by the staff to be highly unlikely at the proposed MFFF.

The staff also finds that the applicant’s safety strategy for the MFFF is consistent with recommendations and current practices documented by IRSN for operations at French reprocessing facilities and, in some cases, are more conservative. For example, the IRSN noted that risks of red oil explosions are mainly associated with evaporators in the following units (IRSN, 2008):

- for intercycle concentration which concentrates reextraction solutions of uranium from the first extraction cycle before their transfer to the second uranium extraction cycle

- for concentration of fission product solutions from the first extraction cycle
- for recovery of tritiated and nontritiated acid where aqueous acid solutions of low and medium activity are concentrated
- for treatment of oxalic mother liquors which concentrate mother liquors before recycling upstream of the phase of oxalic precipitation of plutonium
- for treatment of organic effluents by distillation

The MFFF will employ only three of the five noted unit operations with which the risk of red oil explosions is associated. The MFFF flowsheet does not concentrate the plutonium product stream by evaporation as is done in reprocessing facilities such as La Hague and THORP in the United Kingdom (and as was done with the uranyl nitrate stream at the Hyderabad facility). There are no evaporation units in the MFFF flowsheet until after the plutonium product has been recovered by oxalate precipitation. Before oxalate precipitation, the plutonium product solution is subjected to [redacted] If TBP or any of its degradation products were present in detectable amounts, it would be discovered in the sample before transfer to downstream equipment, and the transfer would not be allowed.

Organic effluents (spent solvent) will not be treated by distillation at the MFFF. Solvent will be washed in the solvent recovery (KPB) unit. When the solvent has exceeded its useful life, it will be disposed of through the SRS, rather than recovered by distillation.

Among the safety measures that have been adopted at French reprocessing facilities is to limit the TBP content of aqueous solutions supplying the evaporators. The supply vessels of the units are fitted with solvent-flushing devices to separate, by decantation, any TBP drawn into these solutions, and the various aqueous flows from the extraction cycles are washed with diluent. Diluent washing of aqueous streams to remove residual entrained solvent is performed in the purification unit (aqueous raffinates) and the KPB unit of the MFFF. In addition, the plutonium nitrate product stream passes through a slab settler [redacted] before moving downstream to the oxidation columns. The slab settler's function is to act as a decanter by which to remove any solvent that may remain entrained in the aqueous stream. Also, vessels subject to the evaporative cooling strategy (i.e., process vessels in the nitric acid recovery (KPC) and liquid waste (KWD) units) have IROFS administrative controls that require periodic flushing of accumulated solvent from the vessel. The vessels are to be flushed at intervals of 6 months or 1 year, depending upon the size of the vessel.

Another safety measure that has been adopted at French reprocessing facilities is to systematically monitor for significant quantities of TBP in the different flows arriving in the evaporators, especially for the few flows that are not subject to diluent. The MFFF red oil safety strategy employs [redacted] to ensure the absence of organics (TBP) before allowing transfers to vessels that operate at temperatures above the solvent safety-basis temperature limit.

The French reprocessing facilities also limit the temperature below 135 degrees C in normal operation inside thermosyphon evaporators (evaporators in intercycle concentration units, acid

[redacted]

[redacted]

In addition, equipment with the potential for pressure transients (e.g., the KPC and KCD evaporators) is also equipped with normal pressure monitors and controllers.

The NRC staff found that the applicant's safety strategy for preventing potential runaway TBP-nitric acid (red oil) reactions is consistent with DNFSB guidance documents (DNFSB, 2003), practices employed at DOE facilities, and practices employed at the La Hague facility in France.

The staff finds that the applicant has described the facility, equipment, and processes in sufficient detail to meet the requirements of 10 CFR 70.22 and 10 CFR 70.65, consistent with the acceptance criteria of NUREG-1718, Section 8.4.3.

The credited IROFS discussed above provide a diverse method to control each of the parameters that can contribute to red oil runaway reactions. The applicant has also committed to have fail-safe and redundant active engineered IROFS [redacted] and use them in combination with passive engineered IROFS [redacted] and administrative IROFS [redacted]).

The NRC staff finds that the applicant's approach is acceptable for complying with the single-failure criterion. The single-failure criterion, in combination with consistency with DNFSB recommendations, the application of experience from DOE facilities and the La Hague facility, management measures (as described in Chapter 15 of the LA), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS, give the NRC staff reasonable assurance that this high consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61.

The applicant uses features to reduce the challenge to IROFS, where practical. [redacted]

[redacted] The NRC staff finds that these features comply with the defense-in-depth requirements of 10 CFR 70.64(b).

The staff finds with reasonable assurance that the applicant has identified the hazards and accident sequences associated with TBP-nitric acid (red oil) reactions and credited IROFS sufficient to meet the performance requirements of 10 CFR 70.61, consistent with the acceptance criteria of NUREG-1718, Sections 7.4 and 8.4.

8.1.2.4.6 Plutonium (VI) Oxalate Explosion (EXP12)

Event Description

Plutonium can exist in several different valances in aqueous solutions. The most common valances are (III), (IV), and (VI). Pu (VI) is initially created in the dissolution (KDB) and dechlorination/dissolution (KDD) units, where it is reduced to Pu (IV), using hydrogen peroxide, before entering the purification (KPA) unit. If Pu (VI) is not fully reduced, it may enter the purification unit. Pu (VI) may also be introduced to the purification unit from the oxalic mother liquid recovery (KCD) unit. The KPA unit is designed to reduce any Pu (VI) to Pu (III) using HAN. However, if Pu (VI) is not adequately reduced in the purification unit, or if it is abnormally generated in the air stripping column, it may reach the precipitator of the oxalic precipitation, filtration, and oxidation (KCA) unit. If Pu (VI) reaches the KCA precipitator, it may be precipitated as Pu (VI) oxalate [i.e., $PuO_2C_2O_4 \cdot 3H_2O$], which would then be introduced to the calcining furnace. The temperatures in the calcining furnace can reach up to 600 degrees C. Upon heating, Pu (VI) oxalate exhibits an endothermic peak around 142 degrees C because of dehydration (it is usually present as the trihydrate) and a subsequent rapid exothermic peak at approximately 219 degrees C associated with rapid decomposition of the oxalate. If a sufficient quantity of Pu (VI) oxalate is introduced to the calcining furnace, an explosion could occur.

Safety Strategy

The applicant's safety strategy for Pu (VI) oxalate explosions involves the application of IROFS to meet the performance criteria of 10 CFR 70.61. Pu (VI) oxalate explosions are postulated to occur in the KCA calcining furnace if Pu (VI) reaches the KCA unit where oxalic acid is present.

[redacted]

Staff Evaluation and Findings

The NRC staff finds that the applicant has adequately evaluated the hazards with the potential to lead to this event and that the safety strategy of preventing transfer of process fluids containing greater than [redacted] of Pu (VI) from the KCA batch constitution tanks to the precipitators to be acceptable.

Section 11.2.1.3.12 of this SER discusses the staff evaluation of the process sampling control. Based on the determination of reliability and robustness of the identified IROFS sampling, the NRC staff finds that this approach is acceptable for complying with the single-failure criterion. The single-failure criterion, in combination with management measures (as described in Chapter 15 of the LA), quality assurance requirements (described in the MPQAP), and the use of codes and standards for engineered IROFS give the NRC staff reasonable assurance that these high consequence scenarios are highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61.

[redacted]

8.1.2.5 Habitability Issues

The purpose of this section is to review the criterion used by the applicant to ensure the habitability of the facility during routine operations and during an accident or event.

In the CAR phase, while no specific required operator action had been identified, MOX Services assumed that there would be required operator actions to be performed in the emergency control room (ECR) in order to meet 10 CFR 70.61 performance requirements. Based on this assumption, the ECR would be required to maintain a habitable environment (i.e., protect the operator from radiological or chemical exposures or both). No PSSCs were identified to protect the worker from radiological exposure. In determining adequate protection for the workers, MOX Services proposed using TEEL-2 limits for protection during an event, while the NRC staff believes that TEEL-1 are the appropriate limits. The NRC staff position resulted in the following construction authorization condition (NRC, 2005):

> In addition to the safety functions and design bases specified in the revised CAR, the facility will be designed so that a safety function of the Emergency Control Room (ECR) Air Conditioning System will maintain hazardous chemical concentrations in each ECR below CAR Table 8-5 TEEL-1 limits for the duration of credible hazardous chemical release events.

It was recognized that during the performance of the ISA, the applicant would do one of the following:

- Provide acceptable justification that the TEEL-2 limits were appropriate for protection of the operators in the ECR from chemical exposure.

- Identify IROFS that protect the operators to TEEL-1 limits.

- Demonstrate that there are no required operator actions in the ECR to meet 10 CFR 70.61 (which could only be validated when the ISA was complete). This would make the TEEL-1 or TEEL-2 issue moot.

The safety function of the ECR HVAC (HVC) system is to provide cooling to the ECRs, emergency electrical rooms, emergency electronics rooms, and emergency battery rooms. The system provides these functions during normal and emergency conditions, including postulated natural phenomena and external radiological events, as well as leaks and spills of hazardous chemicals. The HVC system also provides ventilation to remove hydrogen from the emergency battery rooms.
For releases on the MFFF site, the chemical consequences to outside receptors were bounded by calculating the consequences of the largest possible release of each hazardous chemical in inventory at the MFFF. Only accident sequences involving those chemicals that could potentially cause adverse health effects to the IOC, irreversible health effects to the site worker, or concentrations exceeding standards at outside air intakes were evaluated further. As a result, the only chemicals requiring further evaluation by the applicant were nitrogen tetroxide, hydrazine monohydrate, and uranium (MOX, 2009b). However, the applicant's ISA did not identify any safety function to be performed in the ECR during a radiological or chemical event. Thus, the HVC system is not credited with providing a habitable environment.

The ECR HVC system is equipped with two 100-percent capacity HEPA filter/hazardous gas removal filtration units: one for train A and one for train B. Each unit consists of a stainless steel housing, an inlet isolation damper, one stage of prefilter, one stage of HEPA filter, two stages of hazardous chemical adsorption filter media, a second stage of HEPA filter, and an outlet isolation damper. The HEPA filter media is a glass boron silicate microfiber and contains a waterproofing binder that increases strength under wet and dry conditions. The HEPA filters have a particulate removal efficiency of 99.97 percent for 0.3-micron particles. Injection ports are upstream and test ports downstream of the chemical adsorbers to test their removal efficiency. The filtration units are seismically qualified.

The outside air intake for each HVC train is located on a different side of the shipping and receiving building (BSR) and is approximately 50 feet above the ground. Multiple sensors monitor each supply air inlet for habitability protection.

Smoke detectors with alarms are also provided in the ductwork downstream from each air conditioning unit and inside the emergency battery and electrical rooms. When smoke is detected, a signal is sent to the ECR panel. Upon seeing the alarm, the operator can manually switch the HVC system to the filtration mode.

Each emergency battery room is provided with a hydrogen sensor. If an excessive amount of hydrogen gas is detected in either battery room, this condition is annunciated in the control room and the operator takes appropriate action.

The HVC system is operated in either an automatic or manual control mode. In the automatic mode, which is the normal mode of operation, the control logic is automatically implemented through hardwired connections between sensors and actuators. The HVC system does not use programmable logic controllers. The manual control mode is used at the operator's discretion to contend with emergencies and when maintenance operations are performed on the HVC system.

When the HVC system is in the automatic control mode, both the train A and train B subsystems are in operation to maintain the temperature and positive pressure of their respective emergency designated rooms and are independent of other MFFF ventilation systems. This helps ensure the availability of the ECRs when the main control room is unavailable. The pressurizing fans are continuously run, maintaining a positive pressure differential in the ECRs. The pressurizing fans are operated at a fixed speed, and the room dampers are periodically manually adjusted to maintain the proper pressure differential. The emergency battery rooms are separately exhausted to prevent the buildup of potentially flammable concentrations of hydrogen gas.

Under normal conditions, the airflow bypasses the HEPA filter/hazardous gas removal filtration units. The four air conditioning units load and unload as necessary to maintain the ECRs and the emergency electrical rooms at the desired temperature. When necessary, the duct-mounted electric heaters automatically cycle to maintain temperatures in these rooms at the desired setpoint.

Each ECR air intake is continuously monitored for radiation, smoke, and hazardous chemicals. Upon detection of radiation, smoke, or a hazardous chemical above allowable limits in the intake air, the HVC system will automatically switch to the filtration mode. An alarm is transmitted to the ECR, the isolation damper in the outside air intake ductwork closes, the isolation damper in the corridor air supply ductwork opens, the pressurizing fan suction to the building corridor opens, and the flow is routed through the HEPA filter/hazardous gas removal filtration units.

The applicant stated that the allowable limits in the intake air to the ECRs for radiation, smoke, and hazardous chemicals are as follows:

- Radiation: Sampling for airborne radiation in the inlet to the HVC system is in accordance with American National Standards Institute (ANSI) 13.1-1999, “Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities” and modified for permissible thresholds for air inlets and not exhausts. The allowable limit on intake is in accordance with Table 1 of Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage,” to 10 CFR Part 20, “Standards for Protection against Radiation.”

- Smoke: Standard duct smoke detector, NFPA 72 (National Fire Alarm and Signaling Code), duct size, airflow rate, background, smoke from wildfires, contaminants (e.g., dust).

- Hazardous Chemicals (based on NIOSH, 2005): Hydrazine and dinitrogen tetroxide limits will be maintained below TEEL-2 concentrations, as shown in Table 8.3-4 of the LA (MOX, 2009a). For hydrazine monohydrate, this limit corresponds to 0.06 milligrams per cubic meter (mg/m^3); for dinitrogen tetroxide, the TEEL-2 limit is 15 mg/m^3. It is noted that the TEEL values for hydrazine are based on ERPG concentrations.

<u>Staff Conclusions on Habitability</u>

Upon completion of the ISA, the applicant did not identify a credible event that required operator actions in the ECR in order to meet 10 CFR 70.61 performance requirements. As a result, the applicant did not identify any IROFS function to maintain control room habitability for the ECR air conditioning system. Therefore, the HVC system is not credited with providing a habitable environment during a chemical release event. The staff finds that the applicant's ISA result and subsequent conclusion that there are no credible events that require ECR operator action to meet the performance requirements of 10 CFR 70.61 and satisfy the intent of the construction authorization condition stated above.

The staff reviewed the applicant's assessment of ECR habitability and agrees that there is no identified accident sequence in which action of an ECR operator is a safety control to prevent or mitigate the consequences of that accident sequence. As a result, ECR habitability is not relied on for safety and thus does not need to be IROFS.

8.1.2.6 Baseline Design Criteria

Chapters 8 and 11 of this SER discuss the design bases of the AP process associated with chemical processing.

The applicant stated that the BDC, as described in 10 CFR 70.64, were applied from the outset of the MFFF design work and were primarily focused on physical design and facility features, with the intent to achieve a conservatively designed facility tolerant of both process upsets and human errors (Section 12.0 of the LA (MOX, 2009a)). The applicant stated that information demonstrating compliance with these criteria is provided in the applicable chapters of the LA (MOX, 2009a).

To ensure that all event sequences with consequences exceeding the low consequence threshold of 10 CFR 70.61 meet the performance requirements identified in 10 CFR 70.61, the applicant applied the following qualitative design criteria and commitments to those events and the associated IROFS:

- application of the single-failure criteria or double contingency (for nuclear criticality)
- application of the provisions of 10 CFR Part 50, Appendix B, and NQA-1
- application of industry codes and standards
- management measures, including surveillance of IROFS (i.e., failure detection and repair, or process shutdown capability)

For those credible events in which the single-failure criterion or double contingency are not applicable (i.e., sole IROFS or passive IROFS feature), IROFS features are identified and the commitments for IROFS listed above are applied.

Chapter 8 of NUREG-1718 (NRC, 2000) contains guidance and references to other peer-reviewed work on the subject of chemical safety. The applicant indicated that reagents are stored and chemical mixtures are prepared in the reagent processing building and the reagent storage part of the AP area. They are generally separated from each other and radioactive materials. The applicant will avoid mixing incompatible materials by using appropriate designs, controls, and procedures. The AP and MP facilities are broken down into process functional units, which are made up of one or more subassemblies performing consistent and elementary tasks. The applicant stated that the breakdown into control functional units allows each entity to be operated relatively independently in the given operating mode. The staff review notes that this separation and independence is consistent with accepted industry practices for safe operations.

The applicant will control process storage and operation conditions to prevent exothermic and potential autocatalytic reactions in the reagent processing building and the AP and MP areas. Autocatalytic and exothermic reactions of chemicals will be prevented through the application of IROFS and defense-in-depth features, which prevent the challenge to IROFS functions. The applicant has adequately identified these controls, along with management measures and applicable codes and standards to ensure that the IROFS are available and reliable to perform their safety function when needed.

The applicant has demonstrated that there is reasonable assurance that the IROFS will be sufficiently reliable and available. This assurance is further reinforced through the use of standard nuclear industry engineering practices (e.g., reasonably and generally accepted good engineering practices). These practices are incorporated into the facility general design philosophy, design bases, system design, and commitments to applicable management measures. These practices ensure that applicable industry codes and standards are utilized, adequate safety margins are provided, engineering features are utilized to the extent practicable, the defense-in-depth philosophy is incorporated into the design, and the IROFS will be appropriately maintained.

The staff's review, as summarized above, finds that the applicant has provided sufficient information to meet the requirements of 10 CFR 70.64(a)(3) and 10 CFR 70.64(a)(5).

8.1.3 Chemical Process Safety Interfaces

Interfaces with Programmatic Areas

The MFFF is a relatively highly automated facility based in large part on the design and operating experience of existing facilities (the La Hague and MELOX facilities in France). The highly automated nature of the facility limits the number of personnel activities designated IROFS. Chapter 12 of the LA (MOX, 2009a) describes the application of human factors engineering to MFFF IROFS in detail.

Interfaces with Management Measures

Management measures supplement MFFF IROFS by providing the administrative and programmatic framework for configuration management, maintenance, training and qualification, procedures, audits and assessments, incident investigation, and records management. The MPQAP describes the quality assurance program and Chapter 15 of the LA discusses management measures.

Personnel responsible for performing activities involving chemical safety will be qualified and trained in accordance with the MFFF training and qualification program; specifically, applicable training for IROFS associated with chemical hazards will be provided.

Activities associated with IROFS will be conducted in accordance with approved procedures. MFFF plant procedures govern operations, maintenance, and administrative actions to ensure that IROFS are operated in a manner consistent with the results of the ISA. Plant procedures associated with IROFS will take into account chemical hazards, as well as radiological and criticality hazards, as appropriate for the activity.

Audits and assessments will be used to determine the effectiveness of management measures, including those associated with chemical safety. Audit and assessment attributes (e.g., independence of auditors from personnel responsible for the chemical safety activities being audited, reports to management) will be consistent with those for other MFFF IROFS.

Incident investigation activities identify corrective actions for, and root causes of, incidents that involve MFFF IROFS, including those related to chemical safety. Chapter 15 of the LA provides a general discussion of the applicant's incident investigation and corrective action implementation process.

The applicant will control chemical safety records in accordance with configuration management processes, the requirements of the MPQAP, and the records management program. Chemical safety records are processed and retained in the same manner as records associated with other IROFS and related programs.

The staff finds that the applicant has established controls against risks as described in 10 CFR 70.64(a)(5) in a manner consistent with those established for other areas, such as radiological safety and criticality, along with the associated management measures. Therefore, the staff finds that the applicant's safety program, as it pertains to chemical safety, is acceptable.

8.2 Evaluation Findings

The staff reviewed the LA (MOX, 2009a) for the MFFF to possess and use special nuclear material according to Chapter 8 of NUREG-1718 (NRC, 2000). The staff evaluated information provided by the applicant in the LA, ISA Summary (MOX, 2009b), nuclear safety evaluations, various technical reports, and responses to requests for additional information. The staff found that the applicant's facility and system design and facility layout pertaining to chemical safety are based on defense-in-depth practices. The staff also found that the applicant's facility design and IROFS provide adequate protection against chemical risks produced from licensed material, events which affect the safety of licensed material, and hazardous chemical produced from licensed material at the facility for routine operations, off-normal conditions, and potential accidents. Based on the review of the LA, the staff concluded that the applicant adequately described and assessed accident consequences having potentially significant consequences and effects that could result from the handling, storage, or processing of licensed materials. The LA and ISA Summary identified chemical process hazards and potential accidents that affect the safety of licensed material and established safety controls to ensure safe facility operation. To ensure that the performance requirements in 10 CFR Part 70 are met, the applicant will maintain the availability and reliability of these controls. The staff reviewed these safety controls and the applicant's plan for managing chemical process safety and its potential effects on licensed radioactive materials and finds them acceptable.

The staff concludes that the applicant's plan for managing chemical process safety and the chemical process safety controls meet the requirements to possess and use special nuclear materials in accordance with 10 CFR Part 70.

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49 CFR Part 178, Specifications for Packagings.

9.0 RADIATION SAFETY

In Section 9 of the "Mixed Oxide Fuel Fabrication Facility—License Application" (MOX, 2010a), Shaw AREVA MOX Services (the applicant) described the proposed radiation protection (RP) program for the mixed oxide fuel fabrication facility (MFFF). The program addresses both radiation safety design and implementation of the RP program. In performing its review, the staff used the review guidance in NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility" (also referred to as the SRP) (NRC, 2000). The SRP addresses design and implementation separately but with similar requirements for each topic. To avoid duplication in this chapter of the safety evaluation report, the U.S. Nuclear Regulatory Commission (NRC) staff first addresses in general terms the design features applied to each topic in Section 9.1 of the SRP, "Radiation Protection Program," including the principle of as low as reasonably achievable (ALARA), facility design features, source identification, ventilation, and the Integrated Safety Analysis (ISA) Summary (MOX 2010b). Next, the staff presents its review of the RP program as it covers in depth each topic in Section 9.2 of the SRP, including ALARA, procedures, training, air sampling, contamination control, exposure, respiratory protection, and instrumentation. The application includes the commitment to implementing design features throughout the facility, which is evident in the RP program.

9.1 Regulatory Requirements

The staff reviewed how the information in the license application (LA) (MOX, 2010a) addresses the regulatory requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection against Radiation," and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material." Chapter 9 of the SRP states the specific regulatory references (NRC, 2000).

9.2 Facility Design Features

9.2.1 As Low as Reasonably Achievable

The ALARA principle involves keeping radiation exposures as far below the regulatory limits as is reasonably achievable. ALARA is evident in the design features as demonstrated through the use of automated and remote controlled systems, isolation of the processes, monitoring, and shielding. The containment systems include gloveboxes, process cells, and ventilation which keep the radioactive material out of work areas. The confinement systems are designed to minimize buildup of contamination and simplify cleaning through features such as smooth surfaces, rounded corners, and automatic removal of radioactive material. The separation between personnel and radioactive material is the primary design feature used throughout the MFFF to implement ALARA.

The applicant's design staff is trained and qualified to identify and minimize radiation hazards. These individuals, with assistance from the radiation protection manager (RPM), regularly review operations to ensure that issues arising from criticality control, exposures, shielding, and other areas are resolved and iteratively incorporated into the design. ALARA principles are incorporated into the design in the following ways: confinement of radioactive material within process equipment and in gloveboxes, multiple-zone ventilation that sweeps air from low to high potential contamination zones, continuous remote monitoring with local and remote readout and alarms, automated and remotely operated equipment, removal of radioactive material before maintenance, shielding that is commensurate with the penetrating power of the radiation, equipment designed to require a minimum of maintenance or repair, facility layout to keep administrative activities away from radiation areas, and personnel and area radiation monitoring. These characteristics demonstrate that ALARA is incorporated into the facility design features and meets the guidance in Section 9.1.4.1 of the SRP.

9.2.2 Facility Design Features

The site drawings and descriptions depict the radiation protection provided by the facility design features. The design incorporates documentation of features including scaled drawings of the facility superimposed with radiation zones based on expected worker occupancy; radiation shielding calculations for each zone; definitions of the radiation sources and features relied on to reduce doses ALARA; location of radiation protection equipment; general requirements for radiation detectors and alarm systems; locations of permanent shielding and confinement design; locations and access control points for radiation areas, controlled areas, and restricted areas; and location of change rooms. The RP principles are also incorporated by design into the facility procedures. These procedures, developed by the RPM, are audited regularly to ensure that they remain accurate and include ALARA principles. Radiation work permits (RWPs) are implemented for nonroutine tasks that do not have standard existing procedures. These RWPs require the design to be approved by the RPM or his or her representative. These commitments demonstrate that RP principles are incorporated into the design features through written procedures and documentation and meets the guidance in Section 9.1.4.2 of the SRP.

9.2.3 Source Identification

Internal exposure to special nuclear material (SNM) has the largest potential to result in radiation doses at the MOX facility. Therefore, the design relies on containment of the material through the use of confinement vessels, gloveboxes, sealed containers, and ventilation throughout the facility to prevent internal exposure. Direct radiation is the second source of exposure. Therefore, the confinement vessels incorporate design features, such as borated concrete and lead glass, for shielding to attenuate ionizing radiation. Automation and remote controls are used to limit exposure times and maximize distance between the source and the individual. The design incorporates routine monitoring of both airborne and direct radiation in work zones to detect contamination and provide estimates of actual exposure. The design incorporates consideration of sources of radiation for implementation of RP principles and meets the guidance in Section 9.1.4.3 of the SRP.

9.2.4 Ventilation Systems and Glovebox Design

The MFFF ventilation design incorporates radiation protection principles throughout the facility. The ventilation is divided into zones that move air from less contaminated areas to more contaminated areas. Within each zone, clean air enters the work area from the head level and travels towards the foot level to reduce inhalation of contamination. The design also minimizes the spread of contamination by maintaining a negative pressure gradient between zones. Air monitors and pressure differential monitors ensure that any degradation in containment barriers is rapidly identified so corrective action can be implemented. The design incorporates airlocks between zones to minimize migration of airborne contaminants. Redundant power and heating, ventilation, and air conditioning systems ensure that the ventilation system remains operable. These ventilation system design features throughout the facility maintain internal exposures ALARA and meets the guidance in Section 9.1.4.4 of the SRP.

9.2.5 Shielding

The development of the design features drew extensively on experience from the MELOX and La Hague facilities, which are operated by AREVA in France. MELOX and La Hague operating experience influenced the design throughout the MFFF facility (e.g., the occupancy rates, proximity to radiation sources, stay times, shielding, zoning). Incorporation of hands-on experience in the design of facility features improves the radiation protection of the facility.

Design drawings and descriptions of the shielding for high and very high radiation areas clearly identify the penetrations, shield doors, and labyrinths incorporated to meet the shielding design criteria. Radiation shielding analyses were used by the applicant to verify the shielding for each process room, including the dose rates for each position workers are required to take to perform routine and non-routine maintenance. A radiation shielding test program will be implemented prior to the start of operations for protection of personnel from high radiation dose rates.

The applicant used several industry standard computer codes in the shielding calculations (e.g, SCALE, MCNP, and SNID). The shielding design complies with 10 CFR §20.1406 requirements for the minimization of contamination and uses the reference facilities' design experience for guidance. The shielding design features are sufficient to minimize external and internal doses and meets the guidance in Section 9.1.4.5 of the SRP.

9.2.6 Integrated Safety Analysis Summary

As indicated above, passive (e.g., vessels, gloveboxes, containers) and active (e.g., ventilation, negative pressure, direction of airflow) confinement barriers provide the primary design features to prevent or minimize internal exposures. The effectiveness of these features is verified through the use of air monitors. Shielding, minimizing the time spent in radiation work zones, and distance from the source material (e.g., through automation, remote control, work zones) are the primary design features to minimize external exposures. The ISA identifies accident scenarios, including loss of confinement events that could result in internal and external exposures. Items relied on for safety and management measures are incorporated into the design to comply with the performance requirements and meets the guidance in Section 9.1.4.6 of the SRP.

9.2.7 Evaluation

The staff followed the guidance in Section 9.1.6 of NUREG-1718 (NRC, 2000) in reviewing the LA for the MFFF to possess and use SNM. Based on its evaluation of the LA and the ISA, the staff finds that radiation protection principles are incorporated throughout the facility design. The applicant supplied information on the radiation safety design features and design process that demonstrates, with reasonable assurance, that radiation doses will be within the limits of 10 CFR Part 20 and will be ALARA. The applicant considered contamination control, decommissioning facilitation, and waste minimization in developing the design features of the facility, as required by 10 CFR § 20.1406, "Minimization of contamination." The applicant also incorporated radiation safety design features resulting from its radiation safety design review and from radiation dose experience gained during the operation of AREVA's French facilities, MELOX and La Hague.

The NRC staff concludes that there is reasonable assurance that the applicant's radiation safety design process and design features are adequate and, in concert with an effective RP program as outlined in SRP Section 9.1 (NRC, 2000), satisfy the requirements of 10 CFR Part 20 and 10 CFR Part 70.

9.3 Radiation Protection Program Implementation

In Section 9 of the LA (MOX, 2010a), the applicant described the proposed RP program for the MFFF. MOX Services committed to implementing a quality RP program consistent with Regulatory Guide (RG) 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable," Positions C.1.a and b (NRC, 1978). This portion of the RG describes criteria for establishment of an ALARA program, including management oversight, an effective measurement system, regular audits, approved procedures, and application of sufficient resources. The RP program is developed, documented, and will be implemented commensurate with the risk posed by processing MOX fuels such that operations will meet the requirements of 10 CFR Part 20.

The RP program is designed, monitored, and maintained by the RP function and overseen by an RPM with support throughout operations from the engineering function. Section 9.2.2 of the LA (MOX, 2010a) outlines the RP program's organizational structure and the responsibilities of key program personnel. The plant manager is responsible for ensuring the health and safety of the public and workers and protection of the environment. This includes compliance with applicable NRC regulations and the facility license. With the support of the RP function, the RPM is responsible for implementing the RP program. The RPM and staff, including contractors, will also be responsible for the following:

- establishing the RP program
- ensuring adherence to the RP program in operations
- establishing and maintaining the ALARA program
- ensuring that ALARA principles are incorporated in training and practiced by all personnel
- adequately staffing the RP function with individuals qualified to conduct their assigned responsibilities
- reviewing and auditing the efficacy of the program in complying with NRC and other governmental regulations and applicable RGs
- modifying the program, based on experience and facility history
- generating and maintaining procedures associated with the program
- establishing and maintaining a radiation safety training program for personnel working in restricted areas
- ensuring that adequate surveys are conducted to maintain cleanliness and maintain exposures ALARA
- monitoring and documenting worker doses, both internal and external

- establishing and maintaining a respirator usage program
- proper handling of radioactive wastes when disposal is needed
- proper ventilation system with filtration to minimize contamination
- establishing and maintaining the radiological environmental monitoring program
- calibrating and conducting quality assurance activities for all radiological instrumentation, including verification of required lower limits of detection or alarm levels
- posting the restricted areas and, within these areas, posting Radiation, Airborne Radioactivity, High Radiation, and Contaminated Areas, as appropriate, and developing occupancy guidelines for these areas as needed

The staff reviewed these topics and summarized them in the remainder of this chapter of the safety evaluation report in accordance with the acceptance criteria in NUREG-1718, Chapter 9.2 (NRC, 2000). The staff conducted an item-by-item evaluation against this SRP to ensure that each topic was addressed. Details of the staff evaluation follow.

9.3.1 ALARA Program

The RP program uses written policies and procedures to ensure that occupational radiation exposures are maintained ALARA and that such exposures are consistent with the requirements of 10 CFR § 20.1101, “Radiation protection programs.” All design reviews include ALARA considerations. The applicants’ staff tracks to completion any recommendations resulting from the review.

The goals of the ALARA program include maintaining occupational exposures, as well as environmental releases, as far below regulatory limits as is reasonably achievable. The design of facility systems plays a large role in achieving these goals. For example, inhalation exposures are minimized by confining radioactive material within sealed process areas, separated by barriers from personnel access areas. Key ALARA elements of time, distance, and shielding are implemented through automation, remote control, and permanent shielding. These characteristics are designed into the facility wherever feasible. Access is restricted to process rooms through properly ventilated gloveboxes. When direct access to process areas is required for maintenance or other reasons, radiation sources are removed from the work area.

All tiers of the MFFF staff and management structure support ALARA principles. For example, MFFF management ensures that ALARA is emphasized throughout the facility by communicating expectations to the staff through policy statements, regular audits, and empowerment of the RP function to intervene if operations are determined to be unsafe. Management receives periodic reports from the RPM, who is responsible for reviewing key program such as training, maintenance, and operating procedures for consistency with ALARA principles.

The RP function conducts periodic audits to verify that ALARA is incorporated into the facility programs. These audit results are communicated to management for review and input to the staff. The RP function is delegated authority to intervene in any practice that is determined to be unsafe. Management ensures that ALARA principles are incorporated throughout the MFFF and that the RP function has sufficient authority to ensure implementation.

The facility has a well-staffed ALARA Committee, with members including line management and operations management personnel and the RPM. The Committee is responsible for overseeing the review and improvement of the RP program. It reviews the audits of the RP program at least annually and evaluates major design activities, operations activities, and plant modifications that could affect ALARA goals. It also conducts trending analysis to ensure that the goals are accomplished. The Committee meets frequently, in accordance with internal procedures, and more often during activities with the potential for unusual exposures. Recommendations made by the Committee are tracked to completion. Reports on the status of the meetings are provided to management at least annually.

The ALARA program is implemented throughout the facility with oversight from the RP function and the ALARA Committee. Progress is monitored through regular audits and trending. This NRC staff finds that the applicants' program provides reasonable assurance of compliance with 10 CFR 20.1101(b) and the acceptance criteria in NUREG-1718, Section 9.2.4.1 (NRC, 2000).

9.3.2 Organizational Relationships and Personnel Qualifications

The RP function operates under the Health, Safety, and Environment function (ES&H), which conducts licensing and regulatory compliance as described in LA Section 4.2.5. The RPM, who reports to the ES&H Licensing Manager, oversees the RP function. The RPM's role includes ensuring adherence to the RP program, establishing RP policy, administering the RP program, reviewing facility modifications, and managing RP staff. The RPM has direct access to senior management regarding all matters involving RP, is skilled in the interpretation of RP data and regulations, and is familiar with the operation of the facility and RP concerns at the site.

The RPM receives support from senior health physicists and senior staff who are empowered to substitute for the RPM when needed. The RP technicians work under the supervision of these senior staff and conduct the day-to-day responsibilities including surveys, dosimetry, bioassay, independent oversight of exposures, and calibration of instrumentation. Section 9.2.2 of the LA (MOX, 2010a) describes the training of the RPM, senior staff, and technicians for their respective tasks.

The members of the RP function must be qualified commensurate with their responsibilities. The RPM has, as a minimum, a bachelor's degree (or equivalent) in health physics, engineering, or a scientific field and at least 4 years of experience in radiological protection. Certification by the American Board of Health Physics or an additional 4 years of experience may be substituted for the degree requirements. Senior RP staff will have 4-year degrees in science or engineering with at least 1 year of experience at a nuclear facility. Management may waive specific qualifications for the RPM on a case-by-case basis, provided that supporting staff have equivalent qualifications. The RP technicians must have a high school diploma or equivalent and work under the direction of a senior technician or supervisor.

The applicant provides sufficient resources in terms of staffing and equipment to implement an effective RP program for all shifts at the facility. The RP program is independent of the facility's routine operations and is focused on implementing sound RP principles necessary to achieve ALARA goals. The NRC staff finds that the applicant's organization and staffing of the RP function provide reasonable assurance of compliance with 10 CFR 70.23(a)(2) and the acceptance criteria in NUREG-1718, Section 9.2.4.2 (NRC, 2000).

9.3.3 Radiation Safety Procedures and Radiation Work Permits

Written procedures are used for all operations involving licensed materials as described in the MOX Project Quality Assurance Plan. The staff is trained to adhere to the written procedures unless an unplanned or unsafe condition arises. Under these conditions, the staff is trained and given authority to stop work and report the conditions to management. Procedures throughout the facility are maintained in accordance with the MOX Project Quality Assurance Plan. In addition to the NRC-approved quality assurance program, the facility also conducts a 5-year overall review of RP procedures, respiratory protection procedures, and operating and maintenance procedures to ensure their continued applicability and accuracy.

Before working in a radiologically controlled area, individuals are required to read, understand, and follow RWPs. RWPs may be general for routine activities, such as daily operations, or may consist of specific instructions for nonroutine activities. RWPs for specific activities are distributed in maintenance packages, which the assigned crew is required to read before undertaking the work.

Various groups throughout the facility may issue RWPs on an as-needed basis, but the RP function specifies the radiological conditions for the work area, stay times, protective clothing requirements, shielding (if required), dosimetry requirements, and any other relevant information. The RPM reviews and approves the RWPs before their implementation. The RP function takes the additional precaution of reviewing new RWPs with the group developing the document to ensure that the appropriate information is incorporated and understood.

The RWPs have a predetermined period of validity, with a specified expiration or termination time listed on the document. RWPs that are used often or for extended periods of time are periodically reviewed for possible improvements in worker protection. Records of RWPs are kept in accordance with the facility document management program as described in Section 15 of the LA. Sufficient information is retained so that auditors can reconstruct the circumstances necessitating the RWP.

The applicant commits to the use of written procedures and RWPs for activities involving exposure to licensed material. Qualified facility staff will maintain and implement these procedures, which demonstrate reasonable assurance of compliance with 10 CFR § 70.23(a)(4), “Requirements for the approval of applications,” and the acceptance criteria in NUREG-1718, Section 9.2.4.2 (NRC, 2000).

9.3.4 Training

The radiation safety training program is designed and implemented to provide a tiered level of knowledge to all personnel who enter radiation-controlled areas. Visitors receive site-specific safety information and must be accompanied throughout the radiation-controlled area by qualified personnel. An individual’s level of training is based on the potential radiological health risks associated with his or her assigned responsibilities. All employees are trained commensurate with the provisions of 10 CFR § 19.12, “Instruction to workers,” as outlined in Section 9.2.4 of the LA (MOX, 2010a).

Individuals must pass a written exam to demonstrate satisfactory completion of classroom training. This training includes such topics as risks of exposure, regulatory and administrative limits, RP concepts, facility-specific emergency actions, event response, and individual responsibilities including ALARA. The training also includes practical demonstrations of donning personal protective equipment, conducting self-monitoring with survey instruments, and methods for decontamination. Individuals who are required to wear respiratory protection receive specialized training and a medical evaluation.

Refresher training is conducted annually, and as necessary, to address changes in policies, procedures, requirements, and the facility ISA. The training program is regularly updated with lessons learned from operational experience and as needed for new material. In addition, the RPM conducts a formal review of the training program at least every 3 years, to ensure that the program is current and adequate.

Line management is fully responsible for ensuring that personnel are properly trained. The personnel are required to acknowledge in writing that they have received and understand the material. The training records are maintained in accordance with internal procedures.

The applicant has developed a comprehensive tiered program to train staff with various levels of responsibility. Personnel receive regular refresher training, and the course materials are regularly updated to incorporate lessons learned from operations. The staff finds that the provisions demonstrate reasonable assurance of compliance with the training requirements in 10 CFR § 19.12 and 10 CFR § 70.23(a)(2) and the acceptance criteria in NUREG-1718, Section 9.2.4.4 (NRC, 2000).

9.3.5 Air Sampling

The air sampling program uses portable air samplers, fixed air samplers, and continuous air monitors (CAMs) to detect airborne contamination throughout the facility. The type and frequency of sampling are based on the potential for exposure and implemented based on ALARA goals. The combination of these sampling methods provides comprehensive monitoring of the facility's airborne contamination.

The CAMs are used extensively throughout the facility, such as at work stations, area monitors, ducts, and stack exhaust, to provide a baseline measure and rapid detection of contamination. The number and location of CAMs are determined by the type of operations and the potential for uptake. The CAMs provide early warning of contamination to personnel by providing dual alarm setpoints. The first alarm setpoint warns individuals that contamination is nearing an administrative limit and corrective actions are necessary. The second alarm setpoint warns individuals to take immediate protective actions. The CAMs also send readouts to the control room and the RP function so the readouts can be recorded for trending purposes and to facilitate the RP response.

Personnel are trained on the procedures they should follow after a CAM alarm. These procedures define the followup actions, which include investigating the cause, determining the level of contamination, implementing corrective actions, and in some cases, conducting a bioassay dose evaluation. The CAMs are also used to collect an air sample, which is processed by the RP function, to monitor the gross activity and the isotopes of concern.

Fixed and portable air samples are used to verify that CAMs are calibrated and operating correctly. They are also used to monitor areas where workers could receive an annual intake of 2 percent or more of the specified annual limit on intake during normal operations. Portable air samplers, such as lapel samplers, are used when individuals work directly with radioactive materials or when a system boundary is opened for maintenance. These samples are analyzed at the conclusion of each shift. The readouts from the CAM and the air samples are analyzed to calculate an individual's internal exposure.

Internal monitoring equipment is maintained in calibration by the RP function in accordance with internal procedures. Operational checks are routinely conducted using check sources.

The airborne sampling program allows for the early detection and appropriate mitigation of airborne contamination. The CAMs provide real-time monitoring and tracking of contamination augmented by air sampling as needed. The NRC staff finds that the applicant's air sampling program meets the acceptance criteria in NUREG-1718, Section 9.2.4.2 (NRC, 2000).

9.3.6 Contamination Control

The contamination control program seeks to prevent the spread of radioactive material through monitoring and decontamination of personnel, equipment, and controlled areas throughout the facility. Multiple measures are implemented to prevent the spread of contamination, including a survey program, access to controlled areas, and personal protective equipment.

Surveys are conducted in process areas on a graded approach. Access and egress areas are surveyed daily, radiological controlled areas with high occupancy are surveyed weekly, and areas located outside the radiological controlled areas are surveyed quarterly. If surveys identify contamination areas or high contamination areas, as defined by Table 9.2-1 of the LA, the RP function posts and restricts access until the area is decontaminated below established limits.

Surveys are used to verify ALARA and guide the development of radiological protection requirements, such as postings, access controls, and personal protective equipment (PPE) requirements. The survey program is conducted in accordance with written procedures maintained by the RP function. Contamination surveys are conducted using portable survey instruments, swipes, and large-area wipes to detect both removable and fixed contamination. These are in addition to the CAMs and air samples used to detect airborne contamination throughout the MFFF radiation-controlled area and associated ventilation systems. Survey results are documented and reviewed by the RP function for possible trends and to identify potential areas for corrective actions. Corrective actions are implemented to isolate contaminated areas, notify personnel, and clean up the contamination as soon as practical.

The applicant will adjust survey frequencies and survey procedures based on localized conditions and historical data. Survey frequencies will be modified only annually, at most, and require at least 10 routine, consecutive surveys that reveal negligible contamination.

Surveys are used to identify areas that should be posted in accordance with 10 CFR Part 20, Subpart J, "Precautionary Procedures." These areas include controlled areas, radiological control areas, contamination areas, restricted areas, airborne contamination areas, among others. Internal MOX Services procedures are established to define the frequency of surveys and monitoring appropriate for each area. An access control program ensures that signs, labels, and other access controls are properly posted and operative.

Transition areas between contaminated and uncontaminated areas are established where individuals can remove PPE, survey themselves, and conduct decontamination as necessary. If contamination is detected, the event is recorded for tracking purposes, and members of the RP function are summoned to assist in decontamination. Personnel are surveyed when they leave a contamination controlled area to ensure that they comply with the administrative limits in Table 9.2-1 of the LA (MOX, 2010a). Protective coveralls or a lab coat is required in most areas of the facility, with additional PPE requirements based on local conditions. The RP function establishes the PPE requirements. Used PPE is bagged, surveyed, and placed in drums before being laundered.

Contamination is limited by processing material within containment barriers and gloveboxes. The ventilation system maintains these process areas at a negative pressure to prevent migration of material through any breach. Materials and equipment released from controlled areas to uncontrolled areas are considered contaminated until they are surveyed and released. The material remains controlled until surveys confirm that the contamination is below the limits in Table 9.2-1 of the LA (MOX, 2010a).

The RP function is responsible for overseeing the control and accountability of sources. Sealed sources are surveyed for leaks at least annually in accordance with internal procedures. High-radiation sources are kept in locked cabinets with access controlled by the RPM.

The contamination control program includes a regular schedule of surveys to identify, isolate and remove contamination. Additional features such as controlled areas, transition zones, ventilation, and PPE further contribute to contamination control. The program relies on surveys and predefined administrative limits to identify areas used to isolate and remove radioactive material. The NRC staff finds that the applicant's contamination control program provides reasonable assurance of compliance with 10 CFR Part 20.1406 and the acceptance criteria in NUREG-1718, Section 9.2.4.6 (NRC, 2006).

9.3.7 External Exposure

Sound ALARA practices such as remote systems operations, confinement systems, and radiation shielding are the primary methods of limiting external exposure. A monitoring program has been established for MFFF personnel, since most individuals are exposed to both photon and neutron radiation. Thermoluminescent dosimeters (TLDs), which are sensitive to beta, gamma, and neutron radiation, are used to monitor doses. Only dosimeters with the appropriate range and sensitivity are used at the facility. Everyone likely to receive an annual dose of 50 millirem (mrem) for visitors or 100 mrem for workers is required to wear a dosimeter. Individuals likely to receive a neutron radiation dose in excess of 1 millisieverts (100 mrem) annually receive neutron dosimetry.

A vendor accredited by the National Voluntary Laboratory Accreditation Program evaluates dosimetry quarterly. In addition to a TLD, personnel within the MFFF process areas wear an electronic pocket dosimeter to provide rapid warning of excessive exposure. The RP function uses routine surveys to identify areas that require heightened monitoring of external exposure.

The applicant has established an annual administrative limit for total effective dose equivalent of 5 millisieverts (mSv) (500 mrem). When an individual's dose approaches the administrative limit, the individual is considered for temporary reassignment. The RPM, in conjunction with the ES&H vice president, may grant special permission for individuals to exceed the administrative dose limit. An investigation and corrective actions are required when administrative limits are exceeded.

External exposures are minimized by implementing ALARA principles and by monitoring exposures with electronic and TLD dosimetry. In addition, the established administrative limits require corrective actions when doses approach or exceed the limit. The NRC staff finds that the applicant's external exposure program provides reasonable assurance of compliance with external exposure limits in10 CFR Part 20 and the acceptance criteria in NUREG-1718, Section 9.2.4.7 (NRC, 2000).

9.3.8 Internal Exposure

As with external exposure, ALARA principles such as remote operations, containment, and monitoring are key to minimizing internal exposures. For mixed oxides, internal exposure exceeds the radiological health risks for external exposures. For this reason, individuals are physically separated from powdered material by gloveboxes and negative pressure barriers.

CAMs are used to monitor exposure continually and are augmented by air samplers, which are used to estimate internal dose, based on the individual's stay time. The CAMs have established setpoints with associated alarms, which notify personnel of a need to take protective actions. Confirmatory air samples are monitored at the conclusion of each shift.

Bioassay sampling is also used to verify internal uptakes and determine the quantity of radioactive material. All personnel who have the potential to receive 10 percent of the annual limit on intake are subject to routine (at least annual) bioassay monitoring. Bioassays are also required for all personnel who have observed facial or nasal contamination, or when a CAM or air sampler indicates a potential uptake. The bioassay program includes urinalysis, fecal sampling, and whole body scans, as determined by the RP function. The bioassay program is conducted according to the criteria of American National Standards Institute/Health Physics Society (ANSI/HPS) Standard N13.22, “Bioassay Programs for Uranium” (ANSI/HPS, 1995). Bioassay minimum detection levels will be established in accordance with ANSI/HPS Standard N13.30, 1996, “Performance Criteria for Radiobioassay” (ANSI/HPS, 1996), and documented in accordance with internal procedures.

The internal exposure controls are based on limiting exposure to airborne radioactive material through ALARA principles. The amount of exposure is thrice monitored using CAMs, air samples, and bioassay measurements. The NRC staff finds that the applicant's program provides reasonable assurance of compliance with 10 CFR Part 20 and the acceptance criteria in NUREG-1718, Section 9.2.4.8 (NRC, 2000).

9.3.9 Summing Internal and External Exposure

The applicant has established a procedure to sum the internal and external exposure values in accordance with 10 CFR 20.1202, “Compliance with requirements for summation of external and internal doses,” as stated in Section 9.2.9 of the LA (MOX, 2010a). Radiation exposures will be recorded and reported in accordance with internal RP procedures, which will be based on RG 8.7, Revision 2, “Instructions for Recording and Reporting Occupational Radiation Dose Data” (NRC, 2005), and RG 8.34, “Monitoring Criteria and Methods To Calculate Occupational Radiation Doses” (NRC, 1992).

The applicant has developed a radiation survey and monitoring program, which incorporates external exposure, air sampling, bioassay, and contamination control. This program provides reasonable assurance of compliance with 10 CFR §20.1202 and the acceptance criteria in NUREG-1718, Section 9.2.4.9 (NRC, 2000).

9.3.10 Respiratory Protection

The respiratory protection program will be implemented in accordance with written procedures that cover individual qualifications, maintenance, and recordkeeping. In general, the MFFF seeks to minimize the need for respiratory protection by limiting airborne contamination in the work area. However, respiratory protection is used when air sample analysis indicates that concentrations equal or exceed 20 percent of the derived air concentrations listed in 10 CFR Part 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage."

Before becoming qualified for respiratory equipment, individuals must undergo a medical evaluation, receive a fit test, and pass specialized training. The level of respiratory protection is determined by the derived air concentration in the area air and the respirator's protection factor. The RP function will ensure that the equipment has sufficient visibility, communication capability, and skin protection for the assigned task. The licensee will maintain records of the individuals' training and use of respiratory protection equipment.

9.3.11 Respiratory Protection and Ventilation

The MFFF incorporates a ventilation system in its design to minimize exposure and the operation of this ventilation system will assist in determining the need for respiratory protection. The ventilation design incorporates air zone flow with well-defined pressure gradients between zones, so that air flows from areas with lesser to greater contamination potential. In addition, room airflow enters at the head level and exits near the floor to further reduce the likelihood of inhalation contaminants. Dust pots are used as pre-filters for dust removal so that licensed material can be recycled.

Ventilation is exhausted through high-efficiency particulate air (HEPA) filters before being exhausted to the plant stack. Redundant critical ventilation systems such as fans, dampers, and filters are provided to ensure continuous operation of the system. Air monitoring and warning systems are designed with a standby power supply to remain operable during a loss of power event. Differential pressure is monitored across HEPA filters to ensure that contaminated exhaust systems are identified and serviced in a timely manner. In addition, redundant HEPA filters are employed to ensure that ventilation systems remain functional. CAMs and pressure measurements are used to detect abnormal leaks in containment barriers. After filtration, exhaust air is emitted through the plant stack after undergoing a confirmatory air purity measurement.

Containment is maintained by handling material in process vessels, gloveboxes, and sealed containers. The material is manipulated remotely or through gloveboxes, which are maintained under a partial vacuum. This ensures that leaks do not result in a release of radioactive material into work areas. Airflow velocities are monitored at system openings (gloveboxes, exhausted enclosures, or ventilation systems serving these barriers) and are designed to remain at a minimum of 0.64± 0.03 meters per second (125±5 feet per minute), even under abnormal conditions. The airflow rate is sufficient to preclude movement of airborne plutonium or uranium between zones and to minimize the potential for worker intake.

The respiratory protection program combines with the ventilation system to minimize exposures to airborne contaminants. The ventilation localizes and removes airborne contamination from the work environment. When these combined methods are insufficient, qualified personnel use respiratory protection equipment to maintain doses ALARA. The staff finds that together, these programs provide reasonable assurance of compliance with 10 CFR Part 20, Subpart H, “Respiratory Protection and Controls To Restrict Internal Exposure in Restricted Areas,” and the acceptance criteria in NUREG-1718, Section 9.2.4.10 (NRC, 2000).

9.3.12 Instrumentation

The MFFF utilizes multiple types of instrumentation to monitor radiation exposures. These include survey equipment, air samplers and CAMs, personal dosimetry, and bioassay equipment. The RP function oversees the selection and maintenance of radiological measurement instrumentation. It ensures the selection of instruments that will be operable and capable of measuring, at or below the required level, the types of radiation that could be encountered.

The RP function maintains a calibration program in accordance with the guidelines in ANSI/ANS Standard N323, “Radiation Protection Instrumentation Test and Calibration” (ANSI/ANS, 1978). Instruments are calibrated regularly in accordance with manufacturers’ specifications and are tagged to indicate the date the calibration expires. Internal procedures require personnel to notify the RP function of any instrument that appears to be out of calibration. Calibration sources are traceable to the National Institute of Standards and Technology.

The RP function has multiple areas throughout the facility that are reserved to store equipment and conduct RP tasks. The RP function has assigned laboratories and storage rooms for evaluating surveys, conducting isotopic analysis, storing equipment, and keeping records. Real-time monitoring of the facility is maintained at these locations through visual displays of alarms and readouts of radiation sensors throughout the facility. Near the egress point for the radiological control area is a room designated for decontamination, which contains monitors, shower and sinks, and first aid equipment. In addition, a technical support building contains the respiratory protection equipment, clean PPE, and a locker room storage area.

The instrumentation program contains a wide variety of sensors and equipment, which are maintained by the RP function in accordance with written procedures and applicable standards. The MFFF reserves space and resources for the RP function to conduct its responsibilities. The NRC staff finds that the applicant’s instrumentation program provides reasonable assurance of compliance with 10 CFR § 20.1501, “General,” and the acceptance criteria in NUREG-1718, Section 9.2.4.11 (NRC, 2000).

9.3.13 Additional Program Commitments

MOX Services has a records program for tracking key aspects of the RP program such as, radiation surveys, results of corrective action program referrals, RWPs, and planned special exposures. These records will be maintained in accordance with the facility’s document control program, described in Section 15 of the LA (MOX, 2010a).

The facility’s corrective action program is implemented when an individual’s exposure or uptake exceeds the administrative limits. An internal report is initiated in accordance with procedures to ensure that the cause of the exposure is identified and corrective actions are properly implemented. Also, when a CAM setpoint is exceeded, the alarms are recorded remotely so that the process can be terminated and corrective actions implemented. Contamination surveys, investigations, corrective actions, and reviews (along with deficiencies) are documented. The RP organization reviews this documentation for possible trends and to ensure proper implementation of the corrective actions.

The applicant will report to the NRC any event that results in an occupational exposure to radiation exceeding the dose limits in 10 CFR Part 20, within the time specified in 10 CFR § 20.2202, 10 CFR § 30.50, 10 CFR § 40.60, and 10 CFR § 70.74. The applicant will prepare and submit to the NRC an annual report of the results of individual monitoring, as required by 10 CFR § 20.2206(b). The applicant will refer to the facility's corrective action program any radiation incident that results in an occupational exposure that exceeds the dose limits in 10 CFR Part 20, Appendix B, or that is required to be reported by 10 CFR § 30.50, 10 CFR § 40.60, and 10 CFR § 70.74, and will report to the NRC both the corrective actions taken (or planned) to protect against a recurrence and the proposed schedule to achieve compliance.

The NRC staff finds that the applicant's records retention and reporting program provides reasonable assurance of compliance with 10 CFR Part 20, Subpart L, "Records," and Subpart M, "Reports," and the acceptance criteria in NUREG-1718, Section 9.2.4.12 (NRC, 2000).

9.4 Evaluation Findings

The staff reviewed the LA for the MFFF to possess and use SNM in accordance with Section 9 of NUREG-1718 (NRC, 2000).

The applicant's RP program includes the following:

- an effective, documented program to ensure that occupational radiological exposures are ALARA
- an organization with adequate qualification requirements for the radiation safety personnel
- approved written radiation protection procedures or RWPs for radiation protection activities
- radiation safety training for all personnel who have access to restricted areas
- requirements for an air sampling program
- control of radiological contamination within the facility
- a respiratory protection program
- requirements for radiological measurement instrumentation
- a program for monitoring the external and internal radiation exposure of personnel

Conformance to this program should ensure safe operation and provide early detection of unfavorable trends to allow prompt corrective action.

The NRC staff concludes, with reasonable assurance, that the applicant's RP program is adequate and that the applicant has the necessary technical staff to administer an effective RP program that meets the requirements of 10 CFR Parts 19, 20, and 70 for a license to possess and use SNM.

9.5 Exemption Request for Radiation Labeling

The applicant requested an exemption from the labeling requirements of 10 CFR § 20.1904(a) (DCS, 2006).

The regulation in 10 CFR 20.1904(a) requires that each individual container of licensed material bear a label indicating that it contains radioactive material and identifying certain specific information about the contents to enable individuals handling or using the containers, or working in their vicinity, to take precautions to avoid or minimize exposures.

In certain circumstances, it would be impractical for the applicant to mark individual containers to meet the labeling requirements (e.g., pellet boats in the sintering furnace). In lieu of labeling each container, the applicant stated that it would post, in restricted areas that house or store radioactive material, signs that incorporate the radiation symbol with the warning "Caution Radioactive Material: Any Container in This Area May Contain Radioactive Material." The applicant would also post signs at each entrance to a restricted area in which radioactive materials are used or stored.

The posting of areas with containers of radioactive material, coupled with plant personnel who are appropriately trained in radiation protection requirements, provides ample protection to personnel working at the MFFF and should result in no undue hazard to life or property. The NRC has granted similar exemptions to other fuel cycle facilities.

The regulations in 10 CFR § 20.2301, "Applications for exemptions," authorizes the NRC to grant exemptions from the requirements of 10 CFR Part 20. The requested exemption is authorized by law and will not result in undue hazard to life or property. Because of the impracticality of labeling each container at the MFFF and because of the appropriate training of personnel in radiation protection requirements, the staff agrees to the applicant's proposal in lieu of labeling each container. If granted to the applicant after completion of other regulatory requirements in 10 CFR Part 70, the license to possess and use radioactive material will include this exemption.

REFERENCES

(ANSI/ANS, 1978) American National Standards Institute/American Nuclear Society, Standard N323, "Radiation Protection Instrumentation Test and Calibration," 1978.

(ANSI/HPS, 1996) American National Standards Institute/Health Physics Society, Standard N13.22, "Performance Criteria for Radiobioassay," 1996.

(ANSI/HPS, 1995) American National Standards Institute/Health Physics Society, Standard N13.22, "Bioassay Programs for Uranium," 1995.

(DCS, 2006) Duke Cogema Stone & Webster, "Request for Exemption from Radiation Labeling Requirements," Letter from David Stinson (DCS) to Document Control Desk, Aiken, SC, September 27, 2006.

(MOX, 2010a) Shaw AREVA MOX Services, “Mixed Oxide Fuel Fabrication Facility—License Application,” Aiken, SC, March 2010.

(MOX, 2010b) Shaw AREVA MOX Services (MOX Services). “Mixed Oxide Fuel Fabrication Facility Integrated Safety Analysis Summary,” March 2010.

(NRC, 2005) U.S. Nuclear Regulatory Commission, Regulatory Guide 8.7, Revision 2, “Instructions for Recording and Reporting Occupational Radiation Dose Data,” Washington, DC, November 2005.

(NRC, 2000) U.S. Nuclear Regulatory Commission, NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility,” Washington, DC, August 2000.

(NRC, 1992) U.S. Nuclear Regulatory Commission, Regulatory Guide 8.34, “Monitoring Criteria and Methods To Calculate Occupational Radiation Doses,” Washington, DC, July 1992.

(NRC, 1978) U.S. Nuclear Regulatory Commission, Regulatory Guide 8.8, Revision 3, “Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable,” Washington, DC, June 1978

10 CFR Part 19, Notices, Instructions, and Reports to Workers: Inspection and Investigation

10 CFR Part 20, Standards for Protection Against Radiation
,
10 CFR Part 30, Rules of General Applicability to Domestic Licensing of Byproduct Material

10 CFR Part 40, Domestic Licensing of Source Material

10 CFR Part 70, Domestic Licensing of Special Nuclear Material

10.0 ENVIRONMENTAL PROTECTION

10.1 Regulatory Requirements

This chapter of the safety evaluation report (SER) contains the U.S. Nuclear Regulatory Commission (NRC) staff's review of environmental protection measures described by the applicant in Chapter 10 of its March 2010 revision of its mixed oxide (MOX) fuel fabrication facility (MFFF) license application (LA) (MOX, 2010a) to possess and use radioactive material. As noted in Chapter 1 of this SER, the MFFF is located in the F-Area of the U.S. Department of Energy's (DOE's) Savannah River Site (SRS). The staff evaluated the information provided by the applicant for environmental protection by reviewing Chapter 10 and other sections of the LA and supplementary information provided by the applicant. In some cases, the staff also performed independent calculations.

To be considered acceptable, the applicant must satisfy the following regulatory requirements regarding environmental protection:

- Title 10 of the *Code of Federal Regulations* (10 CFR) Section 20.1101(b), "Radiation protection program,, states that "the licensee shall use, to the extent practical, procedures and engineering controls based on sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as reasonably achievable (ALARA).

- 10 CFR § 20.1301(b), "Dose limits for individual members of the public." states that, if the licensee permits members of the public to have access to controlled areas, the limits for members of the public continue to apply to those individuals.

- 10 CFR § 20.1302(c), "Compliance with dose limits for individual members of the public," states that, upon approval from the Commission, the licensee may adjust the effluent concentration values in Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR Part 20, "Standards for Protection Against Radiation," Table 2, for members of the public, to take into account the actual physical and chemical characteristics of the effluents (e.g., aerosol size and distribution, solubility, density, radioactive decay equilibrium, chemical form).

- 10 CFR § 70.61(b)(2), "Performance requirements" states, in part, that the risk of credible high-consequence events must be limited by engineered or administrative controls or both. Under that section, high-consequence events are those internally or externally initiated events that result in an acute dose of 0.25 sieverts (Sv) (25 rem) or greater total effective dose equivalent (TEDE) to any individual located outside the controlled area. The controls shall be applied to reduce the likelihood of occurrence of the event, so that the event is highly unlikely or its consequences are less severe than the acute TEDE stated above.

- 10 CFR § 70.61(c)(2) states, in part, that the risk of credible intermediate-consequence events must be limited by engineered or administrative controls or both. Under this regulation, intermediate-consequence events are those internally or externally generated events that result in an acute TEDE of 0.05 Sv (5 rem) or greater to any individual outside the controlled area. The controls shall be applied to reduce the likelihood of occurrence of the event, so that the event is unlikely or its consequences are less severe than the acute TEDE stated above.

- 10 CFR § 70.61(c)(3) also states, in part, that the risk of credible intermediate-consequence events must be limited by engineered or administrative controls or both. Under this regulation, intermediate-consequence events are those internally or externally generated events that result in a 24-hour averaged release of radioactive material outside the restricted area in concentrations exceeding 5,000 times the values in Table 2 of Appendix B to 10 CFR Part 20. The controls shall be applied to reduce the likelihood of occurrence of the event, so that the event is unlikely or its consequences are less severe than the concentration values stated above.

10.2 Regulatory Acceptance Criteria

Section 10.4 of NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility" (NRC, 2000), provides the acceptance criteria for the NRC's review of the applicant's environmental protection program and was used as guidance for the staff's review.

10.3 Staff Review and Analysis

In its LA, the applicant described its commitment to environmental protection in three areas: (1) radiation safety goals (ALARA) for effluent control and waste minimization, (2) design of effluent and environmental monitoring for normal and off-normal operations, and (3) environmental surveillances to monitor the environmental impact from operations during normal and off-normal operations. The staff evaluated information provided by the applicant on ALARA goals and effluent, environmental monitoring programs, and environmental surveillances.

10.3.1 Radiation Safety

The staff evaluated the applicant's radiation safety measures for environmental protection, including the applicant's goals and controls to maintain public doses ALARA, in accordance with 10 CFR § 20.1101, "Radiation protection programs," as well as the applicant's practices to minimize not only the contamination of the facility and the environment but also the generation of radioactive waste. As noted in Chapter 9 of this SER, the goals of the ALARA program include maintaining occupational exposures, as well as environmental releases, as far below regulatory limits as is reasonably achievable. This is to ensure the health and safety of workers and the public located outside the restricted area boundary (RAB) and to protect the environment.

10.3.1.1 ALARA Design Goals for Effluent Control

Gas Effluent ALARA Goals

The applicant defined ALARA design goals for effluent control in Section 10.1.1 of its LA. The first goal is for airborne radioactive effluents released from the MFFF. This goal is not to exceed 20 percent of the effluent concentration limits in 10 CFR Part 20, Appendix B, Table 2, Column 1, as determined at the MFFF RAB. This fraction is consistent with staff expectations that an initial goal of 10 to 20 percent or less of the values in Appendix B to 10 CFR Part 20 can be achieved by almost all materials facility licensees, as stated in Regulatory Guide 8.37, “ALARA Levels for Effluents from Materials Facilities,” (NRC, 1993).

The applicant has also committed to a dose limit for an individual member of the public in the unrestricted area likely to receive the highest dose from the facility. This goal is 0.01 millisieverts (1 millirem (mrem)) per year TEDE, which is well below the 10-mrem constraint on air emissions specified in 10 CFR § 20.1101(d). This fraction is consistent with staff expectations for an initial goal of 10 to 20 percent of the 10 CFR Part 20 constraint described in NUREG-1718, Section 10.4.3, and, therefore, is acceptable to the staff.

Liquid Effluent ALARA Goals

The applicant has not defined liquid effluent ALARA goals, because the MFFF will not discharge liquid effluent directly to the environment. This is acceptable because the applicant’s proposal is to transfer low-level waste containing NRC-licensed material from the MFFF to DOE at the SRS in a manner consistent with the SRS waste acceptance criteria (WAC). DOE will take possession of the liquid waste before it reaches the RAB and is responsible for moving it safely. DOE will perform additional treatment before discharging this material. Therefore, DOE would manage any discharges of liquid effluent and would subject them to its ALARA considerations.

10.3.1.2 Air Effluent Controls to Maintain Public Doses ALARA

The heating, ventilation, and air conditioning (HVAC) system and the off-gas treatment ventilation system remove radionuclides and hazardous materials and thus control airborne emissions. Airborne waste from MFFF processes is routed through the HVAC system, which is designed to handle the expected volume of potentially radioactive waste, compartmentalize airborne waste, provide safe shutdown, and achieve an acceptable decontamination factor for each radionuclide. Several design features of the HVAC system, which include items relied on for safety (IROFS), provide confinement of radioactive materials. Ventilation exhaust is passed through multiple banks of filters, including high-efficiency particulate air filters. Airborne emissions are monitored and controlled to maintain doses outside the RAB ALARA.

The applicant’s design bases for these systems rely for guidance on NRC Regulatory Guide 3.12, “General Design Guide for Ventilation Systems of Plutonium Processing and Fuel Fabrication Plants,” issued 1973 (NRC, 1973b), and the American Society of Heating, Refrigerating and Air-Conditioning Engineers document, “Design Guide for Department of Energy Nuclear Facilities,” issued 1993 (DOE, 1993).

The staff concludes that the applicant’s commitment to regulatory guides and standards, together with the process controls and procedures that augment engineered controls as part of its ALARA program, ensures that engineered effluent controls will meet the regulatory requirements for capacity, compartmentalization, safe shutdown, and efficiency required during normal and likely facility conditions to maintain public doses ALARA; therefore, it is consistent with the acceptance criteria in Section 10.4 of NUREG-1718 and is acceptable to the staff.

10.3.1.3 Liquid Effluent Controls to Maintain Public Doses ALARA

As noted in Section 10.1.2.2 of the applicant's LA, the MFFF would not have liquid effluents that discharge directly to the environment. Separate systems that have do not interconnect collect and manage liquid radioactive and nonradioactive wastes. Radioactive process fluids are transferred using gravity flow, air jets, and steam jets, where practical. Drains within the radiation control area are routed to the liquid waste system. Liquid radioactive wastes are collected in the aqueous liquid waste system or in the solvent liquid waste system and will be transferred to DOE facilities on the SRS in a manner consistent with the SRS WAC for appropriate storage and disposition by DOE. The staff concludes with reasonable assurance that the applicant's procedures and controls will maintain public doses ALARA, and, therefore, are consistent with the acceptance criteria in Section 10.4 of NUREG-1718 and are acceptable to the staff.

10.3.1.4 ALARA Review and Reports to Management

Sections 9.2.1 and 10.1.3 of the applicant's LA describe the ALARA program and management involvement. MOX Services management receives reports summarizing the ALARA program, including trending information, so that it can compare the analytical results to ALARA goals. The applicant has committed to a program of measuring trends in environmental monitoring and surveillance data against the effluent ALARA goals on a quarterly basis. Abnormal increases in the trend of analytical results are reported to MOX Services management as soon as practical. ALARA goals are evaluated annually, and new goals are established for the following year, as appropriate. In addition, recommendations are made to MOX Services senior management, as needed, for changes in facilities and procedures to achieve ALARA goals. The staff concludes that the applicant's review and reporting program is likely to maintain ALARA goals and, therefore, is consistent with the acceptance criteria in Section 10.4 of NUREG-1718 and is acceptable to the staff.

10.3.1.5 Waste Minimization

Waste minimization reduces worker and public exposure to radiation and to radioactive and hazardous materials. The applicant has provided an overview of its commitment to waste minimization practices in Sections 9.1.2.3.3 and 10.1.4 of its LA. The applicant's proposal for incorporating waste minimization practices into the design process focuses on recycling and reuse of materials, as well as minimizing the introduction of materials that can become contaminated. During operations, the applicant proposes to rely on waste management procedures to separate and segregate solid and liquid wastes and remove packaging and shipping materials before they enter contaminated areas.

The applicant will use active and passive confinement systems and vacuum systems inside gloveboxes. These systems are designed to allow recycling of materials from the secondary waste streams in the aqueous polishing (AP) process and MOX process scraps back to the main processes. Specific AP process waste minimization steps include acid recovery, silver recovery, and solvent regeneration. Liquid waste is minimized in the AP process by use of recycling to the extent practical. For example, nitric acid is recovered by evaporation from the process and is partly reused as a reagent feedstock for the plutonium dissolution process.

Waste minimization documentation includes a statement of senior management support and identification of management, employees, and organizational responsibilities for waste minimization. Waste minimization goals, which are reevaluated annually, will be established based on operational data. Management is informed quarterly of the trends measured against waste minimization goals. New goals are established for the upcoming year, as appropriate. Recommendations are made to MOX Services senior management, as needed, for changes in facilities and procedures to achieve waste minimization goals. The staff concludes that the applicant's waste minimization program is likely to maintain ALARA goals and, therefore, is consistent with the acceptance criteria in Section 10.4 of NUREG-1718 and is acceptable to the staff.

10.3.2 Effluent and Environmental Monitoring

10.3.2.1 Concentrations of Radionuclides in Air Effluents and Public Doses

In its environmental report, the applicant provided an estimate of maximum radionuclide concentrations in the controlled area based on annual releases, a 50-percent atmospheric dispersion parameter value (X/Q) of 2.5×10^{-4} seconds per cubic meter, a distance to a receptor from the plant stack of 52 meters (171 feet) , and the assumption that releases occur from ground level. This calculation demonstrates that the average concentration in the controlled area immediately outside the restricted area would be less than 40 percent of its ALARA goal. The staff performed an independent calculation using the methodology described in Report 123, "Screening Models for Releases of Radionuclides to Atmosphere, Surface Water and Ground, Recommendations of the National Council on Radiation Protection and Measurements of the National Council on Radiation Protection and Measurements," dated January 22, 1996 (NCR, 1996), as described in NUREG-1718. In its calculation, the staff assumed a 28-meter (92 feet) stack height, no plume rise, a site-specific 3.6 meter per second annual average windspeed, and a wind direction toward an individual member of the public 100 percent of the time. The staff's estimate of the X/Q is 5×10^{-5} seconds per cubic meter at a distance of 400 meters (1312 feet). The staff's calculation demonstrates that the concentration in the controlled area would be less than 10 percent of the applicant's ALARA goal.

The applicant's estimate of the maximum potential dose to an individual member of the public in the unrestricted area is 4.1×10^{-6} millesieverts (4.1×10^{-4} mrem) per year. The staff performed independent analyses using GENII, the Hanford Environmental Radiation Dosimetry software system. The staff's result is 2.5×10^{-6} millisieverts (2.5×10^{-4} mrem) per year, which closely agrees with the applicant's value, well below the design ALARA goal. Both the applicant's and the staff's dose estimates to the public are less than a .01 microsieverts (1 microrem) per year.

Based on the staff's independent calculation, the known or expected concentrations of radioactive material in airborne effluents from the MFFF would be well below the limits in 10 CFR Part 20, Appendix B, Table 2, and, therefore, are consistent with the acceptance criteria in Section 10.4 of NUREG-1718 and are acceptable to the staff.

10.3.2.2 Physical and Chemical Characteristics of Radionuclides in Discharges

With regard to the provisions of 10 CFR § 20.1302(c), the applicant does not propose to adjust the effluent concentration values that appear in 10 CFR Part 20, Appendix B, Table 2, for members of the public, by taking into account the actual physical and chemical characteristics of the effluents (e.g., aerosol size distribution, solubility, density, radioactive decay equilibrium, chemical form). This is because the applicant demonstrated compliance with the annual dose limit of 10 CFR § 20.1301, "Dose Limits for Individual Members of the Public," by using the dose methodology in 10 CFR § 20.1302(b)(1), and not by using the concentration-based methodology in 10 CFR § 20.1302(b)(2). The applicant's approach is consistent with the requirements of 10 CFR Part 20 and, therefore, is acceptable to the staff.

10.3.2.3 Air Effluent Discharge Location and Effluent Monitoring

In Section 10.2.1 of its LA, the applicant identified the facility stack located on the roof of the MOX process building as the discharge location for radioactive air effluents from the MFFF. This stack is 28 meters (92 feet) tall and would discharge up to approximately 5,720 Cubic meters (202,000) cubic feet per minute of air during normal operations. The applicant has committed in the LA to the use of two redundant continuous air monitors and two fixed samplers of airborne particulate matter to monitor MFFF air effluent. The applicant has also committed in the LA to separately quantify the contributions from the AP and MOX processes, using two additional continuous air monitors, before the two streams are commingled and discharged from the single stack. The applicant will also sample air effluent contributions from areas not used for processing special nuclear material.

Based on Regulatory Guide 4.16, "Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Production Plants," Revision 1, issued December 1985 (NRC, 1985), particulate effluent from the stack would be collected continuously during operations to determine quantities and average concentrations of principal radionuclides that would be released. Table 10.2-2 of the applicant's LA identifies the analytical methodologies used to characterize airborne emissions (e.g., liquid scintillation, alpha spectrometer).

The staff finds that these commitments are consistent with the regulatory requirements for 10 CFR § 20.1302(a) and the staff's review guidance in Section 10.4 of NUREG-1718 and, therefore, are acceptable to the staff.

10.3.2.4 Liquid Effluent Discharge Location and Effluent Monitoring

The MFFF does not discharge radioactive liquid to the environment during normal or off-normal operations. The aqueous or solvent liquid waste systems collect the liquid radioactive waste and transfer it to the DOE SRS for disposition. Tanks used for storage of radioactive material are located inside the MFFF buildings and are equipped with drip pans and leak detection.

10.3.2.5 Environmental Monitoring Program

Preoperational and operational environmental monitoring activities determine baseline values and assess the environmental impact of licensed activities of the MFFF. As noted above, the MFFF would not discharge radioactive nuclides directly to the aquatic environment. Thus, environmental surveillances will focus on potential airborne radiological releases.

<u>Preoperational Environmental Monitoring Program</u>

The MFFF preoperational monitoring program, which begins 2 years before facility operations start, is based on the data collected over several years at the SRS and additional data collected by the applicant. The objectives of the environmental preoperational monitoring program are to establish a baseline of existing radiological and biological conditions at or near the MFFF site; evaluate procedures, equipment, and techniques used in the collection and analysis of environmental data; and train personnel in their use.

The applicant will take direct radiation measurements and samples of air, soil, and vegetation with analyses for uranium and plutonium and other radionuclides of interest. These activities will establish a baseline for isotopic composition and concentrations that will then be compared to results from operational environmental surveillances. The applicant's LA Table 10.3-1 identifies preoperational airborne monitoring locations, frequency of sampling, collection methodology, and radionuclide analyses.

The applicant's LA Table 10.3-2 contains an analysis of the lower limits of detection for various radionuclides. Sufficient volumes of samples (e.g., rainwater) are to be collected to ensure the attainment of lower limit of detection thresholds in the analysis.

Preoperational terrestrial sampling and analysis will provide a comprehensive baseline of radiological conditions related to the deposition of airborne emissions in the environs (including water bodies and sediment) of the MFFF.

Operational Environmental Monitoring Program

The applicant's operational monitoring program will be similar to its preoperational monitoring program. However, locations and sampling frequency for air, water, and terrestrial sampling and analysis may be altered, based on results from preoperational or operational emissions monitoring.

To ensure that the regulatory limits for doses to the public found in 10 CFR § 20.1301 are not exceeded, MOX Services has established administrative limits and action levels, as shown in Table 10.3-9 of the LA. If an action level is exceeded for sampling, the applicant would investigate to determine the source of elevated activity. As noted above, emissions data are trended as an analytical tool. Based on the operating history and trending analyses of the facility and operating data, the applicant would adjust operational data and sampling and analysis programs, as necessary.

Quality Control

Analytical quality control, addressed by the applicant in Section 10.3.7 of its LA, is described in laboratory procedures and is consistent with Chapter 15 of the LA. Analytical procedures are consistent with national or international consensus standards, or their performance is equivalent or superior to such methods. Analytical instrumentation is standardized and calibrated in accordance with the manufacturers' recommendations. Calibration standards are traceable to the National Institute of Standards and Technology.

The applicant's preoperational and operational environmental surveillance programs are consistent with applicable regulatory criteria and the staff's review guidance in Section 10.4.3 of NUREG-1718 and, therefore, are acceptable to the staff.

10.3.2.6 Consequence Assessment Methodologies

In its safety assessment, the applicant calculated committed doses to individuals outside the controlled area (i.e., the public) and concentrations of radioactive material in the environment outside the restricted area from each postulated accident to demonstrate that risks from event consequences were reduced to acceptable levels. The consequence assessment methodology used by the applicant for dose consequences at the controlled area boundary is the same methodology used for the site worker, as described in Chapter 9.0 of this SER, with the exception of the value of the atmospheric dispersion factor. The atmospheric dispersion factor that the applicant derived for the distance from the MFFF to the controlled area boundary is 3.7×10^{-6} seconds per cubic meter. The staff confirmed this value using MACCS2 and site-specific meteorological data and found it acceptable.

The RAB is approximately 52 meters (171 feet) from the MFFF discharge stack. The atmospheric dispersion factor that the applicant derived for this location is 8.39×10^{-4} seconds per cubic meter. The applicant also derived an atmospheric dispersion factor for the secured warehouse, which contains stocks of depleted uranium. This value is 2.71×10^{-3} seconds per cubic meter, based on a distance from the warehouse to the RAB of approximately 28 meters.

As a result, the equation used to calculate environmental consequences is

$$[EC]_x = \{[\text{Source Term}/RF] \times [X/Q]^{RA} \times [f]_x\}/(3600 \text{ s hr}^{-1} \times 24 \text{ hr}),$$

where source term is the same as described in Chapter 9.0 of this SER, RF is the respirable fraction (which is divided back into the source term to negate the reduction applied for consequence source terms), the value for “f” is the specific activity and the fraction of the total quantity of the material at risk; that is, the radionuclide X, and $[X/Q]^{RA}$ is the value of the atmospheric dispersion factor for either the MFFF stack or the secured warehouse, as described above.

The use of this equation is consistent with the staff’s guidance in NUREG/CR-6410, “Nuclear Fuel Cycle Facility Accident Analysis Handbook,” issued March 1998 (NRC, 1998), and the regulations in 10 CFR § 70.61(c)(3) and, therefore, is acceptable to the staff.

10.3.3 Integrated Safety Analysis Summary

In its Integrated Safety Analysis (ISA) Summary (MOX, 2010b), the applicant identified various sequences for radiological and nonradiological accidents, which were then evaluated to ensure adequate protection of worker health and safety. Protecting the worker by ensuring that all credible high-consequence events within the controlled area are rendered highly unlikely and that all the intermediate-consequence events within the controlled are rendered unlikely would also ensure that the environmental performance requirements in 10 CFR § 70.61(c)(3) for the area outside the controlled area would be met. The staff determined that adverse environmental consequences could occur only if unmitigated intermediate- or high-consequence events were also present. Because all such events were mitigated, the staff did not identify any accident sequences that would fail to meet the environmental performance requirements.

In Chapter 5 of the LA, the applicant presents the mitigated bounding-event consequences for the five major categories of events: fire, explosion, loss of confinement, load-handling events, and criticality. In Chapter 5 of this SER, the staff evaluates the applicant’s ISA Summary and documents its conclusion that the ISA Summary is complete, provides reasonable estimates of the likelihood and consequences of each accident sequence, and provides sufficient information to determine whether the applicant identified adequate engineering or administrative controls for each accident sequence. In its review of Chapter 15 of the LA, the staff evaluated the management measures used to ensure that the IROFS would adequately perform their intended safety functions. The applicant mitigated each event by employing various IROFS, which can be in the form of active or passive engineered controls or administrative controls. After it employed the IROFS to mitigate the consequences of the bounding events, the applicant determined that the occurrence of each bounding event was highly unlikely or unlikely, as required. This, in turn, resulted in a determination, under 10 CFR 70.61(c)(3), that the environmental performance requirements would be met. Thus, there would be no significant adverse environmental impact beyond the controlled area from the bounding events identified above.

Based on the use of IROFS, the implementation of management measures, and quality assurance, the staff finds that the applicant's methodology of public consequence analysis and environmental consequence determination is acceptable.

10.4 Evaluation Findings

The NRC staff issued a final environmental impact statement in January 2005 for this licensing action, as required by 10 CFR § 51.20, "Criteria for and Identification of Licensing and Regulatory Actions Requiring Environmental Impact Statements." After weighing the environmental impacts of the proposed operation of the MFFF, the NRC staff recommended, in the final environmental impact statement, that, unless safety issues mandated otherwise, the proposed license be issued to MOX Services.

The applicant has developed a program to implement adequate environmental protection measures during operation. These measures include (1) environmental and effluent monitoring and (2) effluent controls to maintain doses to the public ALARA as part of the radiation protection program. The NRC staff concludes that the applicant's program, as described in its application and environmental report, is adequate to protect the environment and the health and safety of the public and complies with regulatory requirements imposed by the Commission in 10 CFR Part 20; "Standards for Protection Against Radiation", 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material"; 10 CFR Part 40, "Domestic Licensing of Source Material"; 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions"; and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material."

REFERENCES

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10 CFR Part 20; Standards for Protection Against Radiation.

10 CFR Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material”.

10 CFR Part 40, “Domestic Licensing of Source Material”.

10 CFR Part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions”.

10 CFR Part 70, “Domestic Licensing of Special Nuclear Material”.

11.0 PLANT SYSTEMS

11.1 Mixed Oxide Process Description

The mixed oxide (MOX) fuel fabrication process consists of four major steps: (1) powder master blend and final blend production, (2) pellet production, (3) fuel rod production, and (4) fuel assembly production. The first operation is the production of the powder master blend. Polished plutonium dioxide (PuO_2) is mixed with depleted uranium dioxide (DUO_2) and recycled powder/pellet material to produce an initial mixture that is approximately 20 percent plutonium. This mixture is subjected to micronization in a ball mill and mixed with additional DUO_2 and recycled material to produce a final blend with the required plutonium content (typically between 2 and 6 percent). This final blend is further homogenized to meet plutonium distribution requirements. During the final homogenizing steps, a lubricant and pore-former are added to control density. The final homogenized powder blend is pressed to form green pellets, which are then sintered to obtain the required ceramic qualities. The sintering step removes organic products dispersed in the pellets and the previously introduced pore-former. The sintered pellets are ground to a specified diameter in centerless grinding machines and sorted. Powder recovered from grinding and discarded pellets are recycled through a ball mill and reused in the powder processing.

Fuel rods are loaded to an adjusted pellet column length, pressurized with helium, welded, and then decontaminated. The decontaminated rods are removed from the gloveboxes and placed on racks for inspection and assembly. Fuel rods are inserted into the fuel assembly skeleton, and the fuel assembly construction is completed. Each fuel assembly is subjected to a final inspection before shipment in a U.S. Department of Energy (DOE) fresh fuel shipping cask.

11.1.1 Conduct of Review

This section of the safety evaluation report (SER) contains the U.S. Nuclear Regulatory Commission (NRC) staff's review of the applicant's (i.e., MOX Services) description of the safety of the MOX process (MP), which is contained in Section 11.1 of the license application (LA) (MOX, 2010a), with supporting process safety information from Chapters 5, 8, and 11 of the LA and Section 4.1 of the Integrated Safety Analysis (ISA) Summary (MOX, 2010b). The objective of this review is to determine whether the chemical process items relied on for safety (IROFS) and their design bases provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents. The staff coordinated its review of MP safety design bases and strategies with the review of the radiation and chemical safety aspects of accident sequences described in the safety assessment of the design bases (see Chapter 5 of this SER), the review of fire safety aspects (see Chapter 7 of this SER), and the review of plant systems (see Chapter 11 of this SER).

The staff evaluated the MP process and chemistry information in the LA and ISA Summary against Title 10 of the *Code of Federal Regulations* (10 CFR) 70.64, "Requirements for New Facilities or New Processes at Existing Facilities," which requires that the design of new facilities or new processes at existing facilities incorporate baseline design criteria (BDC) and defense-in-depth practices. With respect to chemical protection, 10 CFR 70.64(a)(5) requires that the MOX fuel fabrication facility (MFFF) design provide for adequate protection against chemical risks produced from licensed material, facility conditions which affect the safety of licensed material, and hazardous chemicals produced from licensed material. Related to chemical protection, 10 CFR 70.64(a)(3) requires that the facility design provide for adequate protection against fires and explosions, such as those that could be initiated by or involve chemicals at the facility.

The review of the LA and ISA Summary focused on the design basis of chemical process IROFS, their components, and other related information. For each IROFS, the staff reviewed information provided by the applicant for the safety function, system description, and safety analysis. The review also included other design-basis considerations, such as redundancy, independence, reliability, and quality. The staff used Chapters 7.0 and 8.0 of NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility,” issued August 2000 (NRC, 2000), as guidance in performing the review.

As stated in the memorandum of understanding (MOU) between the NRC and the Occupational Safety and Health Administration entitled, “Worker Protection at NRC-Licensed Facilities” (Volume 53, Number 210, of the *Federal Register* dated October 31, 1998, pages 43950–43951), the NRC oversees chemical safety issues related to (1) radiation risk produced by radioactive materials, (2) chemical risk produced by radioactive materials, and (3) plant conditions that affect the safety and safe handling of radioactive materials. These types of chemical safety issues represent an increased radiation risk to the workers. However, the NRC does not oversee facility conditions that result in an occupational risk but do not affect the safe use of licensed material. The NRC has codified the MOU provisions applicable to the MFFF in 10 CFR 70.64(a)(5).

The NRC staff reviewed the following areas of the LA and ISA Summary applicable to process safety:

- MP description
- hazardous chemicals and potential interactions affecting licensed materials
- MP accident sequences
- MP chemical accident consequences
- MP safety controls

The staff also reviewed, as necessary, additional documentation from the applicant, responses to requests for additional information, and the open literature to understand the process and safety requirements. The following sections present the staff’s detailed evaluation of the MP.

11.1.1.1 System Description of the MOX Process

This section provides a description and overview of the MP, which includes design, operational, and process flow information.

The MP area receives polished PuO_2 from the aqueous polishing (AP) process, DUO_2, and the required components for assembling light-water reactor MOX fuel assemblies. The process mixes the plutonium and uranium oxides to form MOX fuel pellets. The pellets are loaded into fuel rods, which are then assembled into MOX fuel assemblies for use in commercial reactors. The MP area is designed to process up to 87 metric tons of heavy metal (uranium and plutonium) annually.

The MFFF uses the Advanced Micronized Master blend (A-MIMAS) process for manufacturing MOX fuel assemblies. A-MIMAS represents the latest evolution of the successive MIMAS fabrication processes, adopted by BELGONUCLEAIRE and COGEMA, to produce MOX fuel pellets. A-MIMAS uses a two-step mixing process. In the first step, the PuO_2 powder is mixed with DUO_2 and recycled scrap powder (from pellets that were out of dimensional specifications) to form a primary blend (master blend) with a nominal PuO_2 content of 20 percent of the total mass. This mix is then micronized. In the second step, the primary blend is forced through a sieve and poured into a jar and mixed with DUO_2 and recycled scrap powder to obtain the final blend with the specified plutonium content. The maximum PuO_2 content in the final blend is nominally 6 percent of the total mass. The two-step mixing process is used to ensure a consistent product.

The MP consists of five process areas divided into individual process units or systems. This section of the SER also describes the associated waste process units and the canning and sample pneumatic transfer systems used to move materials between process units.

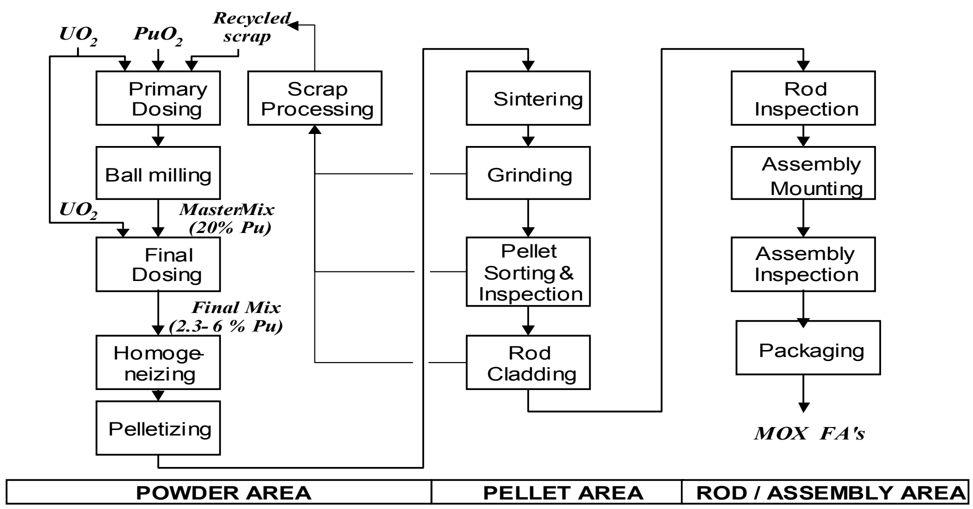

Figure 11.1-1 MP process overview

<u>Receiving Area</u>

This area includes truck unloading, PuO_2 container handling, counting, and storage before and after transfer to the AP line. The function of the receiving area is to receive, unload, and store PuO_2 and UO_2 powder. The receiving area comprises the following units:

- UO_2 receiving and storage unit
- UO_2 drum emptying unit
- PuO_2 receiving unit
- PuO_2 3013 storage unit
- PuO_2 buffer storage unit

<u>Powder Area</u>

This area has equipment for dosing and milling MOX powder at the specified plutonium content and properties, final blend homogenizing, and pelletizing. The powder area receives UO_2 and PuO_2 powders and produces a mixture of specific plutonium content suitable for the production of MOX fuel pellets. The powder area comprises the following units:

- PuO_2 can receiving and emptying unit
- primary dosing unit
- primary blend ball milling unit
- final dosing unit
- homogenization and pelletizing units
- scrap processing unit
- scrap ball milling unit
- powder auxiliary unit
- jar storage and handling unit
- additives preparation unit

Pellet Process Area

In this area, MOX pellets are sintered, ground, and sorted. The function of the pellet process area is to receive, store, process, and handle fuel pellets. The pellet process area comprises the following units:

- green pellet storage unit
- sintering units
- sintered pellet storage unit
- grinding units
- ground and sorted pellet storage unit
- pellet inspection and sorting units
- quality control and manual sorting unit
- scrap box loading unit
- pellet repackaging unit
- scrap pellet storage unit
- pellet handing unit

Fuel Rod Process Area

In this area, pellets are loaded into rods and the rods are inspected. The function of the fuel rod process area is to assemble, inspect, and store fuel rods. The fuel rod process area comprises the following units:

- rod cladding and decontamination unit
- rod tray loading unit
- rod storage unit
- rod tray handling unit
- helium leak test unit
- x-ray inspection unit
- rod scanning unit
- rod inspection and sorting unit
- rod decladding unit

Assembly Area

In this area, rods are loaded into assemblies and the assemblies are inspected and stored. The functions of the assembly area are to receive fuel rods and the required fuel assembly components and to assemble, inspect, and store completed MOX fuel assemblies. The assembly area comprises the following units:

- assembly mockup loading unit
- assembly mounting unit
- assembly dry cleaning unit
- assembly dimensional inspection unit
- assembly final inspection unit
- assembly handling and storage unit
- reserve pit unit
- assembly packaging unit

Waste Area

In this area, solid radioactive waste generated during the MP is processed, stored, and packaged for shipment. The waste area comprises the following units:

- filter dismantling unit
- maintenance and mechanical dismantling unit
- waste storage unit
- waste nuclear counting unit

Pneumatic Transfer Systems

The MP design includes pneumatic transfer systems to move materials between certain processing units and between process units and the laboratory. A pneumatic transfer system is also provided within the laboratory. The pneumatic transfer systems include the following:

- can pneumatic transfer system
- sample pneumatic transfer system

Section 11.1 of the LA (MOX, 2010a) and Section 4.1 of the ISA Summary (MOX, 2010b) provide a detailed description of the main units.

11.1.1.2 Staff Review of MOX Process Safety

11.1.1.2.1 Potential Depleted Uranium Dioxide Pyrophoricity and Burnback Concerns

The MP will blend the DUO_2 powder with the PuO_2 to form the matrix for the MOX fuel. Other nuclear fuel fabrication facilities handle UO_2 powder. The staff review noted a potential concern regarding the pyrophoric nature (sometimes referred to as burnback) of some fine UO_2 powders that can result in oxidation, damage to equipment (essentially a thermal oxidation and heating effect), and a potential release path resulting from the damage of confinement and filter systems (NRC, 1992). This is a known hazard; such rapid oxidations have occurred in NRC-licensed facilities. Those events involved burnback reactions that started in process equipment, causing localized damage, and then spread through the ventilation system. After those events, relatively large quantities of UO_2 powders were found on the damaged filters and equipment, including polycarbonate barriers and filters (prefilters and primary high-efficiency particulate air (HEPA) filters). The ventilation system carried the hot UO_2 particles to the filters, where a combination of the hot particles and continued oxidation reactions damaged the HEPA filters. The health consequences of those events were low because of rapid response by personnel and the low (relative to plutonium) radiological hazard of uranium.

The fuel fabrication process generates several oxides of uranium. According to information provided in NRC Information Notice 92-14, "Uranium Oxide Fires at Fuel Cycle Facilities," issued February 1992 (NRC, 1992), the final and most stable oxide in the process is UO_2. The oxidation reactions can be complex, with their rates, heat evolution, and final products dependent on several parameters, the most important of which are the fineness of the powder and the temperature. According to sources cited in NRC Information Notice 92-14 (NRC, 1992), normally stable UO_2 may be pyrophoric or oxidize rapidly even at room temperatures when in very fine powder form (i.e., when specific surface area is greater than 10 square meters per gram (m^2/g)). Coarser powders, which are more common, may require elevated temperatures (i.e., greater than 300 degrees Celsius) to oxidize.

The following equation characterizes the typical burnback (oxidation) reaction:

$$3UO_2 + O_2 \rightarrow U_3O_8 \qquad (11.1.1.2.1\text{-}1)$$

The staff notes that a number of laboratory studies of uranium ignition have been conducted under well-defined boundary conditions. These studies determined that the primary factor influencing uranium ignition was the specific surface area of the sample (Stakebake, 1994). Based on these studies, a specific surface area equal to 10 m^2/g (which is a condition for spontaneous burnback at ambient temperatures) represents a relatively small particle size. The applicant indicated that the DUO_2 feed to the MP is expected to have a cumulative particle size distribution that includes 95 percent of the particles smaller than 100 microns (μm) in diameter and the remaining 5 percent between 100 μm and 400 μm in diameter. The applicant further indicated that, after ball milling in the MP, the PuO_2/DUO_2 mix is expected to have a particle size distribution between 2 and 10 μm in diameter. As a result, and consistent with previous staff positions, the staff assumes that only the DUO_2, which has been ball milled which have the smaller particle size, are at risk for burnback (NRC, 2005).

The NRC information notice recognizes that, by the very nature of the fuel manufacturing process, unstable uranium powder must be handled, and it recommends that certain preventive measures be taken to reduce the potential for fires at licensed facilities. These preventive measures include the following:

- Limit the type of feed to stable powder whenever possible.
- Store unstable powder in closed metal containers.
- Replace the combustible components of powder transfer lines and equipment with components made of noncombustible materials, to the extent practicable.

- Require an operator to be present when a process is under way and improve visibility around vulnerable equipment.

- Incorporate fire safety of vulnerable equipment into the operator training program, including use of portable fire extinguishers.

- Implement a preventive maintenance program for vulnerable equipment. Periodic inspection may alert the operator to telltale signs of overheating.

The staff reviewed the applicant's documentation and found the following:

- The DUO_2 is delivered from offsite sources to the secured warehouse building by truck in palletized 55-gallon drums.

- [redacted]

- Most of the fuel fabrication process operations are conducted within an inert atmosphere to eliminate the adverse effects of atmospheric oxygen on the process or fuel. The nitrogen system provides a nitrogen atmosphere and ventilation within the gloveboxes and UO_2 receiving hoppers in the units that handle UO_2 powder.

- During normal operating conditions, most of the MP operations are fully automatic (although there are some manual operations) and all are supervised by an operator.

- Automatic control actions related to prevention of adverse incidents (e.g., accidental criticality, load drop, loss of confinement) are implemented through dual, redundant safety programmable logic controllers.

- The fire detection system monitors the units by means of smoke detectors, temperature detectors, or both that are located in the processing gloveboxes. Fire detectors are also provided external to the gloveboxes in the process rooms.

- A carbon dioxide fire suppression system provides fire suppression capability in the gloveboxes. An automatic, non-halogenated clean agent system protects the process rooms.

- Combustible materials are controlled based on a procedure.

The staff concludes that a potential pyrophoric reaction or burnback of UO_2 cannot be dismissed because such a reaction has occurred previously during fine UO_2 fuel powder processing. Its potential effects upon confinement, such as the entrainment of the potentially hot powder into the ventilation system, deposition on filters, and damage to the filters by the hot powder particles and continuing oxidation reactions, could potentially impact several units in the MP area that handle fine UO_2 powder by itself or blended with PuO_2. Such a burnback event could result in damage to the final HEPA filters and loss of confinement, the release of large quantities of uranium oxides (a chemical toxicity concern), the release of plutonium powders from the commingled blend, the initiation of other loss of confinement events, such as fires, or some combination of these effects.

However, the applicant indicated that gloveboxes will remain under normal supply ventilation (nitrogen for MP gloveboxes). The glovebox very high depressurization (VHD) exhaust HEPA filters, the common VHD intermediate filter (high-strength stainless steel roughing filter and HEPA filter), and the VHD final filter (roughing filter, high-strength stainless steel or glass fiber prefilter, and two HEPA filters) will capture soot generated during the initial stages of the fire.

The applicant analyzed the effects of elevated temperature on VHD and high-depressurization exhaust (HDE) final HEPA filters. The final filter assemblies for the VHD and HDE systems are provided with a roughing filter, a high-strength stainless steel or glass fiber prefilter, and two stages of HEPA filters. The VHD system is split into the main VHD exhaust and the laboratory VHD exhaust. Because significant mixing of the airflow before it enters the final filters results in exhaust gas dilution, the inlet temperature to the final HEPA filters of the main VHD exhaust system and the laboratory VHD exhaust system must remain within the HEPA filter temperature limit of 400 degrees Fahrenheit.

The staff finds the design-basis information associated with the ventilation system and final filtration units to be sufficient to address the potential burnback phenomena associated with the pyrophoric nature of some UO_2 powders. The applicant's proposed operating modes and controls are consistent with guidance provided in NRC Information Notice 92-14 (NRC, 1992).

11.1.1.2.2 Potential Plutonium Dioxide Heating Effects

The staff has reviewed the plutonium handling areas for potential chemical safety concerns. The staff's review during the construction authorization request stage noted concerns related to the potential heat generation by the PuO_2 (NRC, 2005).

The power output of reactor-produced ^{239}Pu metal is usually in the range of 2 to 10 watts per kilogram (W/kg) (DOE, 1994).

Section 4.2 of the ISA Summary (MOX, 2010b) provides information on isotopic distribution in the alternate feedstock (AFS) and pit disassembly and conversion facility/advanced recovery and integrated extraction system (PDCF/ARIES) feeds to the AP process.

[redacted]

The applicant identified the specific heat loads for plutonium as follows:

[redacted]

Using values from the literature (DOE, 1994), the staff estimates heat loads of [redacted], based on unpolished PuO_2 feed material, depending on the isotopic ranges used. Heat loads would be less for polished PuO_2 because the americium would have been removed. These values generally overlap the applicant's heat load estimates. Thus, the staff finds the applicant's values to be reasonable.

The HDE system provides ventilation to the plutonium storage areas to maintain the environmental conditions in the rooms at acceptable levels.

The HDE system consists of ductwork, dampers, fans, and filters that are required to exhaust air from the C3 and certain C2 confinement zones in both the AP and MP buildings. The C3 confinement zones consist primarily of process rooms containing gloveboxes. The C2 zones served by the HDE system are those that require reliable ventilation for heat removal and include the following:

- PuO_2 (3013 container) storage rooms
- trains A and B emergency electrical equipment rooms, which include variable frequency drives that provide electric power to the VHD, HDE, and process cells depressurization exhaust (POE) fans
- rooms that contain the VHD fans and final filter units
- HDE exhaust fan rooms
- POE exhaust fan rooms

Additional discussions of the confinement system can be found in Section 11.3 of this SER. The staff notes that these design bases and approaches are consistent with accepted practice for steels (e.g., American Society of Mechanical Engineers), concrete (e.g., American Concrete Institute), and most plastics (e.g., Perry, 1997) and finds this approach to be acceptable.

11.1.1.2.3 Potential Plutonium Dioxide Pyrophoricity and Burnback Concerns

According to a DOE handbook on pyrophoricity and spontaneous heating (DOE, 1994), large pieces of plutonium metal react slowly with the oxygen in air at room temperature to form plutonium oxides. The rate of oxidation depends on a number of factors, including (1) temperature, (2) surface area of the reacting metal, (3) oxygen concentration, (4) concentration of moisture and other vapors in the air, (5) the type and extent of alloying, and (6) the presence of a protective oxide layer on the metal surface. The rate of oxidation increases with increases in the first four factors and decreases with the last. Alloying can either increase or decrease the oxidation rate, depending on the alloying metal. Of all these factors, moisture has a large effect on the oxidation rate and is especially significant in evaluating conditions for storing plutonium metal and oxide.

Several plutonium oxides can be formed from oxidation of metal or decomposition of plutonium compounds. Oxide phases corresponding to sesquioxide (Pu_2O_3) and dioxide (PuO_2) compositions have been identified and are well characterized. Pu_2O_3 is pyrophoric in air and rapidly forms PuO_2 while releasing heat. The dioxide (PuO_2) is unreactive in air, but reportedly will heat slowly with water vapor at elevated temperatures. Because of its chemical stability and relative inertness, PuO_2 is the preferred form for shipping and storing plutonium at the present time (DOE, 1998).

Plutonium oxide powder from the oxalic precipitation, filtration, and oxidation unit is homogenized (mixed) and sampled in the homogenization, filling, and sampling (KCB) unit. After receiving the sample vial from the sample pneumatic transfer glovebox, the operator mixes the powder in a homogenizer. From the 27 grams of powder contained in the sampling vial, the operator prepares three different kinds of samples using a vibrating funnel. About 4 grams are used for moisture and plutonium content analysis. The 4 grams are divided into two samples in dishes. These samples are used for determining the moisture content and for thermogravimetric analysis, which is used to determine the plutonium content in the PuO_2 powder. About 11 grams of the powder is packaged into a number of specimens (usually about five) that are used for different analyses (e.g., plutonium isotopic composition, impurities determination) performed in the laboratory. About 13.5 grams of powder are kept in the sampling vial and manually transferred to the sample storage glovebox for use as a spare sample for further analyses or backcheck of the product specification. If the spare sample is used, it will be fractionated into specimens in the same way as a normal sampling vial and then transferred to the laboratory.

The staff evaluation concluded that PuO_2 is the most stable form of plutonium oxide, and it is not subject to pyrophoric (burnback) oxidation. The applicant has indicated that it will sample and analyze plutonium oxide powder produced in the AP process for purity and product specification in the KCB unit before transfer to the MP. As a result, the staff has reasonable assurance that plutonium pyrophoricity events will be highly unlikely in the MP area.

11.1.1.2.4 Sintering Furnace Concerns (EXP01 and EXP02)

<u>Hydrogen Explosions (EXP01)</u>

Sintering furnace hydrogen explosions (EXP01) are considered by the applicant in the pellets processing area where hydrogen explosions may occur in the high temperatures of the furnaces that sinter MOX fuel pellets. Section 8.1.2.4.1.1 of this SER discusses this event.

<u>Steam Explosions (EXP02)</u>

Steam explosions or overpressurizations (EXP02) are hypothesized to occur within the sintering furnace because of overfill of the humidifier system (i.e., the high level in the argon-hydrogen (Ar-H) humidifier results in water carryover) or within the humidifier itself because of the introduction of water onto a hot humidifier heater (i.e., bubbler heaters are energized before introduction of cold water), causing steam generation. [redacted]

The applicant's safety strategy for this event is preventive and involves restricting the flow of demineralized water to the sintering furnace and isolating the Ar-H supply upon detection of a high water level in the humidifier mixer drain tank.

[redacted]

The staff finds that the credited IROFS discussed above provide a diverse method to ensure that the flow of demineralized water to the sintering furnace is restricted and that the Ar-H supply is isolated upon detection of a high water level in the humidifier mixer drain tank. In addition, the applicant committed to have fail-safe and redundant active engineered IROFS [redacted] and use them in combination with passive engineered IROFS [redacted] and administrative IROFS [redacted]

The staff finds this an acceptable approach to comply with the single failure criterion. The single failure criterion, management measures (as described in Chapter 15 of this SER), quality assurance requirements (as described in the MOX Project Quality Assurance Plan (MPQAP)), and the use of codes and standards for engineered IROFS give the staff reasonable assurance that this high consequence event is highly unlikely. Therefore, the proposed safety strategy and

IROFS comply with the performance requirements of 10 CFR 70.61, "Performance Requirements."

In addition, the applicant will use features to reduce the challenge to IROFS where it is practical. For example, to minimize the ignition sources, the applicant will ground pipes, vessels, and gloveboxes. The staff finds that these features comply with the defense-in-depth requirements of 10 CFR 70.64(b).

11.1.1.2.5 Design-Basis and Applicable Baseline Design Criteria

Sections 11.1.1.2.1 through 11.1.1.2.4 of this SER discussed the design bases of the MP that are associated with chemical processing.

The applicant stated that it applied the BDC, as described in 10 CFR 70.64, from the outset of MFFF design work and that these criteria were primarily focused on the physical design and facility features, so as to achieve a conservatively designed facility tolerant of both process upsets and human errors (see Section 12.0 of the LA (MOX, 2010a)). The applicant stated that applicable chapters of the LA (MOX, 2010a) provided information demonstrating compliance with these criteria.

To ensure that all event sequences with consequences with medium or high consequences as stated in 10 CFR 70.61 (b) and (c) meet the performance requirements identified in 10 CFR 70.61, the applicant applied the following qualitative design criteria and commitments to those events and the associated IROFS:

- application of the single failure criteria or double contingency (for nuclear criticality)
- application of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and nuclear quality assurance (NQA-1)
- application of industry codes and standards
- management measures, including surveillance of IROFS (i.e., failure detection and repair or process shutdown capability)

For those credible events in which the single failure criteria or double contingency are not applicable (i.e., sole IROFS or passive IROFS feature), IROFS features are identified and the commitments for IROFS listed above are applied.

In terms of chemical protection, 10 CFR 70.64(a)(5) states the following:

> Chemical protection. The design must provide for adequate protection against chemical risks produced from licensed material, facility conditions which affect the safety of licensed material, and hazardous chemicals produced from licensed material.

The regulations in 10 CFR 70.64(a) require that the design of new facilities and processes address certain BDC. The following sections describe these BDC as they apply to the MFFF.

Quality Standards and Records

The applicant developed and implemented the MFFF design in accordance with the MFFF MPQAP. The MPQAP specifies the applicant's conformance to the quality assurance requirements found in Appendix B to 10 CFR Part 50, including quality standards and records and other programmatic management measures.

Natural Phenomena Hazards

The applicant evaluated the MFFF design for applicable natural phenomena hazards as part of the ISA and considered the design-basis natural phenomena hazards. The ISA evaluates the impact of these hazards on the MFFF design (including identification of IROFS).

Fire Protection

The MFFF design provides for fire protection features and systems. The ISA evaluates the fire and explosion hazards and their impact on facility operation (including identification of IROFS).

Environmental and Dynamic Effects

The MFFF design provides adequate protection from applicable environmental conditions and dynamic effects that could lead to a loss of safety function for IROFS. The ISA hazard evaluation includes the identification of applicable environmental considerations (e.g., corrosion) and dynamic effects (e.g., seismic performance) and considers their impacts on the ability of an IROFS to perform a required safety function.

Chemical Protection

The MFFF design provides protection from chemical risks from licensed material, facility conditions that affect the safety of licensed material, and hazardous chemicals produced from licensed material. The ISA evaluates the chemical and explosion hazards and their impact on facility operation (including identification of IROFS).

Emergency Capability

The MFFF design provides instrumentation and control (I&C) systems and other IROFS (e.g., heating, ventilation, and air-conditioning and confinement systems) which are credited for maintaining control of licensed material and hazardous chemicals produced from licensed material during emergency events. The individual event analyses of the ISA identify the relevant IROFS.

Onsite facilities (e.g., safe havens) are provided for employee egress during certain upset conditions. The safe havens are not relied on for safety to meet the performance requirements of 10 CFR 70.61.

Communication systems are provided for use by facility response personnel and to communicate with facility workers during upset conditions. These capabilities are not relied on for safety to meet the performance requirements of 10 CFR 70.61.

Fire-fighting capabilities are not relied on for safety to meet the performance requirements of 10 CFR 70.61.

Nuclear incident monitoring systems are provided to reduce risk to personnel and provide prompt warning and notification of an inadvertent criticality. These capabilities are not relied on for safety to meet the performance requirements of 10 CFR 70.61.

Utility Services

The MFFF design provides continued operation of necessary utility services. The emergency power system provides electrical power to those systems required to operate during loss of offsite electrical power events. The individual event analyses of the ISA identify those IROFS which must function during loss of offsite power events.

Inspection, Testing, and Maintenance

The applicant developed and implemented the MFFF design in accordance with the MFFF MPQAP. The MPQAP specifies the applicant's conformance to the quality assurance requirements found in Appendix B to 10 CFR Part 50, including the requirements for inspection, testing, and maintenance for IROFS to ensure their availability and reliability to perform their function when needed.

Criticality Control

The MFFF design provides protection from inadvertent criticality risks. Criticality safety includes adherence to the double contingency principle. The ISA evaluates criticality hazards and their prevention during facility operations (including identification of IROFS).

Instrumentation and Controls

The MFFF design provides I&C systems for both normal and emergency operations. The I&C system monitors and provides control capability of IROFS. The individual event analyses of the ISA identify IROFS, including I&C systems and components.

The MP facilities are broken down into process functional units, which are made up of one or more subassemblies performing consistent and elementary tasks. The applicant stated that the breakdown into control functional units allows each entity to operate relatively independently in the given mode. The staff notes that this separation and independence are consistent with accepted practices for safe operations.

With respect to chemical protection, the applicant will control process storage and operation conditions to prevent exothermic reactions in the MP area. Exothermic reactions of chemicals will be prevented through the control of the process parameters (e.g., reactant concentration, temperature, inert atmosphere) that affect the reactions. [redacted]

The applicant demonstrated that there is reasonable assurance that IROFS controls will be sufficiently reliable and available based on the use of standard nuclear industry engineering practices. The applicant has incorporated these practices into the facility general design philosophy, design bases, system design, and commitments to applicable management measures. These practices ensure that applicable industry codes and standards are utilized, adequate safety margins are provided, engineering features are utilized to the extent practicable, the defense-in-depth philosophy is incorporated into the design, and IROFS will be appropriately maintained.

The staff review finds that the applicant has provided sufficient information to meet the requirements of 10 CFR 70.64(a)(5). Using the guidance provided in NUREG-1718 (NRC, 2000), the staff concludes that the applicant has satisfied this BDC.

Related to chemical protection, 10 CFR 70.64(a)(3) includes the explosion protection BDC as part of the fire protection BDC as follows:

> Fire protection. The design must provide for adequate protection against fires and explosions.

The applicant stated that there is reasonable assurance that the IROFS will be sufficiently reliable and available based on the use of standard nuclear industry engineering practices. The applicant has incorporated these practices into the facility general design philosophy, design bases, system design, and commitments to applicable management measures. These practices ensure that applicable industry codes and standards are utilized, adequate safety margins are provided, engineering features are utilized to the extent practicable, the defense-in-depth philosophy is incorporated into the design, and IROFS will be appropriately maintained.

The staff review finds that the applicant has provided sufficient information to meet the requirements of 10 CFR 70.64(a)(3). Using the guidance provided in NUREG-1718 (NRC, 2000), the staff concludes that the applicant has satisfied this BDC.

11.1.2 Evaluation Findings

In Section 11.1 of the LA (MOX, 2010a) and Section 4.1 of the ISA Summary (MOX, 2010b), the applicant provided design-basis information for the MP and identified IROFS for the facility. Based on its review of the LA (MOX, 2010a) and ISA Summary (MOX, 2010b), as well as supporting information provided by the applicant relevant to the MP, the staff finds that, for the reasons discussed above, the applicant has met the BDC set forth in 10 CFR 70.64(a)(3) for explosions and 10 CFR 70.64(a)(5) for chemical safety.

The staff finds that the credited IROFS provide diverse methods to ensure that the IROFS are available and reliable to perform their safety functions when needed. Also, the applicant has committed to have fail-safe and redundant active engineered IROFS and to use them in combination with passive engineered IROFS and administrative IROFS.

The NRC staff finds that the applicant's approaches to the accident sequences described above are acceptable and comply with the single failure criterion. The single failure criterion, management measures (as described in Chapter 15 of this SER), quality assurance requirements (as described in the MPQAP), and the use of codes and standards for engineered IROFS give the NRC staff reasonable assurance that these high consequence events are highly unlikely. Therefore, the proposed safety strategies and IROFS comply with the performance requirements of 10 CFR 70.61.

The applicant will also use features to reduce the challenge to IROFS, to the extent practicable. For example, to minimize the ignition sources, the applicant will ground pipes, vessels, and gloveboxes. The NRC staff finds that these features comply with the defense-in-depth requirements of 10 CFR 70.64(b).

REFERENCES

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(MOX, 2010b) Shaw AREVA MOX Services, "MFFF—Integrated Safety Analysis Summary," Aiken, SC, March 2010.

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(Stakebake, 1994) Stakebake,J.L., "The Ignitability Potential of Uranium Roaster Oxide," 1994.

10 CFR 70, Domestic Licensing of Special Nuclear Material.

10 CFR 50, Domestic Licensing of Production and Utilization Facilities.

11.2 Aqueous Polishing Process and Chemistry

11.2.1 Conduct of Review

This section of the safety evaluation report (SER) contains the staff's review of aqueous polishing (AP) process described by the applicant in Section 11.2 of the license application (LA) (MOX, 2010a), with supporting process safety information from Chapters 5, 8, and 11 of the LA, Section 4.2 of the Integrated Safety Analysis (ISA) Summary (MOX, 2010b), and supplemental information provided by the applicant. The staff also reviewed technical literature, as necessary, to understand the process and safety requirements. The objective of this review is to determine whether the AP process safety items relied on for safety (IROFS) and their design bases provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents. The staff coordinated the review of AP safety design bases and strategies with the review of radiation and chemical safety aspects of accident sequences described in the safety assessment of the design bases (see Chapter 5 of this SER), the review of fire safety aspects (see Chapter 7 of this SER), and the review of plant systems (see Chapter 11 of this SER).

The staff evaluated the AP process and chemistry information in the LA and ISA Summary against the following regulations:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 70.64, "Requirements for New Facilities or New Processes at Existing Facilities," requires that baseline design criteria and defense-in-depth practices be incorporated into the design of new facilities or new processes at existing facilities.

- With respect to chemical protection,10 CFR 70.64(a)(5) requires that the Mixed Oxide Fuel Fabrication Facility (MFFF) design provide for adequate protection against chemical risks produced from licensed material, facility conditions that affect the safety of licensed material, and hazardous chemicals produced from licensed material.

- Related to chemical protection, 10 CFR 70.64(a)(3) requires that the facility design provide for adequate protection against fires and explosions, such as those that could be initiated by or involve chemicals at the facility.

The review of the LA and ISA Summary focused on the design basis of chemical process IROFS, their components, and other related information. For each IROFS, the staff reviewed information provided by the applicant for the safety function, system description, and safety analysis. The review also included other design-basis considerations such as redundancy, independence, reliability, and quality. The staff used NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility" (NRC 2000), as guidance in performing the review.

As stated in the Memorandum of Understanding between the U.S. Nuclear Regulatory Commission (NRC) and the Occupational Safety and Health Administration, "Worker Protection at NRC-Licensed Facilities" (Volume 53 of the *Federal Register*, p. 43950; October 31, 1998), the NRC oversees chemical safety issues related to (1) radiation risk produced by radioactive materials, (2) chemical risk produced by radioactive materials, and (3) plant conditions that affect the safety and safe handling of radioactive materials. These types of chemical safety issues represent an increased radiation risk to the workers. However, the NRC does not oversee facility conditions that result in an occupational risk but do not affect the safe use of licensed material. The NRC has codified the provisions in the memorandum applicable to the MFFF in 10 CFR 70.64(a)(5).

The NRC staff reviewed the following areas of the LA and ISA Summary applicable to process safety:

- AP description
- Potential interactions between hazardous chemicals affecting licensed materials
- AP accident sequences
- AP chemical accident consequences
- AP safety controls

The staff also reviewed additional documentation from the applicant, responses to requests for additional information, and the open literature, as necessary, to understand the process and safety requirements. The sections that follow present the staff's detailed evaluation of the AP process.

11.2.1.1 System Description of the Aqueous Polishing Process

This section provides a description and overview of the AP process flowsheet, including design, operational, and process flow information. Section 11.2.1.2 summarizes the major components and their functions.

The MFFF is designed to purify plutonium oxide (PuO_2) and then blend it with depleted uranium oxide (DUO_2) to produce completed mixed oxide (MOX) fuel assemblies for use in nuclear power reactors. The MFFF has two major process operations: (1) an AP process, which serves primarily to remove americium, gallium, and other impurities from the plutonium, and (2) the MOX fuel fabrication (MP) process, which processes the oxides into pellets and manufactures the MOX fuel assemblies. These processes are designed and integrated so that waste and discarded powder/pellet material streams are recycled to the extent practical. The AP process will receive weapons-grade PuO_2 from a planned pit disassembly and conversion facility (PDCF) and alternative feedstock (AFS) which are owned by the U.S. Department of Energy (DOE). The AP process has the following major steps:

[redacted]

[redacted]

[redacted]

[redacted]

11.2.1.2 Major Components and Functions

The AP process can be subdivided into the following four operational areas (unit symbols are provided in parentheses):

(1) The plutonium purification process includes the following units: decanning (KDA); milling (KDM); canning (KCC); recanning (KDR); dissolution (KDB); dechlorination and dissolution (KDD); purification cycle (KPA); oxalic precipitation, filtration, and oxidation (KCA); and homogenization (KCB).

(2) The recovery processes include the solvent recovery (KPB) unit, oxalic mother liquor recovery (KCD) unit, and acid recovery (KPC) unit.

(3) Waste storage includes the liquid waste reception (KWD) unit and waste organic solvent (KWS) unit.

(4) Off-gas treatment includes the off-gas treatment (KWG) unit.

11.2.1.2.1 Decanning Unit (KDA)

The decanning unit receives PDCF PuO_2 powder from the storage unit. It receives AFS powder from the vault storage area (DCM), as well. The powder received from both of these units is packaged in DOE Standard 3013 containers.

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

11.2.1.2.2 Milling Unit (KDM)

The milling (KDM) unit mills AFS PuO_2 powder, samples AFS powder, and feeds powder to the KDD electrolyzers.

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

11.2.1.2.3 Recanning Unit (KDR)

The recanning (KDR) unit receives reusable cans containing discarded AFS material from the KDM unit if, after milling and sampling, the material has been determined to be outside of the AP processing limits. Periodically, the KDR unit will also receive food cans from the KDA food can opening glovebox, whose contents exceed the AP salt percentage or mass limits. These food cans are manually introduced into the KDR convenience can packaging glovebox. The KDR unit repackages the powder from reusable cans into convenience cans and subsequently packages the convenience cans into approved containers; it also repackages the unopened food cans into approved containers. The containers are then stored in a vault storage area (DCM) for either transfer out of the MFFF or future treatment.

11.2.1.2.4 Dissolution Unit (KDB)

The dissolution (KDB) unit dissolves the plutonium oxide powder into a nitric acid solution for further processing in the KPA unit. The PuO_2 processed in this unit comes from either the PDCF or the AFS.

The first process step includes feeding nitric acid and silver nitrate into the electrolyzer along [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

11.2.1.2.5 Dechlorination and Dissolution (KDD)

The dechlorination and dissolution (KDD) unit removes the chlorine from chlorinated PuO_2 coming from the milling (KDM) unit. The KDD unit can process either chlorinated or nonchlorinated powder, but chlorinated powder can only be processed in the KDD unit.

[redacted]

11.2.1.2.6 Purification Cycle (KPA)

[redacted]

[redacted]

[redacted]

[redacted] [redacted]

[redacted]

[redacted] [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

11.2.1.2.7 Solvent Recovery Cycle (KPB)

The solvent recovery (KPB) unit continuously cleans and recycles solvent for reuse in the purification cycle (KPA) unit. Additionally, the KPB unit transfers alkaline waste to the high-alpha liquid waste (HAW) (KWD) unit and solvent waste to the waste solvent reception (KWS) unit. The major steps performed by the KPB unit include reception and cleaning of solvent, diluent washing of alkaline waste, solvent content adjustment and sampling, and solvent distribution.

[redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

11.2.1.2.8 Oxalic Precipitation and Oxidation Unit (KCA)

The KCA unit includes equipment installed in gloveboxes and various process cells in the BAP. The gloveboxes are the precipitation glovebox, the rotating filter glovebox, and the furnace glovebox. The major processing steps performed by the KCA unit are plutonium nitrate solution reception and feed preparation, oxalate precipitation, filtration, and drying and calcination.

[redacted]

[redacted]

[redacted] [redacted]

[redacted]

[redacted]

[redacted]

[redacted] [redacted]

[redacted]

[redacted]

[redacted]

11.2.1.2.9 Homogenization Area (KCB)

[redacted]

[redacted]

[redacted]

[redacted]

11.2.1.2.10 Canning Unit (KCC)

[redacted]

11.2.1.2.11 Oxalic Mother Liquor Recovery Unit (KCD)

The KCD unit receives feed from the KCA unit, transfers distillates to the acid recovery (KPC) unit, and transfers plutonium nitrate solutions to the purification cycle (KPA) unit. The KCD unit is designed to process oxalic mother liquor from the oxalic precipitation and oxidation (KCA) unit in order to remove oxalate ions and concentrate the plutonium nitrate solution before recycling it to the KPA unit.

[redacted]

<u>Concentration of Mother Liquors and Destruction of Oxalic Ions</u>

[redacted]

- [redacted]
- [redacted]
- [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

11.2.1.2.12 Acid Recovery Unit (KPC)

Process operations conducted within the KPC unit involve (1) the batch receipt of effluents from the purification cycle (KPA) unit, the oxalic mother liquor recovery process (KCD) unit, and the waste storage (KWD) unit; the dechlorination and dissolution (KDD) unit; and the batch or continuous receipt of effluents from the off-gas treatment (KWG) unit, (2) the concentration and transfer of the soluble effluent salts (i.e., impurities and radioactive activity) to the KWD unit, and (3) the recovery of nitric acid (13.6 N) and distillates (0.02 N nitric acid) for recycling. The recovered nitric acid is recycled to the AP process, while the distillates are recycled to the KPA and KWG units.

<u>Receipt of Effluents</u>

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

11.2.1.2.13 Off-Gas Treatment Unit (KWG)

The off-gas treatment (KWG) unit is designed to remove hazardous materials and trace amounts of plutonium from all off-gas from each AP unit. [redacted]

11.2.1.2.14 Liquid Waste Reception Unit (KWD)

The aqueous waste reception (KWD) unit receives and manages three types of waste: low-level liquid waste (LLW), americium-rich HAW, and stripped uranium liquid waste (SUW). This unit only processes solution that does not contain organic waste.

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

High-Alpha Liquid Waste

The HAW portion of the KWD system collects, stores, and transfers potentially contaminated radioactive aqueous wastes containing americium and other impurities extracted from the AP process plutonium streams. It also collects alkaline waste and treats any azides contained in those streams. The HAW consists of the following waste streams combined in the KWD unit:

[redacted]

[redacted]

[redacted]

[redacted]

Stripped Uranium Liquid Waste

This portion of the KWD system collects, stores, and transfers liquid waste generated by the stripping process in the KPA unit. [redacted]

11.2.1.2.15 Waste Organic Solvent Unit (KWS)

The waste organic solvent (KWS) unit collects waste organic solvent generated in various AP process units or systems for sampling and transfers it to the reagent processing building (BRP).

[redacted]

[redacted]

11.2.1.2.16 Sampling System

The automatic sampling unit (KPG) automatically collects liquid samples from AP process units for laboratory analysis.

[redacted]

[redacted]

11.2.1.2.17 Laboratory Waste Receipt Unit

The laboratory liquid waste receipt (LGF) unit receives and manages the liquid wastes generated by the MFFF laboratory for subsequent recycling in the KDB unit and for transfer to either the LLW or HAW KWD unit or KWS unit. [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

11.2.1.3 Staff Review of Aqueous Polishing Process Systems

The following section discusses the staff's technical review of AP systems with respect to accident events identified by the applicant in the ISAS. Each of these sections refers to additional discussions of ISA Summary events related to the AP.

11.2.1.3.1 Chloride Concentration of Alternative Feedstock Powder (KDD)

The staff evaluated the risk to the site worker and the individual outside the controlled area (IOC) from chlorine emissions from AP processes into the environment. AFS PuO_2 powder that contains greater than 500 parts per million (ppm) total chloride salts would be processed in the dechlorination (KDD) electrolyzers. The applicant expects the maximum content of chloride salts in a can of AFS feed to be 1 kilogram (kg). Chloride is the main impurity, from 1 to 25 percent of the total can weight. Under normal conditions, high-chloride AFS feed would be processed in KDD electrolyzers. The applicant's chemical consequence analysis (MOX, 2009d) indicated that the anticipated production of chlorine gas from the dechlorination of AFS feeds in KDD electrolyzers is 0–1kg/hour (h). The KDD unit was sized for a maximum chloride content of 15 percent because only a small number cans actually contain 25 percent (by weight) chloride. The applicant anticipates that as few as six cans in the AFS feed campaign will contain up to 25 percent (by weight) chloride. The dechlorination process will proceed batch by batch, [redacted] Each dissolution batch will have one can for high chloride AFS feed, and the dissolution of one batch is expected to last 12 hours. The applicant expects that most of the chlorine gas will be released in the first hour of the dissolution process. If all of the chlorine were released in 15 minutes during an accident, the release rate would be [redacted] If a can containing high chloride were inadvertently directed to a KDB (normal, nondechlorination) dissolution electrolyzer, the consequence calculations for the KDD unit would also bound the release from an inadvertent addition of chlorinated feed to the KDB unit. Thus, a total of up to 1 kg of chlorine gas could be liberated.

The staff has analyzed this event and has concluded that some chloride that enters the KDB electrolyzer would react with the silver nitrate already present in solution to form silver chloride that would subsequently precipitate in the electrolyzer. The applicant's analysis indicates that 1 kg is the maximum chloride content in a can to be added to a KDB electrolyzer. This mass is equal to 28.6 moles of chloride, and one mole of chloride will react with one mole of silver for each mole of silver available until all of the silver is consumed. The silver nitrate solution is expected to be 0.07 M in silver nitrate, and thus 0.07 M in silver ion (Ag^+). With a 51-L working volume in the electrolyzer, 0.07 M in Ag^+ implies that 3.57 moles of Ag^+ are available to react with any chloride inadvertently added to solution. Thus, 3.57 moles of Cl^- would be consumed, leaving approximately 25 moles of chloride available to be released as chlorine gas. This means approximately 125 g of chloride can be consumed, leaving approximately 875 g for release as chlorine gas.

[redacted]

[redacted]

[redacted]

[redacted]

The NRC staff finds that the applicant used a known and accepted model to calculate chlorine concentrations and finds the applicant's approach to be acceptable. Furthermore, the staff agrees with the applicant's conclusion that a chlorine release would be a low-consequence event because, as discussed in the paragraph above, the chlorine concentration that would be encountered outside the MFFF is less than the TEEL-1 value for chlorine gas.

11.2.1.3.2 Electrolyzers (KDB and KDD)

Electrolysis-related explosions are postulated to occur from hydrogen (H_2) that may be generated electrochemically at the cathode of the electrolyzer. The safety strategy for electrolysis-related explosions involves the application of IROFS to meet the performance criteria of 10 CFR 70.61, "Performance Requirements." Sections 8.1.2.4.1.3 and 11.2.1.3.3 of this SER describe the staff's evaluation of this hazard in more detail.

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

11.2.1.3.3 Hydrogen Production (KDB, KDD, KPA, KCA, KWG, and KWD)

Hydrogen Generated by Electrolysis within the Dissolution Process (KDB and KDD)

The catholyte well of each electrolyzer is initially fed with 13.6 N nitric acid from the catholyte storage tank associated with the given electrolyzer in the KDB or KDD unit. During electrolysis, the solution is continuously circulated between the catholyte storage tank and the catholyte well of the electrolyzer, as its acidity decreases as a result of electrolytic dissolution of plutonium.

As the normality of the catholyte decreases, hydrogen ion (H^+) will begin to be adsorbed on the cathode surface and react to form hydrogen gas (H_2), which is subsequently released from solution into the vapor space.

The fraction of hydrogen in the evolved gases from electrolysis is a function of the normality of the nitric acid and the current density of the tantalum cathode. The relevant limits for this event are the lower flammability limit (LFL) of hydrogen (4 percent H_2) and 25 percent of the LFL (1 percent H_2, the safety limit). The applicant indicated that 25 percent of the LFL is exceeded when the catholyte normality falls below a value of approximately 7.8 M.

[redacted]

Section 8.1.2.4.1.3 of this SER describes the applicant's proposed IROFS for this explosion event and the staff's evaluation of it.

Hydrogen Generated by Radiolysis (KPA, KCA, KWG, and KWD)

Radiolysis is the dissociation of molecules that can lead to gas generation. It occurs when organic and aqueous fluids are irradiated, in the case of the MFFF, by plutonium and americium. Since the organic and aqueous fluids are hydrogenous substances, the generated gas of concern is hydrogen. Hydrogen gas can build up in the vapor spaces of tanks and vessels. If an overpressurization occurs or if the concentration of the flammable gas exceeds the LFL, there is a risk for a radiolysis-induced explosion (EXP-03), which can result in the release of licensed material. There is also a risk of radiolysis in the waste handling system because of the confinement of radioactive material.

Section 8.1.2.4.1.2 of this SER discusses the applicant's safety strategy and proposed IROFS and the staff's evaluation for this event.

11.2.1.3.4 Titanium Reactions (KDB and KDD) and Discussion of Fire Event 5, Fire in the Titanium Electrolyzer in the Aqueous Polishing Glovebox Areas

The staff's evaluation found that the applicant's proposed AP process employs oxidation-reduction chemistry, based on the silver (I) to silver (II) couple, to facilitate the dissolution of PuO_2 in the electrolyzers. Silver (II) is corrosive, and special alloys are necessary for electrolyzer equipment that may be exposed to silver (II). The applicant stated that the anode compartment and cathode cover of the electrolysis pot will be constructed of titanium (MOX, 2010b). The applicant will destroy silver (II) by converting it back to silver (I) before electrolyzer solutions contact other equipment (fabricated out of 300-series stainless steels) in the process. The destruction of silver (II) will be accomplished by the addition of hydrogen peroxide, which reduces silver (II) back to silver (I).

The staff notes that the applicant has committed to employ American Society of Testing and Materials (ASTM) Grade 2 titanium alloy to provide adequate corrosion resistance in the presence of the harsh chemical conditions anticipated in the electrolyzer pots. However, titanium can be a reactive metal. Under certain conditions, titanium and its alloys are known to be capable of ignition and combustion. Furthermore, each electrolyzer is expected to operate at 30 volts (V) direct current (dc) (50 V maximum) and a current of 400-ampere dc (450 ampere maximum), which could potentially serve as an ignition source for the titanium. The NRC staff notes that laboratory tests have shown that sheets of commercially pure titanium and several of its alloys can be ignited in air using an oxyacetylene torch and heating the materials to their melting temperatures (approximately 3,000 degrees F) (DMIC, 1964). An electrical arc, as a result of welding or a breakdown of insulation, could cause localized heating of titanium, resulting in ignition. However, the NRC staff also recognizes that bulk titanium components are generally considered noncombustible and will only burn under extreme conditions. Experience has shown that, generally, in the absence of molten iron oxide, massive titanium (i.e., not finely divided powders or mill tailings) cannot be ignited in an ordinary air atmosphere without the application of an external source of heat or flame, and even with the application of heat or flame, metal temperatures approaching the melting point of the metal are necessary to achieve ignition (DMIC, 1964). Furthermore, the applicant indicated that, once ignited, titanium combustion will continue until the titanium is depleted, the air pressure falls below a critical value, or the ignition energy source (i.e., the electrical energy in this case) is removed (MOX, 2010b). The NRC staff notes that the main body and key components of the electrolysis pots of the electrolyzers will be fabricated from solid blocks of titanium alloy.

The applicant described its safety approach for titanium events in the ISA Summary (MOX, 2010b) and supporting documents (MOX Services, 2009c). [redacted]

The applicant's safety strategy is to prevent titanium fire events during normal operations, abnormal operations, shutdown, and maintenance of the electrolyzers. During shutdown and maintenance, the applicant will use administrative controls associated with the isolation of power to the electolyzers when they are drained (e.g., removal of fissile material from the electrolyzer, connecting pipes, or receiving tanks to the extent that is appropriate or practical before performing the maintenance, and the power supply to the electrolyzer will be locked out when the electrolyzer is drained). For normal and abnormal operations, the applicant will employ a combination of active and passive engineered features to prevent the titanium fire event.

[redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted]

[redacted]

[redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted]

[redacted]

[redacted]

The applicant initially identified the silicon nitride barrier as a principal structure, system or component (PSSC) in the revised containment air recirculation (Construction Authorization Request) (DCS, 2002). However, subsequent analysis (MOX, 2009c) determined that the barrier was not necessary to prevent electrical arcing and the resulting titanium fire event. The other IROFS listed above would be sufficient to render the event highly unlikely. The analysis found that the silicon nitride barrier is not necessary as an IROFS because the barrier plays no role in providing electrical insulation. It functions only to physically separate anode and cathode solutions. Ions can still conduct through the barrier. Therefore, rupture of the barrier would constitute no loss of insulation. Furthermore, a well rupture would not cause a temperature increase in an electrolyzer because heat released into the electrolyzer after the rupture is less than the heat released during normal operations. The electrolyzer cooling system will ensure proper cooling of the electrolyzer body.

The NRC staff further notes that, although silver deposition at the cathode in the MFFF electrolyzers is an unwanted event, the UCD plant at the La Hague facility in France has an operating silver recovery electrolyzer for which such deposition is the normal process outcome. The silver recovery electrolyzer operates without a separator (i.e., there is no barrier between anode and cathode) under normal operations. The distance between the anode and cathode in this unit is similar to that in the MFFF electrolyzers, the electrolyzer body is also made of titanium, and the absence of the barrier does not cause abnormal electrical behavior (MOX, 2009c).

[redacted]

The NRC staff review finds that the overall strategy of prevention is appropriate given the postulated consequences. The staff has reviewed the applicant's assumptions and the evaluation of the affected accident sequences and agrees with the applicant's determination that the analysis is complete and adequately addresses the issues related to potential titanium fires in KDB and KDD electrolyzers.

The NRC staff finds that this approach is acceptable to comply with the single-failure criterion. The single-failure criterion and experience at the La Hague facility, management measures (as described in Chapter 15 of this SER), quality assurance requirements (as described in the MOX Project Quality Assurance Plan (MPQAP)), and the use of codes and standards for engineered IROFS give the NRC staff reasonable assurance that this high-consequence event is highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61.

11.2.1.3.5 Loss of Confinement of Process Solutions

The applicant discussed the control strategy for leaks and loss of confinement of process solutions in process cells and outside gloveboxes in Section 8.3.1.2 of the LA (MOX, 2010a) and Sections 5.3.3 and 5.3.11 of the ISA Summary (MOX, 2010b). The applicant postulated that these loss of confinement events would occur in the process cells in the BAP because of leaks from process vessels or pipes. The identified causes for these leaks include corrosion and mechanical stresses.

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The staff's review finds that this approach relies upon practices and codes and standards typically used in the nuclear and chemical process industries to control similar hazards. By analogy, the approach should have the ability to address the potential concerns and is acceptable to the staff.

11.2.1.3.6 Oxalic Precipitation Concerns

The function of the oxalic precipitation, filtration, and oxidation (KCA) unit is to convert the plutonium nitrate solution received from the purification cycle (KPA) unit into plutonium oxide (PuO_2) powder. The major processing steps performed by the KCA unit are plutonium nitrate solution reception and conversion feed preparation, oxalate precipitation, filtration, and drying and calcination.

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11.2.1.3.7 Oxalic Mother Liquor Recovery

The function of the oxalic mother liquor recovery (KCD) unit is to receive and process mother liquor and washing solutions from the oxalic precipitation, filtration, and oxidation (KCA) unit. The KCD unit destroys excess oxalic acid, concentrates and recycles plutonium nitrate solutions to the purification cycle unit (KPA), and recycles evaporator distillates to nitric acid recovery (KPC).

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Based on the above, the staff finds that the applicant's proposed control strategy is acceptable. The staff review finds that the overall preventive strategy is appropriate given the postulated consequences. The staff has reviewed the applicant's assumptions and the evaluation of the affected accident sequences and agrees with the applicant's determination that the analysis is complete and adequately addresses the issues related to potential explosions and criticality events in the oxalic mother liquor recovery evaporator.

11.2.1.3.8 Acid Recovery Unit

The acid recovery (KPC) unit recovers and recycles nitric acid in the AP process effluents. Section 11.2.1.2.12 of this SER discusses the primary processes.

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Based on the above, the staff finds that the applicant's operational strategy for the nitric acid recovery evaporators is appropriate given the postulated consequences. The staff has reviewed the applicant's assumptions and the evaluation of the affected accident sequences and agrees with the applicant's determination that the analysis is complete and adequately addresses the issues related to potential explosions in the nitric acid recovery evaporator.

11.2.1.3.9 Off-Gas Treatment Unit (KWG)

The off-gas treatment (KWG) unit ventilation system treats off-gases generated in AP process systems by removing radionuclides, nitrous fumes, and other hazardous materials before their release to the environment through the plant stack. The KWG unit also provides a confinement barrier by maintaining negative pressure in connected process equipment. The KWG unit consists of scrubbing columns with buffer tanks, demisters, a condenser, electric heaters, HEPA filter trains, exhaust systems, and a common stack.

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Based on the above, the staff concludes that the applicant has provided design bases that sufficiently address the safety concerns regarding the off-gas treatment system. The staff has reviewed the applicant's assumptions and the evaluation of the off-gas treatment system and agrees with the applicant's determination that the analysis is complete and adequately addresses the issues related to operation of the off-gas treatment system.

11.2.1.3.10 Corrosion Control (KDB and KDD)

The staff's evaluation found that the applicant's proposed AP process employs oxidation-reduction chemistry, based on the silver (I) to silver (II) couple, to facilitate the dissolution of PuO_2 in the electrolyzers. Silver (II) is corrosive, and special alloys are necessary for electrolyzer equipment that may be exposed to silver (II). The applicant stated that the anode compartment and cathode cover of the electrolysis pot will be constructed of titanium (MOX, 2010b and MOX, 2009c). The applicant will destroy silver (II) by converting it back to silver (I) before electrolyzer solutions contact other process equipment that is fabricated out of 300-series stainless steels.

The staff notes that the applicant will employ ASTM Grade 2 titanium alloy to provide adequate corrosion resistance in the presence of the harsh chemical conditions anticipated in the electrolyzer pots. With respect to the KDB and KDD electrolyzer construction, titanium was specifically chosen as the material of construction for the electrolyzer bodies for its corrosion-resistance properties in the presence of Ag^{++} ions. The electrolyzer anode, cathode, and body assembly parts are insulated from one another through the use of corrosion-resistant, high-dielectric materials (i.e., PTFE and PVDF).

The insulating and strength properties of these materials are stable over long periods of time, as evidenced through their use in similar applications for more than 15 years of operating experience at the La Hague facility in France (MOX, 2009c). Furthermore, the applicant has committed that the electrolyzer components will be routinely checked and inspected in accordance with an established surveillance schedule, to eliminate corrosion-induced wear as a source of failure.

The staff has reviewed the applicant's assumptions and the evaluation of the affected accident sequences and agrees with the applicant's determination that the analysis is complete and adequately addresses the issues related to corrosion in the KDB and KDD electrolyzers. The staff reviewed the applicant's evaluation of the affected accident sequences and determined that the credited IROFS provide adequate protection for these sequences consistent with the requirements of 10 CFR 70.61 (see the discussion of EXP-17 in SER Section 8.1.2.4.1.3 and of F-05 in SER Section 7.1.6.4).

11.2.1.3.11 Liquid Waste (KWD)

In Section 11.2.2 of the LA (MOX, 2010a), the applicant stated that the aqueous liquid waste system or the solvent liquid waste system collects liquid radioactive waste and transfers it to SRS for disposition. The MFFF will not discharge radioactive liquid to the environment during normal and off-normal operations. Section 11.2.14 of the LA states that the aqueous waste reception (KWD) unit receives and manages LLW, americium-rich HAW, and SUW generated by the AP process units for subsequent transfer to the WSB.

Furthermore, the applicant stated that the AP process will use recycling to the maximum extent practical to minimize liquid waste. [redacted]

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The NRC staff reviewed the applicant's safety evaluation to evaluate whether the applicant will adequately protect workers, the public, and the environment during a postulated inability to transfer HAW from the MFFF. The programs that support chemical safety and the facility design must also protect against facility conditions, operator actions, or both that can affect the safety of licensed materials and present an increase in risk.

Process Description

Within the AP system, the KWD HAW unit collects contaminated radioactive aqueous wastes containing americium and other impurities extracted from the plutonium stream. The unit also collects alkaline waste and destroys any azides in it. The KWD HAW unit collects, processes, and stores these wastes before discharge. These effluents are then batch transferred from the MFFF to DOE's WSB via a dedicated double-walled underground pipeline. Figure 11.2-1 shows the process diagram for the KWD HAW unit.

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Chemical Process Inventories in KWD

The applicant expects to operate the MFFF 42 weeks per year, with 10 weeks of outage time planned for maintenance, repair, and other activities such as periodic testing of equipment. The MFFF will accept feedstock either from DOE's PDCF or AFS provided by DOE. The PDCF feedstock is the limiting volume in that it generates the higher volume of liquid waste raffinates. The maximum expected annual volume of liquid HAW is approximately 39,000 L based on an annual (42 weeks of operation) throughput of 3.5 metric tons of plutonium when processing PDCF material through the MFFF. The maximum expected annual volume of HAW generated during processing of AFS is about 34,000 L. The applicant's analysis uses the more limiting condition of PDCF feed material (39,000 L).

The applicant anticipates five planned HAW batch transfers per year between the MFFF and the WSB, with a nominal batch size of approximately 7,700 L. This batch volume represents approximately 73 percent of the capacity of 10,500-L KWD*TK4050, with more than 25 percent of that tank's capacity (per batch) remaining as buffer.

The storage tanks KWD*TK4020, KWD*TK4030, KWD*TK4040, and KWD*TK4050 have a combined capacity of over 29,000 L. Although the applicant does not expect to use the combined volume of these four vessels during normal operations (since periodic discharges of HAW are planned), the HAW storage capacity of these four vessels represents the expected volume of HAW generated in more than 29 weeks of normal AP operations.

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Accident Sequences

The ISA Summary (MOX, 2010b) includes potential accident sequences and identifies selected controls that either prevent or mitigate the consequences of these accident sequences to an acceptable level.

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The staff has reviewed the information provided by the applicant and evaluated the applicant's analyses for these sequences.

Review of Selected Accident Sequences

The staff reviewed the applicant's evaluation of the selected accident sequences potentially affected by the postulated inability to transfer HAW, as well as the proposed controls to mitigate these effects. The staff evaluated accident sequences identified by the applicant and concludes that the applicant's evaluation of accident scenarios is complete and appropriate and that the new IROFS (in addition to the IROFS already included) are necessary and provide protection against these affected sequences and meet the performance requirements of 10 CFR 70.61.

The staff has reviewed the applicant's assumptions and the evaluation of the affected accident sequences and agrees with the applicant's determination that the analysis is complete and adequately addresses the issues related to the inability to transfer HAW. The staff reviewed the applicant's evaluation of the affected accident sequences and determined that the credited IROFS provide adequate protection for the affected accident sequences consistent with the requirements of 10 CFR 70.61.

Accident Sequences Conclusion

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The staff concludes that the applicant has appropriately identified potential accident sequences resulting from a postulated inability to transfer liquid HAW from the MFFF identified in the safety evaluation document. The information provided by the applicant meets the accident sequence and likelihood guidance in Section 8.4.3 of NUREG-1718 (NRC, 2000), and the staff finds this information acceptable.

<u>Accident Consequences</u>

The staff finds that the applicant has identified the consequences of analyzed chemical accident sequences and estimated them conservatively. Based on its review of the safety evaluation and the affected accident sequences, the staff concludes that the applicant has adequately identified the consequences of the affected accident sequences involving the chemical hazards of licensed materials and hazardous chemicals produced from licensed material. The information provided by the applicant, as described above, meets the guidance in Section 8.4.3 of NUREG-1718 (NRC, 2000) and is, therefore, acceptable.

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The staff reviewed the credited IROFS for affected accident sequences and concludes that the applicant has adequately identified IROFS to prevent or mitigate the consequences of the affected accident sequences involving the chemical hazards of licensed materials and hazardous chemicals produced from licensed material.

Management Measures

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The staff finds that the applicant has sufficiently identified management measures to maintain IROFS controls available and reliable to perform their safety functions when needed, as required under 10 CFR 70.62, “Safety Program and Integrated Safety Analysis.”

Staff Conclusions

The staff reviewed the applicant’s evaluation of the liquid HAW storage capacity of the MFFF and the ability of the applicant to bring the MFFF to a safe configuration in the event that the ability to transfer liquid HAW from the facility is interrupted in the short, intermediate and long term, using the criteria previously listed. Based upon its review of this evaluation, the staff finds that the applicant has described the facility, equipment, and processes in sufficient detail to meet the requirements of 10 CFR 70.22, “Contents of Applications,” and 10 CFR 70.65, “Additional Content of Applications,” consistent with the acceptance criteria of Section 8.4.3 of NUREG-1718 (NRC, 2000). The staff also finds that the 800-L operating limit, supporting the 400-L IROFS safety limit, satisfies the applicant’s setpoint commitment, as described in Section 11.2.1.3.11 (p. 11-48) of NUREG-1821 (NRC, 2005).

Lastly, the staff also has reasonable assurance that the applicant has identified hazards and accident sequences and credited IROFS sufficient to meet the performance requirements of 10 CFR 70.61, consistent with the acceptance criteria of Sections 6.4 and 8.4 of NUREG-1718 (NRC, 2000).

11.2.1.3.12 Sampling Systems

The safety function of IROFS sampling is to ensure that: (1) proper concentrations of reagents are introduced into the process, (2) process solutions remain within the correct composition, and (3) the contents of drip trays are identified (as necessary) and appropriately recovered.

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The staff reviewed the use of IROFS sampling as an enhanced administrative control for the prevention of explosion, loss of confinement, and criticality events and concludes that the applicant has adequately identified these IROFS to prevent or mitigate the consequences of the affected accident sequences involving the chemical hazards of licensed materials and hazardous chemicals produced from licensed material.

11.2.1.4 Design Basis of the Items Relied Upon for Safety and Applicable Baseline Design Criteria

The requirements of 10 CFR 70.64(a)(5), related to chemical protection, states that the facility design must provide for adequate protection against chemical risks produced from licensed material, facility conditions that affect the safety of licensed material, and hazardous chemicals produced from licensed material. The staff finds that the applicant has included sufficient information to demonstrate that the MFFF design provides protection from chemical risks from licensed material, facility conditions that affect the safety of licensed material, and hazardous chemicals produced from licensed material. Chapter 8 of the LA (MOX, 2010a) describes chemical safety. The applicant has adequately evaluated chemical and explosion hazards and their impact on the facility operation (including identification of IROFS) as described in the ISA Summary (MOX, 2010b) to meet the requirements of 10 CFR 70.64(a)(5) (see SER Section 8).

Related to chemical protection, the fire protection baseline design criterion in 10 CFR 70.64(a)(3) includes the criterion for explosion protection:

> Fire protection. The design must provide for adequate protection against fires and explosions.

The staff finds that the MFFF design provides for fire protection features and systems as described in Chapter 7 of the LA (MOX, 2010a) and Section 4 of the ISA Summary (MOX, 2010b), and evaluated in Chapter 7 of this SER. Furthermore, the staff finds that the applicant has adequately evaluated fire and explosion hazards, and their impacts on facility operation (including identification of IROFS), as described in the ISA Summary. Therefore, the staff finds that the applicant has provided sufficient information to meet the requirements of 10 CFR 70.64(a)(3).

The applicant stated that there is reasonable assurance that the IROFS will be sufficiently reliable and available based on the use of standard nuclear industry engineering practices. The facility general design philosophy, design bases, system design, and commitments to applicable management measures incorporate these practices. These practices ensure that applicable industry codes and standards are used, adequate safety margins are provided, engineering features are used to the extent practicable, the defense-in-depth philosophy is incorporated into the design, and IROFS will be appropriately maintained.

11.2.2 Evaluation Findings

In Section 11.2 and Chapter 5 of the LA (MOX, 2010a), the applicant provided design-basis information for chemical process safety IROFS identified for the MFFF. Based on the staff's review of the chapters and supporting information provided by the applicant that are relevant to AP and chemical process safety, the staff finds that, for the reasons discussed above, MOX Services has met the baseline design criteria set forth in 10 CFR 70.64(a)(3) for explosions and in 10 CFR 70.64(a)(5) for chemical safety. Furthermore, the staff concludes, pursuant to 10 CFR 70.23(b), that the design bases of the IROFS identified by the applicant will provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents.

REFERENCES

(DMIC, 1964) Defense Metals Information Center, Battelle Memorial Institute, "Fire Hazards Associated with the Use of Titanium in Aircraft," DMIC Technical Note AD609342, Columbus, OH, June, 1964.

(MOX, 2010a) Shaw AREVA MOX Services, “Mixed Oxide Fuel Fabrication Facility License Application,” Aiken, SC, March 2010.

(MOX, 2010b) Shaw AREVA MOX Services, “Mixed Oxide Fuel Fabrication Facility Integrated Safety Analysis Summary,” Aiken, SC, March 2010.

(MOX, 2009a) Shaw AREVA MOX Services, “Safety Evaluation of the Postulated Inability to Transfer High Alpha Waste from the MOX Fuel Fabrication Facility and the Short, Intermediate, and Long-term Storage of High Alpha Waste in the KWD Unit, DCS01-RRJ-DS-ANS-H-38426-2,” Aiken, SC, 2009.

(MOX, 2009b) Shaw AREVA MOX Services, “Calculation of the Volume Sent to KWD High Alpha Tanks During a Pu Flushout in KPA,” DCS01-KWD-DS-CAL-F-12160-2, Aiken, SC, 2009.

(MOX, 2009c) Shaw AREVA MOX Services, “Analysis of the Potential for a Short Circuit of Arcing in an Electrolyzer due to Failure of the Sintered Silicon Nitride Barrier,” DCS01-AAS-DS-ANS-H-38440-0, Aiken, SC, 2009.

(MOX, 2009d) Shaw AREVA MOX Services, “Chemical Consequences for Potential Chemical Hazard Events,” DCS01-KKJ-DS-CAL-H-35604-2, Aiken, SC, 2009.

(NRC, 2000) U.S. Nuclear Regulatory Commission, NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility,” Washington, DC, August 2000.

(NRC, 2005) U.S. Nuclear Regulatory Commission, NUREG-1821, “Final Safety Evaluation Report on the Construction Authorization Request for the Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina,” Washington, DC, March 2005.

(ANSI/ASME, 1996) American Society of Mechanical Engineers, B31.3-1996, Process Piping, including 1998 Addenda.

11.3 Ventilation and Confinement Systems

The purpose of this review is to determine, with reasonable assurance, whether the applicant’s ventilation and confinement systems will adequately protect workers, the public, and the environment under normal and accident conditions. The review is also to determine whether the ventilation and confinement systems, as well as their subcomponents, identified as items relied on for safety (IROFS) pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 70, “Domestic Licensing of Special Nuclear Material,” and encompassed by the hazard and accident analysis of the Integrated Safety Analysis (ISA) Summary (MOX, 2009a), will be available and reliable to perform their intended safety function when needed.

The staff evaluated the information provided by the applicant for ventilation and confinement systems by reviewing Section 11.3 of the license application (LA) (MOX, 2009b), other sections of the LA, and supplementary information provided by the applicant. The staff closely coordinated the review of ventilation and confinement systems and operating strategies with the review of the ISA Summary (MOX, 2009a) (see Chapter 5 of this safety evaluation report (SER)) and the review of other plant systems. The staff used the guidance in NUREG-1718 (NRC, 2000) to prepare its review.

11.3.1 Regulatory Requirements

For the LA to be considered acceptable, the applicant must satisfy the following regulatory requirements applicable to ventilation and confinement systems:

- 10 CFR 70.22, “Contents of Applications,” specifically relating to the requirement that the applicant is to provide a description of the equipment and facilities and propose procedures to protect health and minimize danger to life and property
- 10 CFR 70.23, “Requirements for the Approval of Applications,” specifically relating to the requirement that the Commission determine that the proposed equipment, facilities, and procedures are adequate to protect health and minimize danger to life and property
- 10 CFR 70.61(e), specifically relating to the requirement that each engineered or administrative control or control system that is needed to meet the performance requirements be designated as an IROFS and relating to the safety program that ensures that each IROFS will be available and reliable to perform its intended function when needed
- 10 CFR 70.62, “Safety Program and Integrated Safety Analysis,” specifically relating to the establishment and maintenance of a safety program and to the performance of an ISA
- 10 CFR 70.64, “Requirements for New Facilities or New Processes at Existing Facilities,” specifically relating to the application of baseline design criteria (BDC) and defense-in-depth practices to new facilities or new processes at existing facilities

11.3.2 Regulatory Acceptance Criteria

Sections 11.4.5.1 and 11.4.5.2 of NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility” (NRC, 2000), provide the acceptance criteria for the U.S. Nuclear Regulatory Commission’s (NRC’s) review of the applicant’s ventilation and confinement systems.

11.3.3 Staff Review and Analysis

11.3.3.1 System Description

In the LA, the applicant proposed a ventilation and confinement system to confine radioactive materials within process areas and gloveboxes and to ensure minimum dispersal of radioactive materials during routine operations and under accident conditions. The applicant’s design uses ventilation zones that are operated at pressure differentials designed such that air leakage moves from areas of low radiation hazard into areas with greater radiation hazard (see Figure 11.3-1). Static barriers, such as process vessels, gloveboxes, and room walls, bound these zones. Separate ventilation systems that operate at different negative pressures provide the proper differential pressures between confinement zones.

The C4 confinement zone or primary confinement zone consists of static and dynamic systems. Static systems consist of gloveboxes, process vessels, and equipment containing radioactive material where the hazards are the greatest. Dynamic systems include the very high depressurization (VHD) exhaust system, which maintains glovebox differential pressures to minimize the spread of radioactive contamination. Airflows in the VHD system pass through high-efficiency particulate air (HEPA) filters located at the glovebox, intermediate filter assemblies consisting of a prefilter and HEPA filter at the fire area boundary, and final filtration assemblies consisting of a roughing filter, a prefilter, and two banks of HEPA filters before exhausting to the facility stack. HEPA filtration is also provided in glovebox supply air systems. Section 11.3.1 of the LA presents details of the C4 confinement system.

The C3 confinement zone or secondary confinement zone consists of process rooms that contain gloveboxes and process vessels where contamination risks are lower than in the C4 zone. Static barriers consist of room walls. The high depressurization exhaust (HDE) system maintains differential pressures in the C3 zones. The HDE system consists of intermediate filter boxes containing prefilters and HEPA filters and final filtration assemblies consisting of roughing filters, prefilters, and two stages of HEPA filtration before exhausting to the facility stack. HEPA filtration is also provided in supply air systems for C3 zones. Section 11.3.2 of the LA presents details of the C3 confinement system.

The process cell zones consist of rooms containing process vessels not enclosed by gloveboxes. These zones have a low risk of contamination but, under accident conditions, the contamination could be high. The process cell exhaust system maintains differential pressures in the process cell zones. The final filtration assembly for this ventilation system consists of roughing filters, prefilters, and two stages of HEPA filtration before exhausting to the facility stack. HEPA filtration is also provided in process cell supply air systems. Section 11.3.3 of the LA presents details of the process cell confinement system.

The C2 confinement zone or tertiary confinement zone consists of process rooms containing sealed plutonium containers, sealed fuel rods or assemblies, corridors surrounding the process cell and C3 zones, and general areas where plutonium is not present. These zones have a very low risk of contamination. The medium depressurization exhaust (MDE) system maintains differential pressures in the C2 zone. The MDE system consists of final filtration assemblies that include roughing filters, prefilters, and two stages of HEPA filtration before exhausting to the facility stack. HEPA filtration is also provided in supply air systems for the C2 zone. Section 11.3.4 of the LA presents details of the C3 confinement system.

C1 confinement zones are areas that open to the outside of buildings and have an extremely low risk of contamination. These areas are at atmospheric pressure.

In addition to the above static and dynamic systems, the facility also has a supply air system, which filters incoming airflows through prefilters and HEPA filters. Section 11.3.5 of the LA presents details of the supply air system. Emergency control rooms, the emergency generator building, the truck bay, the shipping and receiving area, safe havens, the reagents processing building, and entry control areas also have ventilation systems. Sections 11.3.6, 11.3.7, and 11.3.8 of the LA present the details of these systems.

11.3.3.1.1 Functions and Major Components

In the LA, the applicant proposed a ventilation and confinement system to confine radioactive materials within process areas and gloveboxes and to ensure minimum dispersal of radioactive materials during routine operations and under accident conditions (MOX, 2009b). The proposed system meets the recommendations in Regulatory Guide 3.12, “General Design Guide for Ventilation Systems of Plutonium Processing and Fuel Fabrication Plants” (NRC, 1973), and is, therefore, consistent with the staff's acceptance criteria in Sections 11.4.5.1 and 11.4.5.2 of NUREG-1718 (NRC, 2000). In addition, the ventilation and confinement systems consist of the following:

(1) Ventilation zones are operated at pressure differentials designed such that air leakage occurs from areas of low radiation hazard into areas with greater radiation hazard (LA Section 11.3, Figure 11.3-1). Static barriers, such as process vessels, gloveboxes, and room walls, bound these zones. Control devices to maintain proper differential pressures are provided and alarmed. This follows the guidance in Sections 11.4.5.1 and 11.4.5.2.A.i–ii of NUREG-1718 and is, therefore, acceptable.

(2) Controls are also in place to isolate systems, if necessary, and to maintain confinement zone differential pressures. Monitoring instrumentation, alarms, and controls ensure that pressure differentials in confinement zones are maintained, alternative power supplies are actuated when needed, and the consequences of accidents are mitigated (Sections 11.3.1.3, 11.3.2.3, 11.3.3.3, 11.3.4.3, 11.3.5.3, and 11.3.6.3 of the LA). This follows the guidance in Section 11.4.5.2.A.iii of NUREG-1718 and is, therefore, acceptable.

(3) The supply air system controls are interlocked with exhaust fan controllers to prevent supply fan operation unless the exhaust fans are operational and to prevent overpressurization of confinement zones. Section 11.3.5.3 of the LA discusses supply air system controls. This follows the guidance in Section 11.4.5.2.A.iv of NUREG-1718 and is, therefore, acceptable.

(4) A system of final filter assemblies is in accordance with American Society of Mechanical Engineers (ASME) standard ASME AG-1, "Code on Nuclear Air and Gas Treatment" (ASME, 1997), and ANSI/ASME N509, "Nuclear Power Plant Air-Cleaning Units and Components" (ASME, 1996), consisting of HEPA filters, stainless steel/glass fiber prefilters, and stainless steel roughing filters intended to remove radioactive materials from process areas and occupied areas during routine operations and under accident conditions. The stainless steel/glass fiber prefilters and stainless steel roughing filters are designed to remove hot embers and a large percentage of the soot to protect the final HEPA filters from fire damage and excessive soot plugging (Sections 11.3.1.2, 11.3.2.2, 11.3.3.2, and 11.3.4.2 of the LA). This follows the guidance in Section 11.4.5.1 of NUREG-1718 and is, therefore, acceptable.

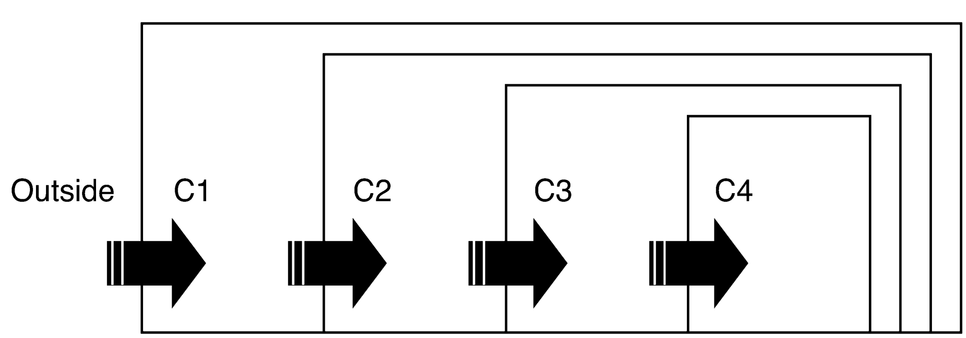

Figure 11.3-1 Ventilation confinement zones

(5) The design is in accordance with ASME AG-1 (ASME, 1997) and ASME N510, "Testing

of Nuclear Air Treatment Systems" (ASME, 1995), to allow in-place filter testing and in-service surveillance of HEPA filters to ensure operability and required functional performance (Sections 11.3, 11.3.1.2, 11.3.2.2, 11.3.10.1.1, 11.3.10.1.2, 11.3.10.1.3, and 11.3.10.1.6 of the LA). The systems are designed also to allow operability and performance testing of fans, filters, and other safety-related components (Sections 11.3.1.2, 11.3.2.2, 11.3.3.2, and 11.3.4.2 of the LA). This follows the guidance in Sections 11.4.5.2.B.i–ii of NUREG-1718 and is, therefore, acceptable.

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(14) Gloveboxes consist of welded stainless steel enclosures with windows, alone and in interconnected groups, which act as a primary barrier to confine hazardous (radioactive, toxic, or flammable) materials and to provide structural support capable of protecting process equipment during a postulated seismic event. The mixed oxide fuel fabrication facility (MFFF) personnel's access to equipment inside the gloveboxes is provided through access holes in the glovebox windows fitted with gloves that maintain the confinement boundary.

Glovebox window panels, viewing ports, or video cameras give visual access to the gloveboxes. Light fixtures provide illumination and are generally located outside the glovebox. The windows are clear rectangular panels that fit into gasketed frames that cover specifically designed openings in the glovebox shell. The windows proposed by the applicant are polycarbonate (Lexan®), which may have lead-glass panels to provide additional radiation protection. The polycarbonate windows are qualified to withstand structural and accident loads and maintain their integrity.

Gloveboxes have pass-through connectors in glovebox shells that are used to bring processes and utilities (e.g., air, electricity, and water) inside the glovebox. These connectors are designed and tested to ensure that glovebox pressure integrity stays within the maximum leakage criteria. Primary process equipment contains mixed oxide (MOX) product in various forms (i.e., powder, pellets, trays, and rods). These MOX forms are manufactured, transferred, stored, and maintained inside of gloveboxes (Section 11.3.9.1 of the LA). Gloveboxes are designed to maintain the confinement boundary and structural integrity during and following a design-basis seismic event.

Gloveboxes are designed, fabricated, installed, operated, and maintained in accordance with practices used at the MELOX and La Hague facilities and national codes and standards (Sections 11.3.9 and 11.3.10.2 of the LA). Fabrication and welding codes include ANSI Standard ANSI N690, "Specification for the Design, Fabrication, and Erection of Safety Related Steel Structures for Nuclear Facilities" (ANSI, 1994), and American Welding Society (AWS) Standard D1.1, "Structural Welding Code" (AWS, 1998). Gloveboxes are tested in accordance with American Glovebox Society (AGS) Standard AGS-G001, "Guideline for Gloveboxes" (AGS, 1998), or American Society for Testing and Materials (ASTM) Standard E499, "Standard Test Methods for Leaks Using the Mass Spectrometer Leak Detector in the Detector Probe Mode 1, 2" (ASTM, 2000). This follows the guidance in Section 11.4.5.2.K.i of NUREG-1718 and is, therefore, acceptable.

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[redacted] [redacted]

(18) The glovebox system is designed to prevent physical interaction with confinement boundary elements under worst-case loading conditions associated with normal, off-normal, accident, and design-basis events in accordance with ANSI N690 (ANSI, 1994) (Section 11.3.10.2 of the LA).

(19) Ductwork is designed, fabricated, and tested in accordance with ASME B31.3, "Process Piping" (ASME, 1998), and U.S. Energy Research and Development Administration (ERDA) 76-21, "Nuclear Air Cleaning Handbook" (ERDA, 1976) (Section 11.3.10.1 of the LA).

11.3.3.1.2 Control Concepts

As described in Sections 11.3.1.3, 11.3.2.3, 11.3.4.3, 11.3.5.3, 11.3.6.3, 11.3.7.3, and 11.3.8.3 of the LA (MOX Services, 2009b), the ventilation and confinement system provides for instrumentation and control (I&C) systems to monitor and control the ventilation and confinement components. Section 11.6 of this SER discusses the I&C systems. These I&C systems include the following:

[redacted] [redacted]

[redacted] [redacted]

- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]

11.3.3.1.3 System Interfaces

Individual confinement zone ventilation systems are interconnected to ensure that the proper negative pressures are maintained within confinement zones. Interlocks are provided to ensure that air from zones of higher radiation hazard cannot flow to zones of lower radiation hazard. The ventilation and confinement system also interfaces with the normal, standby, emergency, and uninterruptible power supplies so that systems can function properly in the event of power loss. Gloveboxes have functional and physical interfaces with ventilation systems; electrical systems; air, gas, chilled water, and demineralized water systems; chemical processing systems; and fire suppression systems (Sections 11.3.1, 11.3.1.4, 11.3.2, 11.3.2.4, 11.3.3, 11.3.3.4, 11.3.4, 11.3.4.4, 11.3.5, 11.3.5.4, 11.3.6, 11.3.6.4, 11.3.7, 11.3.7.4, 11.3.8, and 11.3.8.4 of the LA (MOX, 2009b)).

11.3.3.2 Baseline Design Criteria

Sections 11.3.1, 11.3.2, 11.3.3, 11.3.4, 11.3.5, 11.3.7, and 11.3.10 of the LA discuss the design bases for the ventilation and confinement system IROFS (MOX, 2009b).

The staff reviewed the design bases for the ventilation and confinement system IROFS to ensure that there is reasonable assurance of protection against natural phenomena and the consequences of potential accidents. The design bases in accordance with Regulatory Guide 3.12 (NRC, 1973) and other industry air cleaning standards such as ASME N510 (ASME, 1995) and ASME AG-1 (ASME, 1997). The staff also reviewed the design bases against the BDC in 10 CFR 70.64. The following discusses how the applicant meets the BDC.

The staff reviewed the LA to ensure that the design bases would meet the quality standards and that management measures would be appropriately applied so that there is reasonable assurance that IROFS will be available and reliable to perform their intended function when needed. The MOX Quality Assurance Program Plan discusses the application of quality standards and Chapter 15 of the LA discusses management measures. This meets the regulatory requirements in 10 CFR 70.64(a)(1) and is, therefore, acceptable.

For potential accidents involving fires, consistent with Chapter 7 of the LA, the staff reviewed the proposed design bases and the ISA Summary (MOX, 2009a), which describe the analyses and design features pertinent to fire protection. Design features of the ventilation and confinement system for fire protection include filter assembly redundancy, use of air dilution to mitigate the high-temperature effects of a fire, use of stainless steel roughing filters and stainless steel/glass fiber prefilters to prevent hot particles larger than 1 micron from contacting and starting fires on filters, and use of two redundant banks of HEPA filters in each filter assembly having a temperature rating of 204 degrees C (400 degrees F) and meeting the standards set by ASME AG-1 (ASME, 1997).

[redacted]

[redacted]

[redacted]

[redacted]

[redacted] [redacted]

[redacted] [redacted]

Therefore, based on the review of the applicant's analyses, the staff finds that the filtration assemblies can perform their intended function under conditions of fire and that use of a removal efficiency of 99.99 percent is acceptable.

[redacted]

Based on the above considerations, the staff finds that the fire protection design meets the regulatory requirements in 10 CFR 70.64(a)(3) and is, therefore, acceptable.

The staff reviewed the design bases of the ventilation and containment system IROFS to ensure that they can withstand the effects of environmental conditions and the dynamic effects associated with normal operations, maintenance, testing, and postulated accidents. The ventilation and confinement system IROFS are designed to withstand fire and chemical effects. In-place testing and maintenance of HEPA filters are performed in accordance with ASME AG-1 (ASME, 1997) and ASME N510 (ASME, 1995). HEPA filters are designed to withstand applicable pressure transients considering filter loadings and fan suction pressures. Filter replacement will be performed using bag-in/bag-out procedures to reduce the possibility of spreading contamination. This meets the regulatory requirements in 10 CFR 70.64(a)(4) and is, therefore, acceptable.

The staff reviewed the design bases of the ventilation and confinement system to ensure that the system provides adequate protection against chemical risks produced from licensed material and facility conditions that affect the safety of licensed material and hazardous chemicals produced from licensed material. Gloveboxes are constructed of welded stainless steel to resist the corrosive effects of chemicals used in aqueous polishing and the MOX fuel fabrication processes. In addition, ductwork and filter assemblies upstream of the final filters are stainless steel, and filter materials will be designed to withstand the chemical effects resulting from normal operations. Chapter 8 of the LA discusses chemical safety (MOX, 2009b). This meets the regulatory requirements in 10 CFR 70.64(a)(5) and is, therefore, acceptable.

The staff reviewed the design bases of the ventilation and confinement system to ensure that it provides for emergency capability to control the release of licensed material during normal operations and under postulated accident conditions. Release of licensed material is controlled by the use of redundant HEPA filter banks in redundant filter assemblies. Individual HEPA filters are tested to ensure that they are capable of removing at least 99.97 percent of 0.3-micron particles. Following installation, IROFS HEPA filters are tested in place in accordance with ASME AG-1 (ASME, 1997) and ASME N510 (ASME, 1995) to ensure that leakage around filter banks is less than 0.05 percent (Section 11.3 of the LA (MOX, 2009b)). This meets the regulatory requirements in 10 CFR 70.64(a)(6) and is, therefore, acceptable.

The staff reviewed the design bases of the ventilation and confinement system with respect to the electrical power supply. The C4 confinement system is supplied by normal, standby, emergency, and uninterruptible power supplies. The C3 exhaust system, process cell exhaust system, the emergency control room, and emergency diesel generator systems are supplied by normal, standby, and emergency power supplies. The C2 confinement system and the supply air system are supplied by normal and standby power supplies. These diverse power supply systems will ensure the continued operation of ventilation and confinement system IROFS. (Section 11.5 of this SER presents the staff's review of the electrical systems.) This meets the regulatory requirements in 10 CFR 70.64(a)(7) and is, therefore, acceptable.

The staff reviewed the proposed design bases of ventilation and confinement system IROFS to ensure that they provide for adequate inspection, testing, and maintenance to ensure availability and reliability to perform their function when needed. Redundant filter assemblies are provided so that single filter assemblies can be taken off line for maintenance, testing, and filter replacement. Dampers can be used to isolate individual filter assemblies and fans. The filter assembly design includes provisions for in-place testing of HEPA filters in accordance with ASME AG-1 (ASME, 1997) and ASME N510 (ASME, 1995). Filter assemblies use bag-in/bag-out designs for filter replacement to minimize the possibility of spreading contamination. This meets the regulatory requirements in 10 CFR 70.64(a)(8) and is, therefore, acceptable.

The staff reviewed the proposed design bases of the ventilation and confinement system to ensure that the system provides for criticality control and adherence to the double-contingency principle. Based on experience from the MELOX site, the applicant assumed that up to 3 kilograms (kg) (6.6 pounds (lb)) of plutonium dioxide (PuO_2) could exist in the glovebox HEPA filter located in the pellet grinding glovebox, where material becomes airborne at a rate of 0.3 grams per hour (0.01 ounces per hour), assuming that the HEPA filters are replaced at 450-day intervals. This amount would be subcritical, as a quantity of 3 kg (6.6 lb) of PuO_2 is substantially less than the minimum critical mass. The American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 8.1, "Nuclear Criticality Safety in

Operations with Fissionable Materials Outside Reactors" (ANSI/ANS, 1988), contains single-parameter (i.e., always safe) subcritical limits for $^{239}PuO_2$ containing not more than 1.5 weight percent (wt%) water. At full density, the subcritical limit is 10.2 kg (22.5 lb); at half density, the subcritical limit is 27 kg (59.5 lb). This would bound the worst-case conditions that could be found in the HEPA filters, because the ANSI limits conservatively assume that all of the plutonium is ^{239}Pu (MOX plutonium will have at least 4 wt% ^{240}Pu), and the maximum density for unsintered PuO_2 powder (DCS 2004, Table 6-2) falls within the density range covered by the limits in ANSI/ANS 8.1.

[redacted]

The staff reviewed the proposed design bases of the ventilation and confinement system to ensure that it provides for I&C systems to monitor and control the ventilation and confinement IROFS. These I&Cs include (1) pressure I&Cs to maintain proper negative pressures in each of the separate confinement zones, (2) manual and automatic damper controls to regulate air and gas flows within gloveboxes and confinement zones, (3) controls for the transfer of alternate power supplies, (4) instrumentation to measure differential pressures across filter banks, (5) variable-speed controls for fan operation, (6) air temperature and airflow instrumentation, and (7) nitrogen and dry air supply controls. Section 11.6 of this SER presents the staff's review of I&C systems. This meets the regulatory requirements in 10 CFR 70.64(a)(10) and is, therefore, acceptable.

11.3.3.3 Items Relied on for Safety

Section 5.5.3 of the ISA Summary (MOX, 2009a) identifies IROFS associated with the confinement and ventilation systems.

11.3.3.4 Accident Analyses

The following loss-of-confinement (LOC) accident sequences are applicable to the confinement systems addressed in this SER chapter:

LOC-1 Overtemperature (glovebox dump valves, differential pressure switches, and the VHD and HDE ventilation systems)

LOC-2 Small breach in glovebox boundary

LOC-8 Overpressurization or underpressurization of the glovebox

LOC-9 Excessive temperature in a glovebox

LOC-10 Glovebox dynamic exhaust failure

LOC-12 Sintering furnace confinement boundary failure

11.3.3.4.1 Overtemperature (LOC-1)

In this accident scenario, glovebox confinement is lost as the result of high-temperature conditions caused by equipment malfunctions that result in excessive heat generation (see Section 5.3.3.2.1 of the ISA Summary (MOX, 2009a)). [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Based on the above discussion and a combination of the proposed IROFS, the staff agrees with the applicant that an overtemperature event that has a consequence exceeding the 10 CFR 70.61 performance requirements is highly unlikely. The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 if an overtemperature event occurs.

11.3.3.4.2 Small Breach in Glovebox Boundary (LOC-2)

In this accident scenario, glovebox confinement is lost as the result of equipment malfunctions caused by glove failures, bagport failures, inadvertent opening of manual room air supply valves, insufficient oil in hydraulic valves, vacuum breaker failures, glovebox seal failures, and breaches on the pneumatic transfer system (see Section 5.3.3.2.2 of the ISA Summary (MOX, 2009b)). [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Based on the above discussion and a combination of the proposed IROFS, the staff agrees with the applicant that a small breach in the glovebox event that has a consequence exceeding the 10 CFR 70.61 performance requirements is highly unlikely. The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 if a small breach of the glovebox event occurs.

11.3.3.4.3 Overpressurization or Underpressurization of the Glovebox (LOC-8)

In this accident scenario, glovebox confinement is lost as the result of a loss of structural integrity of the glovebox due to excessive positive or negative pressures. [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Based on the above discussion and a combination of the proposed IROFS, the staff agrees with the applicant that an underpressurization or overpressurization in the glovebox that has a consequence exceeding the 10 CFR 70.61 performance requirements is highly unlikely. The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 if an underpressurization or overpressurization in the glovebox event occurs.

11.3.3.4.4 Overtemperature in Glovebox Due to Radioactive Decay (LOC-9)

In this accident scenario, confinement is lost as the result of high temperature conditions caused by excessive heat generation from the radioactive decay of stored radioactive materials (see Section 5.3.3.2.11 of the ISA Summary (MOX, 2009a)). [redacted]

[redacted] [redacted]

[redacted]

Based on the above discussion and a combination of the proposed IROFS, the staff agrees with the applicant that an overtemperature in the glovebox due to radioactive decay that has a consequence exceeding the 10 CFR 70.61 performance requirements is highly unlikely. The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 if an overtemperature in glovebox due to radioactive decay event occurs.

11.3.3.4.5 Glovebox Dynamic Exhaust Failure (LOC-10)

In this accident scenario, the glovebox and VHD ventilation system fails, resulting in a loss of negative pressure in a glovebox or group of gloveboxes (see Sections 5.3.3.2.11.1, 5.3.3.2.11.2, and 5.3.3.2.11.3 of the ISA Summary (MOX, 2009b)). [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Based on the above discussion and a combination of the proposed IROFS, the staff agrees with the applicant that a glovebox dynamic exhaust failure event that has a consequence exceeding the 10 CFR 70.61 performance requirements is highly unlikely. The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 if a glovebox dynamic exhaust failure event occurs.

11.3.3.4.6 Sintering Furnace Confinement Boundary Failure (LOC-12)

In this accident scenario, the sintering furnace confinement boundary fails, resulting in a release of radioactive material directly to the process room (see Section 5.3.3.2.13 of the ISA Summary (MOX, 2009a)). [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Based on the above discussion and a combination of the proposed IROFS, the staff agrees with the applicant that a sintering furnace confinement boundary failure event that has a consequence exceeding the 10 CFR 70.61 performance requirements is highly unlikely. The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 if a sintering furnace confinement boundary failure event occurs.

11.3.2 Evaluation Findings

In Section 11.3 of the revised LA and Section 4.3 of the ISA Summary, the applicant provided information for the ventilation and confinement systems that it identified as IROFS for the proposed MFFF. The staff evaluated the above information and based on the review of this information and relevant supporting information provided by the applicant, the staff concluded that the applicant's ventilation and confinement system designs and operations satisfy the staff's acceptance criteria in NUREG-1718 and the systems are adequately available and reliable to perform their intended functions when needed. The staff finds that the applicant has satisfactorily complied with the applicable regulatory requirements, including the performance requirements, the baseline design criteria, and the defense-in-depth practices contained in 10 CFR Part 70.

REFERENCES

(AGS, 1998) American Glovebox Society, Standard AGS-G001, "Guideline for Gloveboxes," Santa Rosa, CA, 1998.

(ANSI, 1994) American National Standards Institute, Standard ANSI N690, “Specification for the Design, Fabrication, and Erection of Safety Related Steel Structures for Nuclear Facilities,” New York, 1994.

(ANSI, 1988) American National Standards Institute/American Nuclear Society, Standard ANSI/ANS 8.1, “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors,” New York, 1988.

(ANSI/ANS, 1988) American National Standards Institute/American Nuclear Society, Standard 8.1, “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors,” New York, 1988.

(ANSI/ASHRAE, 1999) American National Standards Institute/American Society of Heating, Refrigeration, and Air Conditioning Engineers, Standard ANSI/ASHRAE 52.2, “Method of Testing General Ventilation Air Cleaning Devices for Removal Efficiency by Particle Size,” New York, 1999.

(ANSI/ASHRAE, 1992) American National Standards Institute/American Society of Heating, Refrigeration, and Air Conditioning Engineers, Standard ANSI/ASHRAE 52.1, “Gravimetric and Dust Spot Procedures for Testing Air Cleaning Devices Used in General Ventilation for Removing Particulate Matter,” New York, 1992.

(ASME, 1998) American Society of Mechanical Engineers, Standard ASME B31.3, “Process Piping,” New York, 1998.

(ASME, 1997) American Society of Mechanical Engineers, Standard ASME AG-1, “Code on Nuclear Air and Gas Treatment,” New York, 1991.

(ASME, 1996) American Society of Mechanical Engineers, Standard ANSI/ASME N509, “Nuclear Power Plant Air-Cleaning Units and Components,” New York, 1996.

(ASME, 1995) American Society of Mechanical Engineers, Standard ANSI/ASME N510, “Testing of Nuclear Air-Cleaning Systems,” New York, 1980.

(ASTM, 2000) American Society for Testing and Materials, Standard E499, “Standard Test Methods for Leaks Using Mass Spectrometer Leak Detector in the Detector Probe Mode 1, 2,” West Conshohocken, PA, 2000.

(AWS, 1998) American Welding Society, Standard AWS D1.1, “Structural Welding Code,” Miami, FL, 1998.
(DCS 2004) Construction Authorization Request for a Mixed Oxide (MOX) Fuel Fabrication Facility, Aiken, SC, 2004.

(ERDA, 1976) U.S. Energy Research and Development Administration, ERDA 76-21, “Nuclear Air Cleaning Handbook,” Oak Ridge, TN, 1976.

(NFPA 1996) National Fire Protection Association 72, National Fire Alarm and Signaling Code, 1996

(NRC, 2000) U.S. Nuclear Regulatory Commission, NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility," Washington, DC, August 2000.

(NRC, 1998) U.S. Nuclear Regulatory Commission, NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," Washington, DC, March 1998.

(NRC, 1988) U.S. Nuclear Regulatory Commission, Regulatory Guide 1.100, Revision 2, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," Washington, DC, June 1988.

(NRC, 1973) U.S. Nuclear Regulatory Commission, Regulatory Guide 3.12, "General Design. Guide for Ventilation Systems of Plutonium Processing and Fuel Fabrication Plants," Washington, DC, August 1973.

(MOX Services, 2009a) Shaw AREVA MOX Services, "Integrated Safety Analysis Summary," Aiken, SC, October 2009.

(MOX Services, 2009b) Shaw AREVA MOX Services, "License Application," Aiken, SC, October 2009.

11.4 Electrical Systems

This section of the safety evaluation report (SER) summarizes the U.S. Nuclear Regulatory Commission (NRC) staff's review and evaluation of the electrical power systems for the Mixed Oxide Fuel Fabrication Facility (MFFF). To conduct this review, the NRC staff evaluated the adequacy of the design and intended operations of these systems, as reflected in the applicant's commitments and goals with respect to that design. Shaw AREVA MOX Services (MOX Services or the applicant) described these commitments in the license application (LA) (MOX, 2010a) and the Integrated Safety Analysis (ISA) Summary (MOX, 2010b) for the Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF).

The purpose of the staff's evaluation is to determine whether the design and intended operations of the MFFF electrical power systems are adequate to ensure that items designated as items relied upon for safety (IROFS) which require electrical power to complete their protective actions will be available and reliable to perform their intended safety function during normal operations, upset conditions, accidents, and natural phenomena events. The NRC staff makes this determination by evaluating the applicant's commitments for completing the design of the MFFF electrical power systems in a manner that addresses specific regulatory acceptance criteria, identified in Section 11.4.1 of this SER. In addition to evaluating the description of the electrical design that is contained in the LA and ISA Summary, the NRC staff conducted "vertical slice" reviews of key accident sequence events described in other sections of the LA and ISA Summary based on risk significance. The NRC staff also reviewed supplementary information the applicant provided, based on RAIs.

The staff performed this review of the MFFF electrical systems design by evaluating the descriptions provided by the applicant in the LA and ISA Summary, along with an evaluation of project design criteria, electrical and site layout drawings, equipment specifications, logic diagrams, procurement documents, and other documents made available to the NRC staff during in-office reviews. In addition to this broad review of the electrical design aspects of the MFFF, the staff performed its review in conjunction with the review of interfacing MFFF instrumentation and control systems and a “vertical slice” review of the expected performance of other facility systems in response to accident sequences described in the applicant’s ISA Summary. In particular, the staff evaluated accident sequences resulting in the release of radiological materials. The purpose of this evaluation was to provide a basis for understanding how the conditions under which the facility dynamic confinement systems (which require the use of continuous electrical power following a release event) will respond.

11.4.1 Regulatory Requirements

The following regulations are applicable to the electrical power systems:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 70.22, “Contents of Applications,” which specifically relates to the requirement that the applicant describe the equipment and facilities and proposed procedures to protect health and minimize danger to life and property

- 10 CFR 70.23, “Requirements for the Approval of Applications,” which specifically relates to the requirement that the Commission determine that the proposed equipment, facilities, and procedures are adequate to protect health and minimize danger to life and property

- 10 CFR 70.61(e), which specifically relates to the requirement that each engineered or administrative control or control system that is needed to meet the performance requirements be designated as an IROFS and that the safety program must ensure that each IROFS will be available and reliable to perform its intended function when needed

- 10 CFR 70.62, “Safety Program and Integrated Safety Analysis,” which specifically relates to the establishment and maintenance of a safety program and to the performance of an ISA

- 10 CFR 70.64, “Requirements for New Facilities or New Processes at Existing Facilities,” which specifically relates to the application of baseline design criteria and defense-in-depth practices to new facilities or new processes at existing facilities

11.4.2 Regulatory Acceptance Criteria

The NRC staff’s evaluation focused on the design bases of the electrical power systems and other related information. The staff reviewed and evaluated the information provided by the applicant for the safety function, system description, and safety analysis for IROFS that require the use of electrical power to perform their safety actions. The review also encompassed the applicant’s adherence to proposed design-basis considerations, such as redundancy, independence, reliability, and quality.

Section 11.4.2.2 of NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility,” issued August 2000 (NRC, 2000), provides the acceptance criteria related to plant systems, including the electrical power systems. These criteria reflect the need to ensure that the baseline design criteria of 10 CFR 70.64 are achieved and that the concept of defense in depth has been applied to the design of the electrical power systems. In addition, the electrical systems’ design and operation should fulfill the functional requirements determined from the ISA, and the electrical systems should be available and reliable to perform their intended safety functions when needed. No code requirements define the specific design criteria that are to be used in the design of the electrical power systems. However, specific design considerations for electrical systems would include the use of a minimum of two physically independent offsite power sources with redundant and independent onsite alternating current (ac) and direct current (dc) power subsystems designed in accordance with the following seven criteria:

(1) provisions so that components of the electrical systems can be tested periodically for operability and required functional performance

(2) electrical and physical separation to ensure that any required independence is maintained

(3) no single-failure vulnerability

(4) sufficient capacity and capability to ensure that the IROFS supported by the electrical systems perform their intended functions

(5) adequate protective relaying and breaker control to ensure required functional performance and adequate response to electrical fault and overload conditions

(6) status monitoring of the behavior of the systems and components that are identified as IROFS

(7) system capability to maintain functionality when subjected to tornadoes, tornado missiles, earthquakes, floods, and any other appropriate severe natural phenomena as established in the ISA

11.4.3 Electrical Power Systems Description

11.4.3.1 Overview of Electrical Power Systems and Their Safety Functions

In Section 11.4 of the LA and Section 4.4 of the ISA Summary, the applicant described the proposed design of the MFFF electrical power supply system. The primary safety function of the electrical power system is to provide a reliable source of ac and dc power for facility IROFS under the full range of conditions expected to be present such that the performance objectives of 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material,” will be met. The proposed design of the electrical power systems for the MFFF consists of several ac and dc subsystems, which together provide a reliable source of power to ensure the continued availability and operation of facility IROFS under normal, abnormal, design-basis accident, and loss of offsite power conditions. The design of the MFFF electrical power supply system addresses the principle of defense-in-depth. The primary facility IROFS requiring such reliable sources of power are the (high depressurization exhaust) HDE, (very high depressurization exhaust) VHD, and process cell depressurization exhaust (POE) fans. These are all needed to ensure that the MFFF heating, ventilation, and air conditioning (HVAC) confinement capability is maintained to allow the facility to meet the performance objectives of 10 CFR Part 70, as well as the controls and control systems supporting the numerous IROFS which must be continually available and reliable to accomplish their intended safety functions under all normal, upset, and emergency conditions. These control systems are provided with a supply of electrical power that has been designated as a system of IROFS. Hence, these control systems are required to be designed, implemented, and maintained in a manner that ensures the availability and reliability of electrical power when needed. In addition, some facility IROFS are designed to place MFFF processes into a safe state upon the loss of electrical power. For these IROFS, reliable non-IROFS power subsystems are provided. These non-IROFS power subsystems are neither designated as quality level (QL)-1 items nor designed to meet the same quality requirements as the IROFS power supplies.

The design of the MFFF electric power supply system incorporates the defense-in-depth concept through the implementation of a normal power system, a standby power system, and an emergency power system. Under normal operating conditions, a reliable (non-IROFS) power system (i.e., the normal power system) provides electrical energy to accomplish all the normal production, safety, and life-safety functions required by the facility. Two separate and independent incoming offsite power feeders supply the ultimate source of electrical power for the normal power system. Facility loads, including the emergency buses that supply power to ensure that IROFS are available to perform facility safety functions, are normally fed from either one of two normally isolated main MFFF buses, each of which is powered from one of these two independent sources of incoming offsite power. In the event that electrical power from one of the two independent sources of incoming offsite power is lost for any reason, the main bus experiencing the loss will be automatically connected, after a brief delay, to the bus being supplied by the remaining source of incoming offsite power. In a response to a request for additional information, the applicant stated that, in the past 35 years, the Savannah River Site, where the MFFF is located, has experienced only one loss of offsite power event, with a duration of approximately 12 hours (MOX, 2009).

In the rare event of a total loss of all incoming power to the facility, a non-IROFS standby power system (the standby power system), with electrical power developed by two independent standby diesel generator subsystems, each sized to carry all IROFS loads, life-safety loads, and loads important to facility production, will automatically start and continue the supply of electrical power to facility loads. Those loads, which cannot tolerate the brief interruption of power that can occur before the standby diesel generator systems are started and loaded, are equipped with batteries or uninterruptible power supplies (UPSs) backed by batteries, as described in further detail below. The design of the standby power system is such that a single standby diesel generator subsystem can supply sufficient power to enable the continued operation of all IROFS loads, life-safety loads, and loads important for facility production for a period of 24 hours without refueling. Although each diesel generator can supply the power needs for the facility for 24 hours, they are also capable of being operated in parallel with one another.

In the very rare event that a total loss of all incoming power occurs and both standby diesel generator systems fail to start and load properly, an independent and redundant Class 1E emergency power system (the emergency power system) has been provided to continue the delivery of electrical power to those IROFS that are needed to ensure that the performance objectives of 10 CFR Part 70 will be met. The emergency power system has been designed to ensure that no single failure can occur that would result in the loss of a facility protective function. The emergency power system consists of two redundant and independent emergency diesel generator systems and Class 1E switchgear and distribution systems, each of which has been sized to carry all IROFS loads for its assigned train for an extended period of time until either the normal or standby power system can be restored. The emergency power system is designed such that either of the two redundant and independent power trains can supply the power needed to accomplish facility safety functions. The emergency power system is qualified to survive the MFFF design-basis earthquake. All emergency power system components are located in QL-1 structures, and all Class 1E circuits and equipment are separated and protected in accordance with nuclear industry codes and design standards, as described below.

[redacted]

[redacted]

The following sections describe the design bases and design features of each of these electrical power subsystems in further detail.

11.4.3.2 Non-IROFS Normal Power System

The normal power system, which is a non-IROFS power supply system, provides the main source of electrical power to service all IROFS under normal operating conditions. The non-IROFS normal power system consists of the normal ac power subsystem and the normal dc power subsystem. Since the normal power supply provides an important role in furnishing electrical power for IROFS loads whenever offsite power is available to the MFFF, the subsystems of the normal power supply are described briefly below.

11.4.3.2.1 Normal Alternating Current Power Subsystem

[redacted]

[redacted]

Provisions are made to facilitate testing and maintenance of the normal ac power subsystem under normal operating conditions.

[redacted]

[redacted]

11.4.3.2.2 Normal Direct Current Power Subsystem

[redacted]

11.4.3.2.3 Normal Direct Current Power Supply Feed for the 208/120-VAC Essential Power Supply System

[redacted]

[redacted]

11.4.3.2.4 Facility Grounding System

[redacted]

11.4.3.3 Non-IROFS Standby Power System

[redacted]

[redacted]

Non-IROFS integrated standby diesel generator PLCs control the starting, synchronizing, and connection of the standby diesel generators to the paralleling bus operations. The normal utility control system PLC sheds noncritical loads from the 4.16-kV normal power switchgear buses and then automatically sequences the application of loads to the standby power system in two sequence groups. Load group 1 (highest priority) includes the emergency distribution systems, including the IROFS ventilation confinement system loads and other IROFS requiring electrical power to perform their required safety functions. Load group 2 includes nonemergency loads, including certain normal process loads and non-IROFS ventilation system loads. In the event that only one standby diesel generator is available, the normal utility control system PLC connects only load group 1 to the 4.16-kV normal power system switchgear buses.

11.4.3.4 Emergency Power System (IROFS)

[redacted]

11.4.3.4.1 Emergency Alternating Current Power System

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11.4.4 Design Bases for Electrical Power Systems and Applicable Baseline Design Criteria

The electrical power systems designated as IROFS are required to be available and to provide reliable electrical power to MFFF IROFS for normal operations, to provide safe-shutdown capability, and to support monitoring during and following credible external events. The electrical systems and equipment required to accomplish these functions must remain operational when subjected to natural phenomena hazards (10 CFR 70.64(a)(2)). They must be adequately protected from fires and explosions (10 CFR 70.64(a)(3)), and they must provide for adequate protection from environmental and dynamic effects associated with normal operations, maintenance, testing, and postulated accidents that could lead to a loss of safety functions (10 CFR 70.64(a)(4)). The design must provide for continued operation of essential utility services (10 CFR 70.64(a)(7)) and adequate inspection, testing, and maintenance to ensure their availability to perform their function when needed (10 CFR 70.64(a)(8)). The design must provide for inclusion of instrumentation and control systems to monitor and control the behavior of IROFS (10 CFR 70.64(a)(10)). Finally, the system design must be based on defense-in-depth practices and incorporate, to the extent practicable, a preference for selection of engineered controls over administrative controls to increase overall system reliability (10 CFR 70.64(b)).

To ensure that the design-basis requirements and applicable baseline design criteria are met, the applicant has committed to complete the design, construction, startup testing, maintenance, and periodic functional and operability testing in accordance with specific industry standards and NRC guidance documents as described in the following sections.

11.4.4.1 Design Criteria Applied to the Emergency Alternating Current and Direct Current Power Systems

As described in the applicant's LA and ISA Summary, the emergency ac and dc power systems are designed to provide reliable power to redundant IROFS to enable the MFFF to meet its performance objectives in the event of loss of offsite power. As described above, the emergency ac power buses within the MFFF receive normal power from two independent sources of offsite power generation. In the event that a loss of offsite power is experienced from one feed, the alternate source of power is made available to both emergency ac power buses. In the event that both sources of offsite power are lost, the standby diesel generators will automatically start and provide power to the emergency ac power buses. The emergency ac and dc power system designs utilize redundant, independent, physically separated, seismically qualified trains of power equipment, with adequate capacity, capability, and protective relaying to ensure the performance of its safety functions and maintain qualification for natural phenomena and environmental and dynamic effects. All emergency ac and dc power system components are designated as Class 1E. The equipment is protected by being located in structures that are designed to withstand tornados, earthquakes, and other natural external hazards. The equipment is located within areas of the MFFF where the expected environmental variations are within the normal design capabilities of the equipment. In the few instances in which any emergency ac and dc power system equipment or components are expected to be operated outside of normal manufacturer design environmental ranges, the equipment will be qualified for the expected environment. Furthermore, cables carrying power and control signals associated with the emergency ac and dc power system will be supported by raceways and cable trays that are seismically supported. Cables exposed to building areas in cable trays are designed to be flame retardant. Power cables to IROFS are routed to them in enclosed conduit to minimize the likelihood of interaction between divisional cables and between divisional cables and non-divisional cables. Where these electrical conductors cannot be separated to the extent identified within the MFFF electrical design criteria (e.g., where they enter small gloveboxes), redundant cables are separated from one another to the maximum extent practicable.

To implement the design criteria described above, the emergency ac and dc power systems have been designed using guidance from the following industry standards:

- Overall AC and DC Power System Design

  IEEE Standard 308-1991 (IEEE, 1991a)

- Overall DC Power Systems Design

  IEEE Std 946-1992, “IEEE Recommended Practice for the Design of Safety Related DC Auxiliary Power Systems for Nuclear Power Plants” (IEEE, 1992b)

- Equipment Seismic Qualification

  IEEE Std 344-1987, “IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Generating Stations” (IEEE, 1987a)

- Equipment Environmental Qualification:

  IEEE Std 323-1983, “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations” (IEEE, 1983)

- Preoperational and Periodic Surveillance Testing

  IEEE Std 308-1991 (IEEE, 1991a)

  IEEE Std 387-1995, “IEEE Standard Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations” (IEEE, 1995a)

  IEEE Std 338-1992, “IEEE Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems” (IEEE, 1992c)

  Regulatory Guide (RG) 1.118, Revision 3, “Periodic Testing of Electric Power and Protection Systems” (NRC, 1995)

- Single-Failure Design

  IEEE Std 379-1994, “IEEE Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems” (IEEE, 1994a)

  IEEE Std 603-1998, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations” (IEEE, 1998)

- Qualification and Fire Protection of Cables Installed in Open Cable Trays

  IEEE Std 383-1992, “IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices and Connections for Nuclear Power Generating Stations” (IEEE, 1992d)

- Protection of Emergency Power Systems Equipment from Explosions Resulting from

Hydrogen Accumulation

National Fire Protection Association (NFPA, 1996) Standard 111-1996, "Standard on Stored Electrical Energy Emergency and Standby Power Systems" (NFPA, 1996)

IEEE Std 484-1996, "IEEE Recommended Practice for Installation Design and Installation of Vented Lead-Acid Batteries for Stationary Applications" (IEEE, 1996a)

- Electrical Independence and Separation

  IEEE Std 384-1992 (IEEE, 1992a), except where circuit breakers and fuses are used as isolation devices, in which case, two will be placed in series

- Electrical Equipment Protection

  IEEE Std 741-1997, "IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations" (IEEE, 1997a)

  IEEE Std 242-1986, "IEEE Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems" (IEEE, 1986a)

- Design, Capacity Sizing, Installation, Testing, and Maintenance of Lead-Acid Batteries

  IEEE Std 484-1996 (IEEE, 1996a)

  IEEE Std 485-1997, "IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications" (IEEE, 1997b)

  IEEE Std 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications" (IEEE, 1995b)

  IEEE Std 1184-1994, "IEEE Guide for the Selection and Sizing of Batteries for Uninterruptible Power Systems" (IEEE, 1994b)

- Design and Installation of Class 1E Transformers

  IEEE Std 638-1992, "IEEE Standard for Qualification of Class 1E Transformers for Nuclear Power Generating Stations" (IEEE, 1992e)

- Design and Installation of Cable Systems and Class 1E Raceway Systems

  IEEE Std 690-1984, "IEEE Standard for Design and Installation of Cable Systems for Nuclear Power Generating Stations" (IEEE, 1984)

  IEEE Std 628-1987, "IEEE Standard Criteria for the Design, Installation, and Qualification of Raceway Systems for Class 1E Circuits for Nuclear Power Generating Stations" (IEEE, 1987b)

- Design and Installation of Battery Chargers, Inverters, and Uninterruptible Power Supplies

IEEE Std 650-1990, “IEEE Standard for Qualification of Class 1E Static Battery Chargers and Inverters for Nuclear Power Generating Stations” (IEEE, 1990)

IEEE Std 944-1986, “IEEE Recommended Practice for the Application and Testing of Uninterruptible Power Supplies for Power Generating Stations” (IEEE, 1986b)

- Design, Installation, Testing, and Maintenance of Diesel Generator Systems and Diesel Generator Fuel Oil Systems

IEEE Std 387-1995 (IEEE, 1995a)

IEEE Std 446-1995, “IEEE Recommended Practice for Emergency and Standby Power Systems for Industrial and Commercial Applications” (IEEE, 1995c)

American National Standards Institute/American Nuclear Society (ANSI/ANS) 59.51-1997, “Fuel Oil Systems for Safety-Related Emergency Diesel Generators” (ANSI/ANS, 1997)

NFPA 30-1996, “Flammable and Combustible Liquids Code” (NFPA, 1996)

NFPA 37-1998, “Standards for the Installation and Use of Stationary Combustion Engines and Gas Turbines” (NFPA, 1998)

- Design, Installation, Measurement, and Testing of Grounding Systems

IEEE Std 142-1991, “Recommended Practice for Grounding of Industrial and Commercial Power Systems” (IEEE, 1991b)

NFPA 70-1999, “National Electrical Code” (NFPA, 1999)

IEEE Std 80-1986, “IEEE Guide for Safety in Substations Grounding” (IEEE, 1986c)

IEEE Std 81.2-1991, “IEEE Guide for Measurement of Impedance and Safety Characteristics of Large, Extended, or Interconnected Grounding Systems” (IEEE, 1991c)

IEEE Std 665-1995, “Guide for Generating Station Grounding” (IEEE, 1995d)

IEEE Std 1050-1996, “Guide for Instrumentation and Control Equipment Grounding in Generating Stations” (IEEE, 1996b)

IEEE Std 1100-1992, “Recommended Practice for Powering and Grounding Sensitive Electronic Equipment” (IEEE, 1992f)

11.4.4.2 Preference for Automatic Engineered Controls

The applicant has stated that the design of the emergency ac and dc power systems makes use of automatic, hardwired controls to detect loss of offsite power conditions, start standby or emergency diesel generators, and shed or connect (or both) loads to the buses, where possible. These measures will ensure timely restoration and delivery of power when needed by facility

IROFS, indicating that the emergency ac and dc power systems have been designed with a preference for automatic engineered controls over administrative controls (10 CFR 70.64(b)).

11.4.4.3 Adherence to Defense-in-Depth Practices

The design of the electrical systems at the MFFF makes use of the practice of defense-in-depth to a significant degree (see footnote to 10 CFR 70.64). First, the emergency ac and dc power systems make use of independent, redundant, and physically separated trains of equipment, such that no single credible failure can occur within one train that would render both trains inoperable, thereby preventing the MFFF from meeting the performance objectives of 10 CFR Part 70. Further, the emergency ac and dc power systems are designed such that there are no fewer than four sources of normal or standby electrical power—two independent alternate sources of offsite power and two standby diesel generators—each of which has sufficient capacity to power the emergency power supply buses for an extended period. In the event that all of these sources of power are unavailable, each train of the emergency ac power system is equipped with its own Class 1E emergency diesel generator that will enable the safety functions of the MFFF IROFS to complete their protective actions. The provision of a minimum of two, independent offsite power sources for the MFFF is consistent with the requirements of IEEE 765-1995, “IEEE Standard for Preferred Power Supply for Nuclear Power Generating Stations,” (IEEE, 1995) which requires the use of two separate and independent sources of power. The standby power system diesel generator systems are designed to meet NFPA 110, “Standard for Emergency and Standby Power Systems” (NFPA, 1999b) and IEEE 446-1995 (IEEE, 1995c).

11.4.5 External Manmade Hazard Event Sequences

Section 5.3.9 of the ISA Summary discusses the results of the applicant’s evaluation of external manmade hazard (EMMH) events. The following EMMH event group is related to the electrical design: EMMH-3, “Loss of Offsite Power.” The Savannah River Site electrical transmission/distribution system is supplied by multiple offsite feeds. The location of these feeds provides a reliable power supply in the event of manmade or natural phenomena hazards. The applicant postulated that EMMHs caused an accidental loss of offsite power for the MFFF site. A loss of offsite power was postulated to occur from a number of possible events, including power generation problems and transportation accidents. This event includes loss of feed from the Savannah River Site. The safety strategy for this event is to mitigate the consequences from a loss of offsite power for the MFFF site by providing emergency power, and the IROFS for this event is the emergency power system.

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With respect to the discussions in Section 11.4.4, the staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61, “Performance Requirements,” in the event of a loss of offsite power.

11.4.6 Evaluation Findings

The NRC staff has evaluated the information provided by the applicant for electrical systems in the LA and the ISA Summary. For the reasons outlined in Section 11.4.4 of this SER, the staff concludes that the baseline design criteria of 10 CFR 70.64 have been achieved and that the concept of defense-in-depth has been applied to the design of the electrical power systems. In addition, there is reasonable assurance that the electrical systems’ design and operation will fulfill the functional requirements of providing reliable power to enable the MFFF IROFS to perform their required safety actions. In addition, the applicant provided reasonable assurance that the electrical systems will be available and reliable to perform their intended safety functions when needed. Further, the design of the MFFF incorporates the specific design considerations outlined in the regulatory acceptance criteria in Section 11.4.3 of this SER. These criteria include the use of a minimum of two physically independent offsite power sources with redundant and independent onsite ac and dc power subsystems designed in accordance with the following seven criteria as listed in Section 11.4.2.2 of NUREG-1718.:

(1) provisions so that components of the electrical systems can be tested periodically for operability and required functional performance

(2) electrical and physical separation to ensure that any required independence is maintained

(3) no single-failure vulnerability

(4) sufficient capacity and capability to ensure that IROFS supported by the electrical systems perform their intended functions

(5) adequate protective relaying and breaker control to ensure required functional performance and adequate response to electrical fault and overload conditions

(6) status monitoring of the behavior of the systems and components that are identified as IROFS

(7) system capability to maintain functionality when subjected to tornadoes, tornado missiles, earthquakes, floods, and any other appropriate severe natural phenomena as established in the ISA

The staff concludes that, with the proposed IROFS, the adherence to designated codes and industry standards, the application of management measures, and application of the MOX project Quality Assurance Plan (MPQAP), the baseline design requirements for the applicant’s proposed electrical systems, equipment, controls, and procedures have been achieved. The staff further concludes that there is reasonable assurance that the electrical power systems will be sufficiently reliable and available to enable the MFFF IROFS to perform their required safety functions when needed.

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(MOX, 2010b) Shaw AREVA MOX Services, "MFFF—Integrated Safety Analysis Summary," Aiken, SC, March 2010.

(MOX, 2009) Stinson, D., Shaw AREVA MOX Services, Letter to U.S. Nuclear Regulatory Commission, RE: Shaw AREVA MOX Services Response to a Request for Additional Information (DCS-NRC-000251), October 5, 2009.

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(NFPA, 1998) National Fire Protection Association, NFPA 37-1998, "Standards for the Installation and Use of Stationary Combustion Engines and Gas Turbines," Quincy, MA, 1998.

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(NRC, 1995) U.S. Nuclear Regulatory Commission, RG 1.118, Rev. 3, "Periodic Testing of Electric Power and Protection Systems," April 1995.

11.5 Instrumentation and Control Systems

This section of the safety evaluation report (SER) summarizes the U.S. Nuclear Regulatory Commission (NRC) staff's review and evaluation of the instrumentation and control (I&C) systems for the Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF). The objective of this review is to determine whether the aspects of the design of the I&C systems that are relied on for safety, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 70, "Domestic Licensing of Special Nuclear Material," and encompassed by the hazard and accident analyses of the integrated safety analysis (ISA) will be available and reliable to perform their intended function when needed. To conduct this review, the NRC staff evaluated the adequacy of the design and intended operations of these systems, as reflected in the commitments and goals with respect to that design made by Shaw AREVA MOX Services (MOX Services or the applicant). The applicant described these commitments and goals in the license application (LA) (MOX, 2010a) and the ISA Summary (MOX, 2010b).

The purpose of the staff's evaluation is to determine whether the design and intended operations of the I&C systems for the MFFF are adequate to ensure that I&C items designated as items relied upon for safety (IROFS) will be available and reliable to perform their intended safety function during normal operations, upset conditions, accidents, and natural phenomena events. The NRC staff makes this determination by evaluating the applicant's commitments in the LA for completing the design of the MFFF I&C systems in a manner that addresses the regulatory acceptance criteria, identified in Section 11.5.1 of this SER below. In addition to evaluating the description of the I&C design that is contained in the LA and ISA Summary, the NRC staff conducted "vertical slice" reviews of key accident sequence events described in other sections of the LA and ISA Summary, as well as reviews of supplementary information provided by the applicant.

The staff performed the review of the MFFF I&C systems design by evaluating the descriptions provided by the applicant in the LA and ISA Summary, along with an evaluation of project design criteria, piping and instrument diagrams, electrical/I&C schematic diagrams, logic diagrams, nuclear safety evaluations, nuclear criticality safety evaluations, system descriptions, and other documents made available to the NRC staff during in-office reviews. In addition to this broad review of the I&C design aspects of the MFFF, the staff performed its review in conjunction with the review of interfacing MFFF electrical power systems and a detailed review of the expected performance of selected higher risk facility systems in response to accident sequences described in the applicant's ISA Summary. In particular, the staff evaluated accident sequences resulting in the release of radiological materials within gloveboxes and process cells to provide a basis for understanding the conditions under which I&C IROFS performing or supporting safety actions of the facility dynamic confinement systems will be required to respond. The staff also assessed the adequacy of the applicant's process for the development life cycle of software for MFFF digital control systems used as IROFS to determine whether there is reasonable assurance that such IROFS will be available and reliable when needed by minimizing the potential for common-cause software errors.

11.5.1 Regulatory Requirements

The following regulations are applicable to the I&C systems:

- 10 CFR 70.22, "Contents of Applications," which specifically relates to the requirement that the applicant describe the equipment and facilities and proposed procedures to protect health and minimize danger to life and property

- 10 CFR 70.23, “Requirements for Approval of Applications,” which specifically relates to the requirement that the Commission determine that the proposed equipment, facilities, and procedures are adequate to protect health and minimize danger to life and property

- 10 CFR 70.61(e), “Performance Requirements”, which specifically relates to the requirement that each engineered or administrative control or control system that is needed to meet the performance requirements be designated as an IROFS and that the safety program must ensure each IROFS will be available and reliable to perform its intended function when needed

- 10 CFR 70.62, “Safety Program and Integrated Safety Analysis,” which specifically relates to the establishment and maintenance of a safety program and to the performance of an ISA

- 10 CFR 70.64, “Requirements for New Facilities or New Processes at Existing Facilities,” which specifically relates to the application of baseline design criteria and defense-in-depth practices to new facilities or new processes at existing facilities

11.5.2 Regulatory Acceptance Criteria

The NRC staff’s evaluation focused on the design bases of the I&C systems and other related information. The staff reviewed and evaluated the information provided by the applicant for the safety function, system description, and safety analysis for IROFS that require use of the I&C systems and equipment to perform their safety actions. The review also encompassed the applicant’s adherence to proposed design-basis considerations, such as redundancy, independence, reliability, and quality.

Section 11.4.3.2 of NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility,” issued August 2000 (NRC, 2000a), provides the acceptance criteria related to plant systems, including the I&C systems. These criteria reflect the need to ensure that the baseline design criteria of 10 CFR 70.64 are achieved and that the concept of defense in depth has been applied to the design of the I&C systems. In addition, the I&C systems’ design and operation should fulfill the functional requirements determined from the ISA, and the I&C systems should be available and reliable to perform their intended safety functions when needed. No code requirements define the specific design criteria that are to be used in the design of the I&C systems. However, specific design considerations for I&C systems include the use of redundant and diverse safety instrument channels with coincident logic providing automatic actuation with additional manual operation capability. The I&C safety systems should be designed in accordance with the following seven criteria from NUREG-1718, Section 11.4.3.2.:

(1) provisions so that I&C system components can be tested periodically for operability and required functional performance

(2) electrical, physical, and control/separation to ensure that any required redundancy and independence are maintained

(3) no single-failure vulnerability

(4) adequate instrument spans, setpoints, and control ranges to ensure proper monitoring and control of IROFS

(5) provisions so that I&C system components fail in a safe failure mode

(6) status monitoring of the behavior of the systems and components that are identified as IROFS

(7) system capability to maintain functionality when subjected to tornadoes, tornado missiles, earthquakes, floods, and any other appropriate severe natural phenomena as established in the ISA

11.5.3 Instrumentation and Control Systems Description

11.5.3.1 Overview of Instrumentation and Control Systems and Their Safety Functions

In Section 11.5 of the LA and Section 4.5 of the ISA Summary, the applicant described the proposed design of the major MFFF I&C systems and devices in detail. The MFFF I&C design includes both safety and non-safety applications of I&C. The safety functions of the I&C systems and equipment serving as IROFS are designed to reduce the risk of high and medium consequence events and to limit the risk of nuclear criticality under all normal and credible abnormal conditions. Some of these safety functions are designed to prevent the occurrence of accident sequences within the MFFF process units, while others are designed to ensure the continued availability of electrical and confinement heating, ventilation, and air conditioning (HVAC) systems needed to limit the release of radioactive materials to the environment. The control systems for these electrical and HVAC confinement processes are designed using defense-in-depth principles, as described below.

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11.5.3.2 Normal Process Control Systems (Non-IROFS)

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11.5.3.2.2 Control Rooms and Control Areas

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11.5.3.2.3 Personnel and Equipment Protection Controls

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11.5.3.3 Utility Control Systems (Non-IROFS)

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11.5.3.4 Emergency Control Systems and the Seismic Monitoring and Trip System (IROFS)

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11.5.3.5 Safety Control System (IROFS)

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The safety controls are treated as Class 1E equipment and are designed with redundancy, independence, separation, capabilities for periodic testability, provision of system status information, control of access, reliability, appropriate equipment qualification, and continued maintenance of the facility within acceptable limits in the presence of a common-cause software failure. The safety controllers have been designed to address the applicable guidance in Regulatory Guide (RG) 1.153, Revision 1, "Criteria for Safety Systems," and the applicable portions of IEEE Std 603-1998, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations" (IEEE, 1998a), and IEEE Std 7-4.3.2-1993, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations" (IEEE, 1993).

To ensure that the software for the redundant safety controllers has been developed using high quality systems development processes consistent with the quality assurance commitments for the facility and to minimize the potential for a common-cause software failure simultaneously rendering both safety controllers inoperable, the applicant has committed to develop software for these systems following the guidance contained in RG 1.168, "Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants" (NRC, 1997a), with clarifications regarding the applicability of IEEE Std 1028-1997, "IEEE Standard for Software Reviews" (IEEE, 1997a); RG 1.169, "Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants" (NRC, 1997b); RG 1.170, "Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants" (NRC, 1997h); RG 1.171, "Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants" (NRC 1997i); RG 1.172, "Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants" (NRC, 1997c), including the exceptions noted with respect to IEEE Std 830-1998 (IEEE, 1998b); and RG 1.173, "Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants" (NRC, 1997d)

11.5.4 Design Bases for Instrumentation and Control Systems and Applicable Baseline Design Criteria

The I&C systems designated as IROFS are required to be available and reliable for all normal operations and upset conditions, provide for safe-shutdown capability, and support monitoring during and following identified event sequences and credible external events. The I&C systems and equipment required to accomplish these functions must remain operational when subjected to natural phenomena hazards (10 CFR 70.64(a)(2)). They must be adequately protected from fires and explosions (10 CFR 70.64(a)(3)). They must provide for adequate protection from environmental and dynamic effects associated with normal operations, maintenance, testing, and postulated accidents that could lead to a loss of safety functions (10 CFR 70.64(a)(4)). The I&C design must provide for the support of continued operation of essential utility services (10 CFR 70.64(a)(7)). The design must provide for adequate inspection, testing, and maintenance of the I&C systems to ensure their availability to perform their function when needed (10 CFR 70.64(a)(8)). The design must also provide for inclusion of I&C systems to monitor and control the behavior of IROFS (10 CFR 70.64(a)(10)). Finally, the system design must be based on defense-in-depth practices and incorporate, to the extent practicable, a preference for selection of engineered controls over administrative controls to increase overall system reliability (10 CFR 70.64(b)).

To ensure that the design-basis requirements and applicable baseline design criteria are met, the applicant has committed in the LA to complete the design, safety requirements specification, software development, validation and verification testing, startup testing, maintenance, and periodic functional and operability testing of the I&C systems in accordance with specific industry standards and NRC guidance documents as described in the sections below.

11.5.4.1 Design Criteria Applied to the MFFF Instrumentation and Control Systems

As described in the applicant’s LA and ISA Summary, the MFFF I&C systems are designed to be available and reliable to support normal, upset, and accident conditions to allow the MFFF to meet its performance objectives while sustaining credible single active failures. The equipment is protected by its location in structures designed to withstand tornados, earthquakes, and other natural external hazards. As described above, the I&C systems and equipment serving as IROFS are designated as Class 1E and are composed of QL-1 components. The emergency control system is composed of hardwired equipment dedicated for use with the electrical equipment and HVAC systems needed to support the radiological material confinement function. The emergency control system is designed with two redundant and independent systems in a train A/train B arrangement, capable of functioning during and following dynamic effects resulting from the occurrence of a design-basis earthquake or a postulated accident sequence.

The applicant has committed to using a software development process for the Class 1E, redundant digital safety PLCs that minimizes the likelihood of common-mode failures rendering both redundant PLCs simultaneously inoperable. The applicant will complete the software development process in accordance with the recommendations and guidance of a set of generally accepted industry standards which promote high functional reliability and design quality in software used in safety systems, as well as the RGs that describe acceptable methods for designing such systems using these standards.

The equipment is located within areas of the MFFF where the expected environmental variations are within the normal design capabilities of the equipment. Notwithstanding, the applicant has committed to qualify all equipment with the potential for exposure to elevated environmental conditions in accordance with applicable nuclear industry standards for equipment environmental qualification. Cables carrying instrument power and control signals will be installed in conduit that is seismically supported. Cables exposed to building areas in cable trays are designed to be flame retardant. Cables for instrument sensor and actuation devices associated with IROFS are routed in enclosed conduit to minimize the likelihood of interaction between divisional cables and between divisional cables and nondivisional cables. Where these electrical conductors cannot be separated to the extent identified within the MFFF electrical design criteria (e.g., where they enter small gloveboxes), redundant cables are separated from one another to the maximum extent practicable.

To implement the design criteria described above, the applicant has committed to complete the design of the MFFF I&C systems in accordance with the recommendations and guidance from the following nuclear industry standards and NRC guidance:

- Overall I&C Safety System Design (Including the Seismic Monitoring and Trip System)

  IEEE Std 603-1998 (IEEE, 1998a)
  RG 1.153, Revision 1 (NRC, 1991)

- Single Failure Design

  IEEE Std 379-1994, “IEEE Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems” (IEEE, 1994a), supplemented by the following:

  – Branch Technical Position HICB-17, “Guidance on Self-Test and Surveillance Test Provisions” (NRC, 1997e)

  – IEEE Std 603-1998 (IEEE, 1998a)

  – RG 1.153, Revision 1 (NRC, 1991)

- Software Programmable Electronic Systems

  IEEE Std 7-4.3.2-1993 (IEEE, 1993)

  IEEE Std 730-1998, “Software Quality Assurance Plans” (IEEE,1998c)

  IEEE Std 828-1998, “IEEE Standard for Software Configuration Management Plans” (IEEE, 1998d)

  IEEE Std 830-1998 (IEEE, 1998b)

  IEEE Std 1012-1998, “IEEE Standard for the Software Verification and Validation” (IEEE, 1998e)

  IEEE Std 1028-1997, “IEEE Standard for Software Reviews” (IEEE, 1997a)

  IEEE Guide 1042-1987, “Software Configuration Management” (IEEE, 1987a)

  IEEE Std 1074-1997, “IEEE Standard for Developing Software Life Cycle Processes” (IEEE, 1997b)

  IEEE Std 1228-1994, “IEEE Standard for Software Safety Plans” (IEEE, 1994b)

RG 1.152, Revision 1, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants" (NRC, 1996b)

RG 1.153, Revision 1 (NRC, 1991)

RG 1.168 (NRC, 1997a)

RG 1.169 (NRC, 1997b)

RG 1.170 (NRC, 1997h)

RG 1.171 (NRC, 1997i)

RG 1.172 (NRC, 1997c)

RG 1.173 (NRC, 1997d)

Electric Power Research Institute, TR-106439, "Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications" (EPRI, 1996)

NRC Safety Evaluation, "EPRI Topical Report TR-106439" (NRC, 1997f)

International Electrotechnical Commission, 61131-3 (1993-03), "Programmable Controllers—Part 3: Programming Languages" (IEC, 1993)

NUREG/CR-6090, "The Programmable Logic Controller and Its Application in Nuclear Reactor Systems" (NRC, 1993)

NUREG/CR-6463, "Review Guidelines for Software Languages for Use in Nuclear Power Plant Safety Systems: Final Report" (NRC, 1996a)

- Electrical Independence, Separation, and Qualification of Isolation Devices

IEEE Std 384-1992, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits (IEEE, 1992a)

NUREG-0800, Standard Review Plan, Branch Technical Position HICB-11, "Guidance on the Application and Qualification of Isolation Devices" (NRC, 1997g)

RG 1.75, Revision 2, "Physical Independence of Electric Systems" (NRC, 1978)

- Seismic Qualification of Equipment

IEEE Std 344-1987, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Generating Stations" (IEEE, 1987b)

RG 1.100, Revision 2, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants" (NRC, 1988)

- Environmental Qualification of Safety Equipment

IEEE Std 323-1983, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations" (IEEE, 1983)

- Establishment of Process Instrument Setpoints

  ANSI/ISA-67.04.01-2000, "Setpoints for Nuclear Safety Related Instrumentation" (ANSI/ISA, 2000)

  RG 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation" (NRC, 1999)

- Evaluation of Human-System Interfaces

  IEEE Std 1023-1988, "IEEE Guide for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations" (IEEE, 1988)

  NUREG-0700, Revision 2, "Human System Design Review Guidelines" (NRC, 2002)

- Design of the Seismic Monitoring and Trip System (Recording Function

  RG 3.17-1974, "Earthquake Instrumentation for Fuel Reprocessing Plants" (NRC, 1974)

- Periodic Testing

  IEEE Std 338-1992, "IEEE Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems" (IEEE, 1992b)

  NUREG-0800, Standard Review Plan, Branch Technical Position HICB-17 (NRC, 1997e)

  RG 1.118, Revision 3, "Periodic Testing of Electric Power and Protection Systems" (NRC, 1995)

- Reduction of Electromagnetic and Radiofrequency Interference and Proper Grounding

  IEEE Std 518-1982, "IEEE Guide for the Installation of Electrical Equipment to Minimize Electrical Noise Inputs to Controllers from External Sources" (IEEE, 1982)

  IEEE Std 1050-1996, "Guide for Instrumentation and Control Equipment Grounding in Generating Stations" (IEEE, 1996)

  RG 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems" (NRC, 2000b)

- Design of Data Communications Networks

  ANSI/IEEE 802.3 Standards Series, "IEEE Standards for Local Area Networks: Carrier Sense Multiple Access with Collision Detection (CSMA/CD) Access Method and Physical Layer Specifications" (ANSI/IEEE, 2000)

- Evaluation of Commercial-Grade Equipment Dedicated for Use in Safety Applications

  NUREG/CR-6421, “Guideline on the Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications” (NRC, 1996)

11.5.4.2 Preference for Automatic Engineered Controls

The applicant has described its plans for the automatic actuation of safety functions by the facility’s safety I&C digital control systems and the hardwired emergency control system, which serve as IROFS to support the safety actions of the facility. The applicant also indicated that the system has been designed with a preference for automatic engineered controls over administrative controls (see 10 CFR 70.64(b)).

11.5.4.3 Adherence to Defense-in-Depth Practices

The design of the I&C systems at the MFFF makes use of the practice of defense in depth to a significant degree (see footnote to 10 CFR 70.64). The safety control system and the emergency control system make use of independent, redundant, and physically separated trains of equipment in a manner such that no single credible failure can occur within one train that would render both trains inoperable, thereby preventing the MFFF from meeting its licensed performance objectives.

11.5.5 Evaluation Findings

The NRC staff has evaluated the information provided by the applicant for I&C systems in the LA and the ISA Summary. The staff has determined that the design guidance and recommendations contained in the RGs, industry codes and standards, and licensing review guidance documents to which the applicant has committed to use in completing the design of the MFFF will provide reasonable assurance that the design criteria identified in the Section 11.5.1 of this SER will be adequately addressed. For the reasons outlined above, the staff concludes that the baseline design criteria of 10 CFR 70.64 have been achieved and that the concept of defense in depth has been applied to the design of these systems. In addition, there is reasonable assurance that the I&C systems design and operation will be available and reliable to enable the MFFF IROFS to perform their required safety actions when needed. Further, the I&C systems design has incorporated the specific design considerations outlined in the regulatory acceptance criteria in Section 11.5.3 of this SER. These considerations include the use of redundant or diverse safety instrument channels, or both, with coincident logic providing automatic actuation with additional manual operation capability. The I&C safety systems are designed in accordance with the following seven criteria as specified in Section 11.4.3.2 of NUREG-1718:

(1) provisions so that I&C system components can be tested periodically for operability and required functional performance

(2) electrical, physical, and control/separation to ensure that any required redundancy and independence are maintained

(3) no single failure vulnerability

(4) adequate instrument spans, setpoints, and control ranges to ensure proper monitoring and control of IROFS

(5) provisions so that I&C system components fail in a safe failure mode

(6) status monitoring of the behavior of the systems and components that are identified as IROFS

(7) system capability to maintain functionality when subjected to tornadoes, tornado missiles, earthquakes, floods, and any other appropriate severe natural phenomena as established in the ISA

As outlined in the discussions above, the staff concludes that, with adherence to designated codes and industry standards, the application of management measures, and the application of the MFFF quality assurance program, the baseline design requirements for the applicant's proposed I&C systems and procedures have been achieved. The staff further concludes that there is reasonable assurance that the I&C systems will be sufficiently reliable and available to enable the MFFF IROFS to perform their required safety functions when needed.

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10 CFR Part 70, Domestic Licensing of Special Nuclear Material.

11.6 Material-Handling Systems

This section of the safety evaluation report (SER) contains the staff's review of the design and operation of material-handling systems, as described in Section 11.6 of the license application (LA) provided by Shaw AREVA MOX Services (the applicant) (MOX, 2010a) and the corresponding sections of the Integrated Safety Analysis (ISA) Summary (MOX, 2010b). The primary purpose of this review is to determine whether the proposed material-handling equipment, including items designated as items relied on for safety (IROFS), will be available and reliable to perform their intended safety function during normal operations, upset conditions, accidents, and natural phenomena events. This review evaluated whether the applicant provided reasonable assurance that workers, the public, and the environment will be protected from the radiological consequences of an accident in accordance with the applicable regulations.

For this review, the staff evaluated the information provided by the applicant for material-handling equipment and controls in Section 11.6 and other applicable sections of the LA. The review of the design and operation of the material-handling systems was closely coordinated with the review of applicable portions of Chapters 4 and 5 of the ISA Summary, which discusses the material-handling operations and potential load-handling events.

11.6.1 Regulatory Requirements

The following regulations are applicable to material-handling equipment and controls:

- 10 CFR 70.61(e), specifically relating to the requirement that each engineered or administrative control or control system that is needed to meet the performance requirements be designated as an item relied on for safety and relating to the safety program that ensures each item relied on for safety will be available and reliable to perform its intended function when needed

- 10 CFR 70.64, "Requirements for New Facilities or New Processes at Existing Facilities," specifically relating to the application of baseline design criteria and defense-in-depth practices to new facilities or new processes at existing facilities

11.6.2 Regulatory Acceptance Criteria

The review focused on the design bases of material-handling components and other related information. For material-handling systems, the staff reviewed and evaluated the information provided by the applicant for the safety function, system description, and safety analysis. The review also encompassed proposed design-basis considerations, such as redundancy, independence, reliability, and quality.

Section 11.4 of NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility," issued August 2000 (NRC, 2000), identifies the acceptance criteria related to plant systems. No section in NUREG-1718 specifically addresses Section 11.6 in the LA and Chapter 5.3.5 in the ISA Summary. However, the function, description, control concepts, and system interfaces pertaining to each material-handling system are addressed in the staff review guidance in Section 11.4 of NUREG-1718.

11.6.3 Material-Handling Equipment Description

Section 11.1, "MOX Process Description," of the LA describes the individual process units and associated equipment. Section 1.1.2 of the LA describes the general methodology and requirements for determining the seismic response of process equipment, systems, and components. Section 1.1.2 also provides the general design bases, functional requirements, and acceptance criteria for qualifying the process material-handling equipment under all conditions, including normal operation, credible accidents, and design-basis natural phenomena events in accordance with 10 CFR 70.61, "Performance Requirements."

Material-handling equipment classified as QL-1 is designed and qualified to perform one or more of the following functions: (1) prevent a criticality by maintaining the structural integrity of material-handling equipment relied upon to control the geometry and configuration of fissile material or (2) protect QL-1 structures, systems, and components (SSCs) from physical interaction as a result of a seismic or material handling.

The functions assigned to material-handling system elements include the following:

- Transfer mixed oxide (MOX) fuel material and components from one point in the process to another in accordance with process throughput, positioning tolerance, mechanism reliability, and radiological shielding requirements.

- Prevent component impact or overtravel that could potentially damage a container.

- Maintain structural integrity and control of process containers to ensure that the confinement boundary is not breached.

- Maintain structural integrity and control of process containers to ensure that criticality control functions are performed.

- Limit the applied gripping force where excessive force could potentially damage a container.

- Detect the successful receipt or release of a container by measuring weight or detecting the physical presence of a container.

- Detect the successful alignment of the container and handling device before engaging the gripping device.

- Work with fire barriers to transfer material across process atmosphere or fire area boundaries, as necessary.

- Transfer tooling and equipment spare parts from point to point inside the glovebox system during maintenance operations.

The following sections describe the major equipment associated with material-handling equipment.

Powder-Handling Equipment

Fuel production powder materials handled in the aqueous polishing (AP) and MOX processing (MP) areas include the plutonium oxide and depleted uranium oxide feed materials, material additives, and recovered dust. Plutonium oxide powders are received at the facility packed in qualified shipping packages. Each shipping package holds a single container designed to meet the requirements of the U.S. Department of Energy's (DOE) standard DOE-STD-3013, "Stabilization, Packaging, and Storage of Plutonium-Bearing Materials," issued September 2000 (DOE, 2000).

Equipment used to handle palletized shipping packages includes turntables and bridge cranes. Individual shipping packages are transported by forklifts equipped with drum grips and roller conveyor systems. Automated pick-and-place cranes and robots, slide tables, or roller conveyors handle the 3010 containers outside of the gloveboxes. Powder materials are transported inside of convenience cans, reusable cans, dust pots, sample vials, or in one of a series of powder jars inside glovebox enclosures. Convenience cans, reusable cans, and sample vials are loaded into shuttles that are transferred pneumatically from one glovebox to another. Powder jars and dust pots are transferred between gloveboxes inside of shielded transfer casks along sections of live roller conveyors. Elevators and rotary tilters are used to raise, lower, or dump jars as required for emptying, filling, and weighing. Powder is also transported in bulk form over short distances by a gravity-fed, vibrating conveyor.

Material-handling equipment designed to carry powder containers and pallets includes roller conveyors, ball-screw elevators, pick-and-place robots equipped with gripping manipulators, and pneumatic transfer (LTP) systems. Roller conveyor and elevator systems installed inside of glovebox enclosures are equipped with positive stops and guide rails to prevent interactions between the load and the walls or the floor of the glovebox confinement boundary. Confinement for powder materials handled outside of the glovebox enclosure is provided by the container, which is qualified for a drop from a height greater than the maximum handling height. Confinement for powder materials handled inside the glovebox enclosures is provided by the enclosure.

Pellet-Handling Equipment

[redacted]

Rod-Handling Equipment

[redacted]

Assembly-Handling Equipment

[redacted]

[redacted]

Waste-Handling Equipment

[redacted]

Liquid Process Area Material-Handling Containers

AP process liquids are contained in sample vials used for transfer of liquid samples from AP process units to the laboratory and for use inside the laboratory. The liquid sample vial is composed of a polyethylene bottle and a polyethylene screwed plug. The liquid sample vials are designed to be transferred through the LPT system.
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Powder Process Area Material-Handling Containers

[redacted]

Pellet Process Area Material-Handling Containers

[redacted]

Rod Process Area Material-Handling Containers

[redacted]

Waste Areas Material-Handling Containers

[redacted]

11.6.4 Design Bases for Material-Handling Items Relied on for Safety

Material-handling systems that are designated as IROFS are designed and qualified to perform their safety functions during normal operations, upset conditions, and design-basis events.

Material-handling equipment and support structural members are designed to prevent physical interaction with confinement boundary elements or IROFS under worst-case loading assumptions associated with normal, upset, and design-basis events. The design principles applied to prevent physical interactions include: (1) maintenance of clearance between the equipment and the confinement barrier, (2) equipment that uses actuating mechanisms to grip payloads capable of breaching confinement and are designed to retain their payload under all conditions, and (3) equipment used to hoist loads that could impact confinement boundary elements and is designed and qualified with appropriate margins of safety. Section 11.6.3 of the LA provides the codes and standards used to qualify material transport IROFS.

11.6.5 Load-Handling Accident Sequences

The mixed oxide fuel fabrication facility (MFFF) handles plutonium in the form of solutions, powders, pellets, fuel rods, and fuel assemblies. Depleted uranium dioxide (UO_2) is also handled. A load-handling event could occur when a lifted load is dropped or when either the lifted load or the loading equipment impacts other nearby SSCs, causing a breach of confinement and dispersal of radioactive or hazardous material into the workplace or the environment. Load-handling events are hypothesized to occur throughout the MFFF facility. Locations considered for these events include the following:

- operations inside of a glovebox
- areas surrounding the glovebox and external impacts to the glovebox
- material-handling and transfer events in the MFFF hallways, operational, and storage areas
- events occurring in AP process cells
- events external to the MFFF
- events involving the waste transfer line

The load-handling analysis centered around those events in which the primary confinement barrier is breached (e.g., glovebox window is broken; container, fuel rod, or the waste transfer line is breached). These types of events may result in the dispersal of radioactive or hazardous material.

The MFFF ISA Summary evaluated the following load-handling event groups:

LH-01 Process vessel breaches as a result of maintenance activities in AP process cell

LH-02	Load-handling events during normal operations within the confinement capabilities of the gloveboxes
LH-03	Powder jar falls from a conveyor and impacts a glovebox window
LH-04	Maintenance operations cause a glovebox breach
LH-05/ LOC-7	3013 Container-handling events outside of the gloveboxes
LH-06	Handling of shipping package for the 3013 container
LH-07	Handling of fuel assemblies
LH-08	Handling of MOX fuel transport cask
LH-09	Handling of waste container
LH-10/ LOC-7	Handling of transfer container
LH-11	Load impacts to final very high depressurization (VHD) high-efficiency particulate air (HEPA) filter
LH-12	Consolidated with LH-02
LH-13	Breaching of waste transfer line outside MFFF building
LH-14	Heavy loads or load-handling equipment damaging principal structures or primary confinement boundaries of MFFF building
LH-15	Load handling of depleted of UO_2 container
LH-16/ LOC-6	Rod-handling operations

11.6.5.1 LH-01 Process Vessel Breaches as a Result of Maintenance Activities in Aqueous Polishing Process Cell

In this event, process vessel breaches are postulated to occur as a result of maintenance activities. [redacted]

[redacted]

[redacted]

The staff agrees with the applicant that the combination of process cell entry controls and the process cell exhaust system prevents the receptors from receiving radiation doses in excess of the limits defined in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material." The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 in the event of process cell breaches resulting from maintenance activities in AP process cells.

11.6.5.2 LH-02 Load-Handling Events during Normal Operations within the Confinement Capabilities of the Gloveboxes

These load-handling events involve impact energies within the confinement capability of the gloveboxes during normal operations and can result from a variety of factors, including mechanical failures, control system errors, or operator errors. [redacted]

[redacted]

The gloveboxes are robustly designed, and the staff agrees with the applicant that credible load-handling events will not breach the glovebox. The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 with respect to load handling during normal operations.

11.6.5.3 LH-03 Powder Jar Falls from a Conveyer and Impacts the Glovebox Window

These load-handling events consider the impacts of powder jar falls on the glovebox windows, walls, and floors. These events could be related to jar mispositioning on the jar lift or other mechanical failures.

[redacted]

[redacted]

[redacted] The gloveboxes are also robustly designed, and the staff agrees with the applicant that credible load-handling events will not breach the glovebox. The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 with respect to powder jar falls and their impact on the gloveboxes based on the robust glovebox design and the combination of IROFS to protect against the event.

11.6.5.4 LH-04 Maintenance Operations Cause a Glovebox Breach

These load-handling events relate to maintenance operations in the powder auxiliary (NXR) system that could cause a glovebox breach and radioactive material release. [redacted]

[redacted]

[redacted] The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 with respect to glovebox breaches during maintenance operations based on the IROFS to protect against the event, the robust design of the glovebox, and the administrative control to remove radioactive material during maintenance.

11.6.5.5 LH-05/LOC-7 3013 Container-Handling Events outside of the Gloveboxes

These load-handling events relate to the handling of 3013 containers by automated pick-and-place cranes and robots, slide tables, and roller conveyors that occur outside of the gloveboxes. [redacted]

[redacted]

[redacted]

The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 with respect to the handling of 3013 containers.

11.6.5.6 LH-06 Handling of Shipping Package for the 3013 Container

The PuO_2 Receiving (DCP) unit consists of the material-handling equipment necessary to transfer the 9975 shipping packages from the loading dock in the truck bay to the shipping package unpackaging station. [redacted]

The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 with respect to handling 3013 shipping packages.

11.6.5.7 LH-07 Handling of Fuel Assemblies

[redacted]

[redacted]

The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 with respect to the handling of fuel assemblies based on IROFS administrative controls and the design of the materials handling equipment which limits the potential for load handling events.

11.6.5.8 LH-08 Handling of MOX Fuel Transport Cask

The assembly packaging (TXE) unit consists of the material-handling equipment necessary to load and unload MFFPs from the shipping dock in the truck bay. [redacted]

[redacted]

The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 with respect to MFFP shipping packages.

11.6.5.9 LH-09 Handling of Waste Container

Load-handling events for this event group consist of dropping a waste container during handling. [redacted]

[redacted]

[redacted]

The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 with respect to handling of waste containers.

11.6.5.10 LH-10/LOC-7 Handling of Transfer Container

This event relates to transfer container or SS double door docking system (DDDS) load-handling events which are postulated to occur within the MFFF HDE area. [redacted]

[redacted]

Based on the design of the IROFS for this event and the IROFS administrative controls to prevent use of either the incorrect bin or multiple bins, the staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 with respect to the handling of a transfer container.

11.6.5.11 LH-11 Load Impacts to Final Very High Depressurization High-Efficiency Particulate Air Filter

This event involves load impacts to the final C4/KWG confinement HEPA filters that breach the HEPA filter housing and allow material from the HEPA filters to pass directly to the stack. [redacted]

[redacted]

The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 with respect to load impacts to the final VHD HEPA filter because of IROFS administrative controls limiting material handling near the final HEPA filters.

11.6.5.12 LH-13 Breaching of Waste Transfer Line outside Mixed Oxide Fuel Fabrication Facility Building

This event involves breaching the waste transfer line that runs between the MFFF and the waste solidification building, which releases radioactive liquid waste. [redacted]

[redacted]

The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 with respect to load impacts resulting from a breach in the waste transfer line outside the MFFF building based on the depth of burial of the waste transfer line and administrative controls to prevent damage from trucks or excavation activity near the pipe.

11.6.5.13 LH-14 Heavy Loads or Load-Handling Equipment Damaging Principal Structures or Primary Confinement Boundaries of Mixed Oxide Fuel Fabrication Facility Building

This event involves heavy loads or load-handling equipment that damages the principal structures or primary confinement boundaries of the MFFF building. [redacted]

[redacted]

[redacted]

The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 with respect to heavy loads or load-handling equipment damaging the principle structures or primary confinement boundaries of the MFFF building based on the robust MFFF building structure that would not be damaged by a load handling event and the administrative controls for material handling..

11.6.5.14 LH-15 Load-Handling of Depleted Uranium Dioxide Container

This event group relates to load-handling events that may result in releasing depleted UO_2. The potential causes for the releases can be attributed to, but are not limited to, operator error or mechanical failure or malfunction during UO_2 container-handling operations using a drum lift truck, monorail, hoist, or similar transport means. [redacted]

[redacted]

[redacted]

The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 with respect to load handling of UO_2 containers.

11.6.5.15 LH-16/LOC-6 Rod-Handling Operations

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 with respect to rod-handling operations.

11.6.5 Codes and Standards

Process material-handling cranes and hoists are designed, qualified, and tested to perform the required safety functions, as well as to perform normal operating and maintenance lifts, in accordance the general design codes listed in Section 11.6.5 of the LA.

Process material-handling equipment used to transfer payloads employing fixed geometry devices are designed and qualified to perform required safety functions. The equipment also transfers payloads during normal operation and maintenance in accordance with the general codes and standards listed in Section 11.6.5 of the LA.

11.6.6 Evaluation

The staff evaluated the information provided by the applicant for material-handling equipment and controls in Section 11.6 and other applicable sections of the LA. The review of the design and operation of the material-handling systems was also closely coordinated with the review of applicable portions of Chapters 4 and 5 of the ISA Summary, which discusses the material-handling operations and potential load-handling events.

The staff concluded that the applicant's proposed equipment, facilities, and procedures provide a reasonable level of assurance that load-handling events that cause a release of radioactive material or radiation exposures in excess of the performance requirements of 10 CFR 70.61 are highly unlikely, given the use of the designated IROFS, codes and standards, and management measures, as well as the quality assurance program. The staff further concludes that the baseline design requirements of 10 CFR 70.64 are satisfied.

REFERENCES

(MOX, 2010a) Shaw AREVA MOX Services, "MFFF-License Application," Aiken, SC, October 2009.

(MOX, 2010b) Shaw AREVA MOX Services, "MFFF-Integrated Safety Analysis Summary," Aiken, SC, October 2009.

(DOE, 2000) U.S. Department of Energy, "Stabilization, Packaging, and Storage of Plutonium-Bearing Materials," DOE-STD-3013, Washington, DC, September 2000.

(NRC, 2000) U.S. Nuclear Regulatory Commission, NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility," Washington, DC, August 2000.

10 CFR Part 70, Domestic Licensing of Special Nuclear Material

10 CFR Part 71, Packaging and Transportation of Radioactive Material

Part 178.350, Specification 7A, general packaging, Type A

11.7 <u>Fluid Transport Systems</u>

This chapter of the safety evaluation report (SER) contains the staff's review of the fluid transport systems described by the applicant in Section 11.7 of the license application (LA) (MOX, 2010a) and Section 4.7 of the Integrated Safety Analysis (ISA) Summary (MOX, 2010b).

The objective of this review is to determine whether the fluid transport systems' items relied on for safety (IROFS) and their design bases identified by the applicant provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents. The staff evaluated the information provided by the applicant for fluid transport systems by reviewing Section 11.7 and other sections of the LA (MOX, 2010a), Section 4.7 of the ISA Summary (MOX, 2010b), supplementary information provided by the applicant, applicable fluid transport systems codes and standards, and relevant documents available at the applicant's offices which were not submitted by the applicant. The staff closely coordinated its review of the fluid transport systems' design bases and strategies with the review of other sections in this SER (e.g., the ISA in Chapter 5, fire protection in Chapter 7, and chemical safety in Chapter 8).

11.7.1 Regulatory Requirements

The staff reviewed how the information in the LA (MOX, 2010a) and the ISA Summary (MOX, 2010b) addressed the following regulations:

- Title 10 of the *Code of Federal Regulations* (10 CFR) § 70.61(e), "Performance requirements" stipulates that each engineered or administrative control or control system that is needed to meet the performance requirements be designated as an IROFS and the safety program that ensures that each IROFS will be available and reliable to perform its intended function when needed

- 10 CFR § 70.64, "Requirements for New Facilities or New Processes at Existing Facilities," stipulates requirements of baseline design criteria and defense-in-depth practices to new facilities or new processes at existing facilities

11.7.2 Regulatory Acceptance Criteria

The review focused on the design bases of fluid transport systems. The staff reviewed and evaluated the information provided by the applicant for the safety function, system description, and safety analysis. The review also encompassed design-basis considerations, such as redundancy, independence, reliability, and quality. The staff also performed its review in accordance with Section 11.4.7.2 in NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility" (NRC, 2000). Section 11.4.7.2 lists the following regulatory acceptance criteria for fluid transport systems (referred to as "material transport system (pumps and valves)" in NUREG-1718):

- Capacity is adequate to handle the expected volume of radioactive material during normal operating and accident conditions.

- Redundancy or diversity of components prevents the release of radioactive materials to the environment or contributes to the safe operation of the fluid transport systems.

- The fluid transport system can be safely shut down during normal and accident conditions. Provisions for emergency power are included for critical process components.

- Tank and piping systems are of welded construction to the fullest extent possible.

- Tank and piping systems are designed to take advantage of gravity flow to reduce the potential for contamination associated with pumping and pressurization.

- Criticality will not occur under normal and credible accident conditions.

- All system components expected to be in contact with strong acids or caustics are corrosion resistant.

- Piping is designed to minimize entrapment and buildup of solids in the system.

- The systems are evaluated to determine the need for hoods, gloveboxes, and shielding for personnel protection. Generally, wet processing operations involving gram quantities of plutonium and any operations involving 50 micrograms of respirable plutonium are conducted in a glovebox.

- Surface finishes of materials in the work areas have satisfactory decontamination characteristics for their particular application.

- Fluid transport systems maintain functionality when subjected to tornadoes, tornado missiles, earthquakes, floods, and any other natural phenomena deemed to be credible as established in the ISA.

11.7.3 Fluid Transport System Description

The primary function of the fluid transport systems is to safely and reliably handle the process and utility fluids during the aqueous polishing (AP), mixed oxide process (MP), and utility processes. Section 11.8 of this SER discusses other fluid-containing support systems.

11.7.3.1 Description

The Mixed Oxide Fuel Fabrication Facility (MFFF) fluid transportation system consists of vessels, standardized equipment (e.g., demisters, leak detectors), process columns, heat exchangers, pumps, filters, piping, valves, and some additional miscellaneous equipment. The applicant has identified the following features or components as IROFS:

- system overpressurization controls
- double-wall pipe
- seismic isolation valves

Section 11.8.4.1 of this SER also analyzes seismic isolation valves.

The major components of the fluid transport systems are part of the primary process and are located in the AP area of the facility. In addition to piping and valves, the major components of the fluid transport systems include the following:

- Welded process equipment, which includes vessels, tanks, process columns, and heat exchangers. In general, fully welded process equipment is located in process cells. Storage tanks vary in design at different stages of the primary process. Storage tanks include annular tanks, stab tanks, and conventional tanks. These tanks are fabricated using fully welded construction. Other welded process equipment includes various small tanks used in the AP process, such as separating pots, leak detection pots, barometric seal pots, pulse column pots, drip pots, condensate pots, and demisters. The AP process columns and various AP process heat exchangers used in radiological service are also of a fully welded construction.

- Partially welded process equipment and prime movers, which includes filters, mixing tanks, and precipitators. Other process prime movers include pumps, low-pressure airlifts, ejectors, and siphons. Pump types include centrifugal and positive displacement dosing pumps.

Process fluid transport systems are classified as one of the following fluid transport system categories based on the nature of the fluid contained:

- FTS Category 1—Fluid systems that contain process fluids with significant quantities of plutonium or americium.

- FTS Category 2—Fluid systems that contain process fluids which potentially include trace quantities of plutonium or americium.

- FTS Category 3—Fluid systems that contain radioactive waste fluids which potentially include trace quantities of plutonium or americium.

- FTS Category 4—Fluid systems that contain nonprocess fluids which include no plutonium or americium.

Process fluid transport systems that contain FTS Category 1 or 2 fluids and their structural support elements are designed and qualified to perform the following general safety functions:

- Prevent criticality by maintaining the pressure boundary integrity of fluid transport system components as required to control the geometry of fissile material.

- Confine radioactive or toxic material as required to meet the performance requirements of 10 CFR § 70.61, “Performance Requirements.”

- Prevent interaction between confinement boundary or criticality prevention elements and non IROFS safety systems and equipment which could process fluid transport systems.

The MFFF Fluid transport components are designed to efficiently move fluid with a low head and small flow rates. Systems are laid out so as to minimize fluid traps, dead spots, and other volumes that cannot be completely drained (with the exception of loop seals). Radiological fluids are maintained within at least two levels of confinement. Vessels that contain radiological fluids are mounted over drip trays to collect any leakage. Each drip tray is sized to hold the contents of the largest vessel in a critically safe configuration where appropriate. Radiological fluids are transferred using static transfer means, such as gravity flow, airlifts, air jets, and steam jets when practical. Systems that contain hazardous fluids are either contained within trenches, rooms, or double-walled piping. These systems could also be accessible for inspection and have fully welded construction. Fluid-bearing components located within process cells are specified with corrosion allowances, and the welding joints are radiographed, as appropriate. Each process cell is lined with a drip tray and a sump. The sump is monitored for leakage. The components located in the process cell are not normally accessible. However, the cell can be accessed for maintenance, if required. The process fluid can be isolated from the cell, and decontamination fluid can be used to flush equipment and piping. While many such components contain plutonium in excess of gram quantities, process cell and welded equipment confinement for liquid containing systems provides for adequate protection of personnel since there is no direct contact with the materials.

11.7.4 Fluid Transport System Evaluation

11.7.4.1 Capacity

The fluid transport system is designed to handle the expected volume of radioactive material during normal operating and accident conditions. Table 4.7-2 of the ISA Summary (MOX, 2010b) lists the design pressure, temperature, and flow and volume for the fluid transport system components. This chart lists the additional safety factors, over and above code and standard requirements, that are applied to the design. These safety factors provide an additional level of safety. Vessels that contain radiological fluids are mounted over drip trays. These trays are sized to contain the contents from the largest vessel in the cell.

Based on the information on fluid system capacity and the additional safety factors over and above the code and standard requirements, the staff finds that the fluid system capacity is acceptable and consistent with the guidance in Section 11.4.7.2 (A) of NUREG 1718.

11.7.4.2 Redundancy and Diversity

The fluid transport system is designed for redundancy and diversity of components that will prevent the release of radioactive materials to the environment or needed for the safe operation of the material transport system. The fluid transport system is designed with multiple layers of confinement and is supplemented by administrative programs designed to monitor the integrity of these systems. Radiological fluids are maintained within at least two layers of confinement. Piping systems are double walled, with leak detection systems if they are not located in process cells or gloveboxes. Drip trays with sump monitors are designed to detect leakage. Levels inside the tanks are remotely monitored using level instrumentation. Redundancy and diversity in the design are the result of the various safety factors and types of equipment provided in the design and the layering of active and passive controls that protect the fluid transport system. The staff has reviewed the description of the fluid transport system components and finds these systems to be diverse and redundant and is consistent with the guidance in Section 11.4.7.2 (B) of NUREG-1718.

11.7.4.3 Shutdown and Emergency Power

The fluid transport system can be safely shut down during normal and accident conditions. Provisions for emergency power are included for critical process components. The AP process control system, the electrical power system, and basic system design criteria primarily control the ability of the proposed systems to shut down safely. All fluid transport systems containing radioactive material are designed for the design-basis earthquake. In the event of containment breach, liquid is collected in a drip tray in the process cell or glovebox. This fluid is then recycled through the waste treatment management units. Pipes containing radioactive material outside of a process cell or glovebox are double walled. Seismically qualified IROFS are designed to withstand the design-basis seismic event. All IROFS are designed for normal, off-normal, and design-basis accident environmental conditions. (See Section 11.5 of this SER for additional detail on instrumentation and control system safety.) Emergency uninterrupted power supplies those electrical loads requiring power for safe shutdown. (See the discussion of electrical system safety in Section 11.4 of this SER for more detail.) Based on the information provided by the applicant and the discussions on emergency power systems in Section 11.4 of this SER, the staff finds that the material transport system can be shutdown consistent with the guidance in Section 11.4.7.2 (C) of NUREG-1718.

11.7.4.4 Welded Construction

The welded equipment and piping components handling radiological fluids are fully welded construction and are located in the process cell confinement. Radiological fluid bearing components that do not permit fully welded construction are installed in a glovebox. The design of the fluid transport system components is specified with appropriate corrosion allowances. The welded joints are radiographed, as appropriate, to ensure conformance with construction codes that were committed to by the applicant in the LA.

Radiological fluids are contained within at least two levels of confinement and are transferred using static transfer means, such as gravity flow, airlifts, air jets, and steam jets when practical. Piping components carrying radiological fluids are either fully welded with double-wall construction between two confinements or installed in gloveboxes or process cells.

Material used for the construction of this equipment is specified in accordance with ASME and American Society for Testing and Materials (ASTM) codes and material specifications. ASTM materials are also used for the fabrication of other components. In general, design of equipment to these standards means that the components are designed for the most severe service conditions. Included in the severe service conditions are pressure, temperature, material compatibility, and corrosion. The staff has reviewed the design basis for welding and finds it acceptable, based on the information submitted and references to the codes and standards for the design and construction of the fluid transport system. The information is consistent with the guidance on welded construction in Section 11.4.7.2 (D).

11.7.4.5 Gravity Flow

The fluid transport system’s tank and piping systems are designed to take advantage of gravity flow to reduce the potential for contamination associated with pumping and pressurization. Radiological fluids are transferred using gravity flow, airlifts, air jets, and steam jets when practicable.

Hydraulic seals are used to prevent backflow of process fluid to auxiliary systems during reagent addition. The liquid seal or plug is maintained by the piping configuration. The seal is implemented by a “U” bend in piping or by hydraulic seal pots. The hydraulic seal design ensures that the seal remains filled with liquid at all times, the seal withstands internal pressure differences between connected vessels, and siphon action does not occur.

Check valves are used only in the process fluid pressure boundary. The check valve design is based on effective pressure drop, type of seating material, pressure and flow reversal response time, mounting requirements, and reliability and maintainability. Redundant isolation valves that are IROFS are used to automatically isolate utility and reagent fluids in the process area when earthquake conditions are detected. These IROFS isolation valves close in the event of valve or actuator failure. Isolation valve selection is based on process hydraulics, control system characteristics, mounting requirements, and other valve specifications. The valves will be specified for service after consideration of the chemical characteristics of the fluid, piping material of construction, and operating conditions. The valves will be designed and constructed consistent with good engineering practices and in accordance with ASME and American Petroleum Institute (API) codes stated in the LA (MOX, 2010a).

The valves and their supports will also be designed to withstand and remain operable during the design-basis earthquake, as they are intended to prevent uncontrolled flooding of the MFFF building as a result of a seismic event. The safety function of the isolation valves is to maintain safe isolation between controlled areas and uncontrolled areas that may contain radioactive materials. Based on the proposed design and use of industry practices, the staff finds that the redundancy and diversity of the fluid transport systems adequately prevent the release of radioactive materials to the environment.

Separator or knockout pots are specified for piping in which fluid transfer is made by air or vacuum lift. The separated fluid is allowed to flow by gravity into the desired component, while the airflow vents at the top of the pot. This design prevents backflow siphoning. Steam jet lift transfer system piping is terminated in the receiving vessel vent space to provide an air gap that prevents backflow siphoning. Siphons are used to initiate gravity transfer of fluids in applications in which flow rate is not critical. The siphon transfers liquid from the higher upstream tank to the lower downstream tank. The elevation difference between tanks prevents backflow. Knockout pots, steam jet lifts, and elevation differences between tanks are passive features that help to prevent cross contamination.

The staff has reviewed the facility's design basis and, based on the commitments to appropriate nuclear industry codes and standards and the design which incorporates gravity flow, concludes that the design is acceptable and consistent with the guidance in Section 11.4.7.2 (E) of NUREG-1718.

11.7.4.6 Criticality

The fluid transport system is designed so that criticality will not occur under normal and credible accident conditions. Drip trays in process cells are sized to contain the contents from the largest vessel in the cell and shaped to maintain the leaked fluid in a geometry that is criticality safe. Chapter 6 of this SER evaluates nuclear criticality safety.

11.7.4.7 Corrosion

The fluid transport system components expected to be in contact with strong acids or caustics are corrosion resistant. The construction materials of the facility's fluid transport systems are selected based on compatibility with the physical and chemical characteristics of the process fluids. In general, FTS Category 1 components use Type 304L or 316L stainless steel and alloys of titanium and zirconium. Components of FTS Category 2 and 3 that handle acidic or alkaline fluids are generally constructed from Type 304L or 316L stainless steel. Material used for the construction of this equipment is specified in accordance with ASME and ASTM material specifications. In general, design of equipment to these standards means that the components are designed for the most severe service conditions. Included in the severe service conditions are pressure, temperature, stress, and corrosion. The corrosion allowance for the construction materials is specified for each component in accordance with industry practices and from experience at the La Hague facility. On the basis of the applicant's commitment to design the system to be resistant to corrosion and its commitment to use material maintenance and surveillance programs to detect and limit the damage from corrosion, the staff finds this design to be acceptable and consistent with the guidance in Section 11.4.7.2 (G) of NUREG-1718.

11.7.4.8 Entrapment and Buildup of Solids

The fluid transport system piping is designed to minimize entrapment and buildup of solids in the system. To minimize or eliminate the buildup of solids within the AP liquid process units, the following measures are taken: (1) the process technology does not involve suspension of the solid particles in liquid past the dissolution stage (before precipitation); (2) undissolved particles in the dissolution unit are removed through multiple stages of filters; (3) the liquid solutions used in the processes are not saturated solutions; (4) air spargers are provided as necessary to keep solutions well mixed; (5) decontamination fluid is supplied throughout the AP process units to remove potential buildup; (6) piping layout is designed with an adequate slope, without sharp directional change, and without low point traps; (7) demisters are cleaned periodically with decontamination solution; and (8) freeze jackets and fluid thermal cycling are used to prevent clogging of piping.

The following means are provided to minimize buildup of solids in the powder-handling processes: (1) the dry process environment minimizes ingress of moisture, (2) process equipment is stacked vertically, (3) the surface finish provides non-sticking tendencies, and (4) vibrators maintain free-fall gravity feed at selected locations.

The staff has reviewed the design bases for minimizing entrapment and buildup of solids and finds the design to be acceptable based on the systems and components that are to be used to reduce buildup of solid materials and is consistent with the guidance in Section 11.4.7.2 (H) of NUREG-1718.

11.7.4.9 Containment

The fluid transport system hoods, gloveboxes, and shielding for personnel protection are generally required for wet processing operations involving more than gram quantities of plutonium or general operations involving 50 micrograms or more of plutonium in respirable form. Process cells contain equipment that handles radioactive materials in chemical solutions that is fully welded and does not require routine maintenance. Equipment that cannot be fully welded is installed in gloveboxes. Welded equipment is designed in accordance with good engineering and industrial practices, codes and standards, and any other supplemental requirements. Equipment containing radioactive materials in the powder process (MP) is contained in gloveboxes in process rooms that provide confinement equivalent to fully welded equipment in process cells. The staff has reviewed the list of equipment and agrees that the equipment involving more than gram quantities of plutonium or general operations involving 50 micrograms or more of plutonium in respirable form are properly contained in either gloveboxes or fully welded process equipment. Therefore, because the facility design follows the guidelines of Section 11.4.7.2 (I) of NUREG-1718, the staff finds the design basis to be acceptable.

11.7.4.10 Surface Finishes

The fluid transport system surface finishes of materials in the work areas have satisfactory decontamination characteristics for their particular application. The surface finish in the powder-handling process area possesses nonsticking tendencies. Valves have smooth interior surfaces to present little surface area for particulate matter to plate out. The staff finds these features acceptable.

Inside glovesboxes, material-handling devices are most commonly made of stainless steel, which allows for decontamination. Use of components requiring painting is minimized. Components are designed to be easily accessible and readily dismantled for decontamination. Lubrication is limited to the extent practical. Internal welds are continuous and ground smooth, and reentrant corners have large radii.

The staff has reviewed the design bases related to surface finishes and finds the design to be acceptable based on the materials used in the design and is consistent with the guidance in Section 11.4.7.2 (J) of NUREG-1718.

11.7.4.11 Natural Phenomena

Fluid transport systems maintain functionality when subjected to tornadoes, tornado missiles, earthquakes, floods, and any other natural phenomena deemed to be credible, as further established in the ISA performed by the applicant. Radiological fluids are maintained within at least two levels of confinement. All piping components designated as IROFS are designed to withstand the design-basis earthquake loads.

Fluid transport systems are designed and qualified according to national codes and standards enabling them to perform their safety function during normal operations, upset conditions, and design-basis events. Section 11.7 of the LA lists these codes and standards for the IROFS. The design basis of the seismic monitoring system ensures that it provides sufficient data to evaluate the response of the confinement structure and other IROFS to a seismic event and to initiate a shutdown of process systems in the event of a high seismic event. Section 11.11 of this SER further discusses seismic qualifications and natural phenomena accidents.

11.7.5 Loss of Confinement Event Sequences

Section 5.3.3 of the ISA Summary (MOX, 2010b) discusses the results of the evaluation of loss of confinement events. The following loss of confinement event groups are related to the fluid-handling system and are evaluated in the MFFF ISA:

- LOC-3—Leaks from AP vessels or pipes within process cells
- LOC-4—Leaks from AP vessels or pipes within gloveboxes
- LOC-5—Backflow from process vessels through utility lines
- LOC-11—Process fluid line leak in a C3 area outside a glovebox
- LOC-13—Uncontrolled release of nitrogen tetroxide

11.7.5.1 LOC-3—Leaks from Aqueous Polishing Vessels or Pipes within Process Cells

LOC events are postulated to occur in the process cells in the Aqueous Polishing Area (BAP) as a result of leaks from process vessels or pipes or both. [redacted]

[redacted]

[redacted]

With respect to the discussions in Sections 11.7.4.7 and 11.7.4.9 of this SER, the staff concludes with reasonable assurance that the applicant has demonstrated that the facility will comply with the performance requirements of 10 CFR § 70.61 (b) and (c) in the event of leaks from AP vessels or pipes within process cells. The staff also agrees with the applicant that the event group is highly unlikely, given the use of the designated IROFS, codes and standards, management measures, and quality assurance program.

11.7.5.2 LOC-4—Leaks from Aqueous Polishing Vessels or Pipes within Gloveboxes

Process cells are not accessible during normal operations. AP process equipment that requires frequent operation or maintenance or has electrical connections is placed within gloveboxes outside of the process cells. Such equipment includes valves, pumps, filters, piping, vessels, and associated instrumentation. Unlike equipment in the process cells, AP equipment contained in gloveboxes may not be fully welded; therefore, leaks from valve packing, pump seals, and instrument penetrations may occur. To confine leakage, this equipment is placed in gloveboxes equipped to collect and drain the leakage. This event group addresses a leak of process solution from AP process equipment in a glovebox. [redacted]

[redacted]

[redacted]

With respect to the discussions in Sections 11.7.4.4, 11.7.4.7, 11.7.4.8, and 11.7.4.9 of this SER, the staff concludes, with reasonable assurance, that the applicant has demonstrated that the facility will comply with the performance requirements of 10 CFR § 70.61 in the event of leaks from AP vessels or pipes within gloveboxes. The staff also agrees with the applicant that the event group is highly unlikely, given the use of the designated IROFS, codes and standards, management measures, and quality assurance program.

11.7.5.3 LOC-5—Backflow from Process Vessels through Utility Lines

This event involves the backflow of AP process vessel solutions into interfacing reagent systems and utilities, causing a potential chemical incompatibility (explosion or unplanned evolution and release of chemicals) or radiation and chemical exposure to facility workers in process areas normally expected to be devoid of significant quantities of radiological material. [redacted]

[redacted]

[redacted]

[redacted]

With respect to the discussions in Sections 11.7.4.4, 11.7.4.5, 11.7.4.7, 11.7.4.8, and 11.7.4.9 of this SER, the staff concludes, with reasonable assurance, that the applicant has demonstrated that the facility will comply with the performance requirements of 10 CFR § 70.61 (b) and (c) related to backflows from process vessels through utility lines. The staff also agrees with the applicant that the event group is highly unlikely, given the use of the designated IROFS, codes and standards, management measures, and quality assurance program.

11.7.5.4 LOC-11—Process Fluid Line Leak in a C3 Area outside a Glovebox

Loss of confinement events are postulated to occur because of a leak from a line carrying a process fluid or an off-gas in a C3 confinement zone area outside of a glovebox in the BAP. [redacted]

[redacted]

[redacted]

[redacted]

With respect to the discussions in Sections 11.7.4.4, 11.7.4.7, and 11.7.4.8 of this SER, the staff concludes, with reasonable assurance, that the applicant has demonstrated that the facility will comply with the performance requirements of 10 CFR § 70.61 as they relate to a process fluid line leak in a C3 area outside a glovebox. The staff also agrees with the applicant that the event group is highly unlikely, given the use of the designated IROFS, codes and standards, management measures, and quality assurance program. .

11.7.5.5 LOC-13—Uncontrolled Release of Nitrogen Tetroxide

[redacted]

[redacted]

[redacted]

[redacted]

With respect to the discussions in Sections 11.7.4.4, 11.7.4.7, and 11.7.4.8 of this SER, the staff concludes, with reasonable assurance, that the applicant has demonstrated that the facility will comply with the performance requirements of 10 CFR § 70.61 as they relate to an uncontrolled release of nitrogen tetroxide. Additionally, the administrative process cell entry control and the process cell exhaust system contribute to the reduction of the likelihood of chemical exposures to the workers. The staff also agrees with the applicant that the event group is highly unlikely, given the use of the designated IROFS, codes and standards, management measures, and quality assurance program.

11.7.6 Explosion Event Sequences

Section 5.3.6 of the ISA Summary (MOX, 2010b) discusses the results of the evaluation of explosion events. The following explosion event groups are related to the fluid-handling system and were evaluated in the MFFF ISA:

- EXP-08—AP vessel overpressure explosion
- EXP-09—Pressure vessel overpressurization explosion

11.7.6.1 EXP-08—Aqueous Polishing Vessel Overpressure Explosion

The AP process of the MFFF uses many vessels throughout the process to remove impurities from the incoming plutonium oxide powder. These vessels receive and pass on process solutions, receive reagents, receive utilities, and have various combinations of vents, overflows, and drains. There are multiple chemical reactions that take place in these vessels during the dissolution of the powders, purification of the process solution, and conversion back to the powder form. Some of these chemical reactions sometimes produce off-gases that could potentially overpressure a vessel.

An overpressure explosion is generally described as a sudden release of energy either by a sudden increase in either volume, temperature, or pressure or any combination of these. As examples, energy release can be propagated by pressure waves, the kinetic energy of fragmented vessels and vessel contents, or thermal radiation. Combustion or other chemical-reaction-generated explosions with pressure waves are categorized as deflagrations, if these waves are subsonic, and detonations if they are supersonic (shock waves). An overpressure event remaining below 50.9 pounds per square inch gauge (psig) (0.35 megapascal gauge (MPa-g)) is not considered an explosion because of the lower pressure involved, but is considered a leak or an overpressure breach. Consequences for this type of overpressure event are bounded by those of explosions at 50.9 psig. An overpressure event is an event which results in the normal operating pressure range within a container being exceeded, while pressure is maintained below the lower limit of 50.9 psig (0.35 MPa-g) for events involving liquids confined in a vessel or container and vented above the surface of the liquid after a slow buildup of pressure.

The events covered by this section are process-related overpressures involving reagents or utilities or both in AP vessels, tanks, and piping in AP process cells or gloveboxes that result in an energetic breach of the AP vessels, tanks, and piping and the potentially energetic dispersal of radioactive materials. [redacted]

[redacted]

[redacted]

- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]

[redacted]

With respect to the discussions in Sections 11.7.4.4, 11.7.4.7, and 11.7.4.8 of this SER, the staff concludes, with reasonable assurance, that the applicant has demonstrated that the facility will comply with the performance requirements of 10 CFR § 70.61(c) as they relate to AP vessel overpressure explosions. The staff also agrees with the applicant that the event group is highly unlikely, given the use of the designated IROFS, codes and standards, management measures, and quality assurance program.

11.7.6.2 EXP-09—Pressure Vessel Overpressurization Explosion

[redacted]

[redacted]

[redacted]

[redacted]

Based on the robustness of the process cell walls and the lack of sufficient energy from the potential pressure failure to damage IROFS, the staff finds, with reasonable assurance, that the performance requirements of 10 CFR § 70.61 are met with respect to pressure vessel overpressurization explosions. The staff also agrees with the applicant that the event group is highly unlikely, given the use of the designated IROFS, codes and standards, management measures, and quality assurance program.

11.7.7 Codes and Standards

The design, engineering, and construction of IROFS are guided by the codes, standards, and practices defined in FTS Categories 1 and 2 in Tables 11.7-1 and 11.7-2 of the LA (MOX, 2010a). The fluid transport systems that consist of vessels, pumps, piping, and valves are classified as FTS-1 and FTS-2 and are designed accordingly.

Section 11.7.3 of the LA (MOX, 2010a) identifies other design-basis codes and standards for the fluid transport system.

11.7.8 Evaluation

The staff evaluated the information provided by the applicant for fluid transport equipment and controls in Section 11.7 and other applicable sections of the LA (MOX, 2010a). The staff concluded that the applicant's proposed equipment, controls, and procedures provide a reasonable level of assurance that events related to the fluid transport systems that could cause a release of radioactive material or radiation exposures in excess of the performance requirements in 10 CFR 70.61 are highly unlikely, given the use of the designated IROFS, codes and standards, management measures, and quality assurance program.

REFERENCES

(NRC, 2000) U.S. Nuclear Regulatory Commission, NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility," Washington, DC, August 2000.

(MOX, 2010a) Shaw AREVA MOX Services, “MFFF—License Application,” Aiken, SC, March 2010.

(MOX, 2010b) Shaw AREVA MOX Services, “MFFF—Integrated Safety Analysis Summary,” Aiken, SC, March 2010.

10 CFR Part 70, Domestic Licensing of Special Nuclear Material.

11.8 Fluid Systems

This chapter of the safety evaluation report (SER) contains the U.S. Nuclear Regulatory Commission (NRC) staff’s review of the fluid systems described by the applicant in Section 11.8 of the license application (LA) (MOX, 2010a) and Section 4.8 of the Integrated Safety Analysis (ISA) Summary (MOX, 2010b). The objective of this review is to determine whether the fluid systems’ items relied on for safety (IROFS) and their design bases identified by the applicant provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents. The staff evaluated the information provided by the applicant for fluid systems by reviewing Section 11.8 and other sections of the LA, Section 4.8 of the ISA Summary, supplementary information provided by the applicant, applicable codes and standards, and relevant documents available at the applicant’s offices but not submitted by the applicant. The staff closely coordinated its review of fluid systems’ design bases and strategies with the review of other chapters in this SER (e.g., ISA in Chapter 5, fire protection in Chapter 7, and chemical safety in Chapter 8).

11.8.1 Regulatory Requirements

The staff reviewed how the LA (MOX, 2010a) and ISA Summary (MOX, 2010b) address the following regulations:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 70.61(e), as it specifically relates to the requirement that each engineered or administrative control or control system that is needed to meet the performance requirements be designated as an IROFS and as it relates to the safety program that ensures that each IROFS will be available and reliable to perform its intended function when needed

- 10 CFR 70.64, “Requirements for New Facilities or New Processes at Existing Facilities,” which requires that baseline design criteria and defense-in-depth practices be incorporated into new facility design

- 10 CFR 70.64(a)(2), which requires that new facility designs adequately protect against natural phenomena hazards and consider the most severe documented historical events for the site

11.8.2 Regulatory Acceptance Criteria

The review focused on the design bases of fluid systems, their components, and other related information. For fluid systems, the staff reviewed and evaluated the information provided by the applicant for the safety function, system description, and safety analysis. The review also encompassed proposed design-basis considerations, such as redundancy, independence, reliability, and quality.

Section 11.4 of NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility" (NRC, 2000) outlines the acceptance criteria related to plant systems. No section in NUREG-1718 is specifically designated to address all of the items discussed in Section 11.8 of the LA and Section 4.8 in the ISA Summary, with the exception of the cooling water system which Section 11.4.4 of NUREG-1718 describes. However, the function, description, control concepts, and system interfaces pertaining to each fluid system are addressed by the regulatory requirements identified in the review guidance in Section 11.4 of NUREG-1718.

11.8.3 Staff Review and Analysis

11.8.3.1 Fluid Systems Description

The Mixed Oxide Fuel Fabrication Facility (MFFF) fluid systems consist of mechanical utility fluids, bulk gases, and chemical reagents designed to support the MFFF. The applicant has identified the following IROFS in fluid systems:

- instrumentation and controls (I&C)
- sampling systems
- double-walled piping
- relay lock outs for transfer pumps
- isolation valves
- seismic isolation valves
- emergency diesel generator fuel oil system (EDGFOS)
- emergency scavenging air system (ESAS)
- moisture and temperature sensors
- hydrogen analyzer
- signal for shut down of transfer pump

Non-IROFS systems may contain IROFS components, such as seismic isolation valves. Section 11.8.4 of this SER evaluates seismic isolation valves, EDGFOS, and ESAS IROFS.

11.8.3.2 Mechanical Utility Systems

The mechanical utility systems comprise the following:

- heating, ventilation, and air conditioning (HVAC) chilled water
- process chilled water
- demineralized water (DMW)
- process hot water
- process steam
- building services
- EDGFOS
- standby diesel generator fuel oil
- service air
- instrument air
- decontamination
- breathing air

- vacuum radiation monitoring

Most of these systems are non-IROFS but contain components, such as seismic isolation valves, that are IROFS.

11.8.3.2.1 HVAC Chilled Water System

The HVAC chilled water system supplies chilled water and cooling water for the MFFF HVAC during normal operation. This system consists of an external cooling loop and an internal cooling loop. The heat will be transferred between the two loops via a heat exchanger. The external loop will be maintained at a higher pressure relative to the internal loop. This design reduces the risk of radiological or chemical material being dispersed from inside the MFFF to the environment.

The HVAC chilled water system is not an IROFS, but the system uses redundant and independent seismic isolation valves for IROFS. Section 11.8.4.1 of this SER evaluates seismic isolation valves.

11.8.3.2.2 Process Chilled Water System

The process chilled water system supplies chilled water and cooling water to the MFFF during normal operation. This system consists of an external cooling loop and an internal cooling loop. The heat will be transferred between the two loops via a heat exchanger. The external loop will be maintained at a higher pressure relative to the internal loop. This design reduces the risk of radiological or chemical material being dispersed from inside the MFFF to the environment. Leakage from either loop will be constantly monitored by pressure sensors, as well as a low pressure monitor on the expansion tank. Radiation monitors are used to detect any in-leakage of radioactive material.

The process chilled water system is not an IROFS, but the system uses redundant and independent seismic isolation valves, sampling, and double-walled piping for IROFS. IROFS sampling will be performed in some of the closed cooling water loops to ensure that no water has leaked from the process chilled water system into other units. Double-walled pipe will be used in each moderator controlled room to prevent the release of water.

11.8.3.2.3 Demineralized Water System

The DMW system receives, stores, and transfers pressurized and gravity-fed demineralized water to process equipment and utility systems. The DMW system produces, stores, and transfers pressurized and gravity-fed (i.e., unpressurized) demineralized water to process equipment and utility systems for use in reagent preparation, solution dilution, initial loop filling, humidification of sintering gas, general laboratory functions, sintering furnace cooling, and miscellaneous process purposes. This system is not an IROFS, but contains redundant and independent seismic isolation valves that are IROFS.

11.8.3.2.4 Process Hot Water System

The process hot water system supplies electrically heated demineralized water to various parts of the aqueous polishing (AP) process. This system comprises pumps, expansion tanks, valves, piping, electrical heaters, and heat exchangers. This system transfers heat via heat exchangers to process equipment. Radiation detectors will constantly monitor leakage from either loop. This system is not an IROFS. However, IROFS pressure transmitters are provided on the heating loops to ensure that the pressure in the hot water system is always greater than that of the process fluid.

11.8.3.2.5 Process Steam System

The process steam system supplies and regulates steam to various AP process units. The condensate return will provide makeup water for the steam generator. The steam generator equipment consists of electric boilers, deaerator, pumps, chemical feed equipment, blowdown tank, condensate tanks, valves, I&C, and piping. Samples from the system will be periodically analyzed for chemical additives and corrosion inhabitation. Radiation monitors will constantly monitor leakage into the system.

The process steam system is not an IROFS, but the system uses redundant and independent seismic isolation valves, as well as isolation valves in the supply lines to steam jets in the AP process. These valves are IROFS and are used to prevent the transfer of unanalyzed material into the AP process.

11.8.3.2.6 Building Services System

The building services system supplies potable water and drainage for personnel decontamination. This system consists of a shower, toilet, and sink. The waste water will be sampled and analyzed to determine the appropriate disposal method. The Savannah River Site will supply the potable water system. This system is not an IROFS, but contains redundant and independent seismic isolation valves that are IROFS.

11.8.3.2.7 Emergency Diesel Generator Fuel Oil System

[redacted]

[redacted]

The applicant has committed to design the EDGFOS to comply with American National Standards Institute/American Nuclear Society (ANSI/ANS) 59.51-1997, "Fuel Oil Systems for Safety-Related Emergency Diesel Generators" (ANSI/ANS, 1997), and to meet the requirements of National Fire Protection Association (NFPA) 30, "Flammable and Combustible Liquids Code" (NFPA, 1996), and NFPA 37, "Standard for the Installation and Use of Stationary Combustion Engines and Gas Turbines" (NFPA, 1998a). This is in addition to the applicant's commitment to meet Institute of Electrical and Electronics Engineers (IEEE) 387-1995, "IEEE Standard Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations," and the codes and standards for piping, tanks, and valves shown in Table 11.7.1 of the LA.

11.8.3.2.8 Standby Diesel Generator Fuel Oil System

The standby diesel generator fuel oil system receives, samples, stores, and supplies fuel oil to the standby diesel generators. The standby diesel generators are used to provide power in the event of a loss of primary offsite power. This system consists of a common tank, transfer pump, filters, valves, controls, and piping. The common tank will be stored in a belowground concrete vault. This system and the components contained in the system are not IROFS.

11.8.3.2.9 Service Air System

The service air system pressurizes, dries, filters, stores, and supplies pressurized air to the facility service air headers for maintenance and utility. This system will be supplied on a skid that contains an inlet filter, aftercooler, moisture separator, and trim cooler. This system is designed to minimize moisture by using air dryers after the air has passed by the water aftercooler. A coalescing filter will be the final step that removes contaminant.

The service air system is not an IROFS, but contains redundant and independent seismic isolation valves for IROFS. These valves prevent overpressurization of the AP process equipment. Even though the service air system has a variety of uses, such as tank agitation, this safety function does not depend on this system; therefore, this system is not an IROFS.

11.8.3.2.10 Instrument Air System

The instrument air system (IAS) continuously provides dry, clean, oil-free instrumentation air for bubbling air, air-operated valve operation, glovebox ventilation, and process equipment. The IAS supplies instrument quality air for the following: (1) instruments with buffered storage for control valves, transmitters, ventilation (building) dampers, and inline analyzers; (2) ventilation and cooling for gloveboxes and the pelletizing press bellows; (3) normal bubbling and scavenging for level measurement and hydrogen dilution during normal operation; (4) independent emergency scavenging for plutonium vessels in which radiolysis-related hydrogen buildup can occur following an earthquake, loss of normal instrument air, or loss of power; and (5) super dry air for ventilation and cooling of the AP powder gloveboxes.

The IAS is not an IROFS, but the ESAS is a stand-alone independent IROFS subsystem of the IAS. The ESAS will prevent radiolysis-related hydrogen buildup in process vessels following a loss of normal air supply. The emergency scavenging air supply is automatically activated following a loss of a normal instrument air supply by starting the two parallel air compressors and dryer skids. Each train of scavenging air contains sufficient air to maintain the hydrogen concentration in the vapor spaces of supplied vessels at less than or equal to 1 percent. The IAS contains IROFS, which include moisture sensors, temperature sensors, and redundant and independent seismic isolation valves. These IROFS are necessary to prevent moisture or high-temperature air from entering criticality controlled rooms. Section 11.8.4.3 of this SER evaluates the ESAS.

11.8.3.2.11 Decontamination System

The decontamination system supplies a 3 normal (N) nitric acid solution for the decontamination of certain AP process equipment and lab gloveboxes. The solution will be prepared in a preparation tank and then pumped to the end users.

The decontamination system is not an IROFS, but contains an administrative sampling IROFS. The level in the tank will be measured to ensure that no leakage has occurred, which is part of the sampling IROFS. The transfer pump will be locked out by redundant and independent IROFS instrumentation until the sample is validated.

11.8.3.2.12 Breathing Air System

The breathing air system supplies clean, dry breathing air to the MFFF to support periodic maintenance and emergency usage. The breathing air system contains six quick-connect connections designed to ensure that only breathing equipment can be connected to this system. This system contains a high-energy particular air filter, drain valve, compressors, moisture separator, coolers, piping, and pressure sensors. This system is not an IROFS, but contains redundant and independent seismic isolation valves that are IROFS.

11.8.3.2.13 Radiation Monitoring Vacuum System

The radiation monitoring vacuum system provides a vacuum to draw air from specific areas to the continuous air monitoring detectors at a minimum of 2 cubic feet per minute. This system contains vacuum pumps, filters, nozzles, instrumentation, and piping. This system is designed to minimize the potential for radioactive material buildup by having smooth and wide-radii pipe bends and by using valves that have smooth interiors. This system is not an IROFS and does not contain any IROFS components.

11.8.3.3 Bulk Gas Systems

The bulk gas systems comprise the following:

- nitrogen
- argon/hydrogen
- helium
- oxygen
- methane/argon
- nitrogen tetroxide

Most of these systems are non-IROFS but contain components, such as seismic isolation valves, that are IROFS. The following is a discussion of the bulk gas systems.

11.8.3.3.1 Nitrogen System

The nitrogen system supplies nitrogen to various gloveboxes and other systems and for use as scavenging. Liquid nitrogen can also be used as a backup supply for onsite gaseous liquid nitrogen. Nitrogen will be generated in a separator from ambient air. The system consists of a supply skid, piping, and valves. For normal operations, the nitrogen system ventilates the BMP area gloveboxes, scavenges the sintering furnace airlock, scavenges the calcination furnace, and provides scavenging and bubbling air for the hydroxylamine nitrate (HAN) tanks. For emergency operations, the nitrogen system serves as backup to dry air for the pelletizing press bellows and as backup to the argon/hydrogen system for furnace and airlock scavenging of the BMP sintering furnaces. This system is not an IROFS, but contains redundant and independent seismic isolation valves that are IROFS.

11.8.3.3.2 Argon/Hydrogen System

The argon/hydrogen system supplies an argon/hydrogen mixture for scavenging of the electric sintering furnace and furnace airlocks. The major components of this system are the liquid argon bulk storage system, the hydrogen tube trailer, the backup argon/hydrogen cylinders, inline mixing stations, buffer tanks, pressure transmitters, pressure control valves, alarms, piping, and isolation valves.

There are three levels of backup for the argon/hydrogen system. A multitube trailer of premixed argon/hydrogen gas provides primary backup. This is generally used only to continue operation while the standby portion of the argon/hydrogen system is adjusted and brought online. Secondary backup is provided by 100-percent argon. Tertiary backup is provided by a tie in to the nitrogen system. Local bottles of hydrogen and the argon/hydrogen mix are provided for laboratory use. In addition, argon bottles for emergency purge of both sintering furnaces are provided.

The argon/hydrogen system is not an IROFS, but contains a hydrogen concentration sensor and redundant and independent seismic isolation valves that are IROFS. The hydrogen concentration sensors provide an IROFS isolation function if the argon/hydrogen mixture exceeds a hydrogen concentration of 5 percent. Redundant IROFS isolation valves work in conjunction with hydrogen detectors to isolate the gas mixture in the event of a leak in the room or in the furnace entry/exit airlocks. This isolation function has been determined to be IROFS based on the need to prevent dispersion of nuclear materials from the furnace area resulting from fires or explosions.

11.8.3.3.3 Helium System

The helium system supplies helium for gloveboxes and scavenging for fuel rod welding. The gas will be supplied from tube trailers located outside in the gas storage area. This system also consists of pressure-reducing stations, piping, and valves. This system is not an IROFS, but contains redundant and independent seismic isolation valves that are IROFS.

11.8.3.3.4 Oxygen System

The oxygen system supplies oxygen to the calcination furnace. This system consists of storage cylinders, valves, pressure indicators, switches, and piping. This system is not an IROFS, but contains redundant and independent seismic isolation valves that are IROFS. The fire detection system also sends an IROFS isolation signal to the seismic isolation valves, which are IROFS, if a fire is detected in the furnace room.

11.8.3.3.5 Methane/Argon System

The methane/argon system supplies a reference quenching gas for personal radiation monitors and friskers in the MOX Fuel Fabrication Building (BMF). This system consists of storage cylinders, valves, pressure indicators, and piping. This system will be located in the gas storage area and the laboratory. This system is not an IROFS, but contains redundant and independent seismic isolation valves that are IROFS.

11.8.3.3.6 Nitrogen Tetroxide System

The nitrogen tetroxide (GNO) system supplies nitrous fumes to the AP process for oxidation of plutonium nitrate in the purification cycle (KPA) unit. [redacted]

This system is not an IROFS. However, the GNO system penetrations into the BMF are provided with redundant IROFS seismic isolation valves that are automatically closed upon detection of the seismic monitoring system trip point.

11.8.3.4 Reagent Systems

The reagent systems comprise the following:

- nitric acid
- tributyl phosphate
- HAN
- sodium hydroxide
- oxalic acid
- diluent
- sodium carbonate
- hydrogen peroxide
- hydrazine
- manganese nitrate
- aluminum nitrate
- zirconium nitrate
- silver nitrate
- sodium sulfite
- sodium nitrite
- uranyl nitrate

Most of these systems are non-IROFS but contain components, such as seismic isolation valves, that are IROFS. The following is a discussion of the reagent systems.

11.8.3.4.1 Nitric Acid

The nitric acid system prepares and delivers three concentrations of nitric acid to multiple areas. The nitric acid system provides nitric acid to the AP process units for plutonium dioxide (PuO_2) dissolution, plutonium stripping, acid scrubbing, acidification, and oxalic mother liquor adjustment. This system also provides nitric acid for the preparation of hydrazine, oxalic acid, manganese nitrate, zirconium nitrate, HAN, silver nitrate, and decontamination solution, as well as various other uses.

This system is not an IROFS, but contains redundant and independent seismic isolation valves and an administrative sampling IROFS. The sampling systems for the 6 N nitric acid preparation tank and the 1.5 N nitric acid preparation tank have been determined to be IROFS to ensure that the proper chemical composition of reagents is sent to the AP process. The level in the preparation tanks is monitored after sampling to ensure that no leakage into or out of the tank has occurred. If the results of the analysis are acceptable, the batch is validated and the tank is declared ready for distribution. The mixture is then transferred to a buffer tank from which a circulation pump is used to feed the process users.

11.8.3.4.2 Tributyl Phosphate

The tributyl phosphate system supplies tributyl phosphate for solvent extraction in the purification cycle of the AP process and solvent washing in the solvent recovery cycle. This system's fluid will be transferred from drums into a tank by an air-operated pump. The tanks and pipes are equipped with dip pipes to prevent explosions that could result from static electricity. Equipment suitable for use in an explosive environment will be used. These features will ensure that no electrical ignition source comes in contact with this system's fluid.

The tributyl phosphate system is not an IROFS, but contains redundant and independent seismic isolation valves and an administrative sampling IROFS. The level in the tank will be measured to ensure that no leakage has occurred, and the chemical composition will be verified as part of the sampling IROFS. The fire detection system also sends an IROFS isolation signal to the system pumps and isolation valves in the line if a fire is detected in the AP or the reagent process room.

11.8.3.4.3 Hydroxylamine Nitrate

The HAN system supplies HAN for the stripping of plutonium in the AP purification process. This system's fluid will be transferred from drums into a tank by an air-operated pump. Precautions taken to prevent a HAN explosion include (1) hydrogen peroxide scrubbing and (2) IROFS sampling.

This system is not an IROFS, but contains redundant and independent seismic isolation valves and an administrative sampling IROFS. The sampling systems for the HAN system have been determined to be IROFS to ensure that the proper chemical composition is sent to the AP process. The level in the preparation tanks is monitored after sampling to ensure that no leakage into or out of the tank has occurred. If the results of the analysis are acceptable, the batch is validated and the tank is declared ready for distribution.

11.8.3.4.4 Sodium Hydroxide

The sodium hydroxide system supplies sodium hydroxide to various users for solvent recovery washing and pH control. The system's fluid will be prepared, mixed, and then pumped to a distribution tank. This system is not an IROFS, but contains redundant and independent seismic isolation valves that are IROFS.

11.8.3.4.5 Oxalic Acid

The oxalic acid system supplies oxalic acid for converting plutonium nitrate to plutonium oxalate in the oxalic precipitation, filtration, and calcination unit. This fluid will be prepared, mixed, and then pumped to a distribution tank.

This system is not an IROFS, but contains redundant and independent seismic isolation valves and an administrative sampling IROFS. The sampling systems for the oxalic acid preparation tanks have been determined to be IROFS to ensure that the proper chemical composition of reagents is sent to the AP process. The level in the preparation tanks is monitored after sampling to ensure that no leakage into or out of the tank has occurred and that the concentration is correct. If the results of the analysis are acceptable, the batch is validated and the tank is declared ready for distribution

11.8.3.4.6 Diluent

The diluent system supplies a dodecane isomer mix for washing in the purification and solvent recovery cycles and diluent for preparation of the 30-percent tributyl phosphate solvent solution. An air-operated pump will transfer this fluid from drums into a tank.

This system is not an IROFS, but contains redundant and independent seismic isolation valves and an administrative sampling IROFS. Sampling will be performed to ensure that the flashpoint of the fluid is within acceptable parameters.

11.8.3.4.7 Sodium Carbonate

The sodium carbonate system supplies a sodium carbonate solution for washing the solvent solution. This fluid will be transferred from bags into a tank and mixed with water.

This system is not an IROFS, but contains redundant and independent seismic isolation valves and an administrative sampling IROFS. The level in the preparation tanks is monitored after sampling to ensure that no leakage into or out of the tank has occurred and that the concentration is correct. If the results of the analysis are acceptable, the batch is validated and the tank is declared ready for distribution.

11.8.3.4.8 Hydrogen Peroxide

The hydrogen peroxide system supplies hydrogen peroxide to the AP process for valence adjustment of the dissolution units. This fluid will be transferred from drums into a tank by a pump and then mixed with water.

This system is not an IROFS, but contains redundant and independent seismic isolation valves and an administrative sampling IROFS. The level in the preparation tanks is monitored after sampling to ensure that no leakage into or out of the tank has occurred and that the concentration is correct. If the results of the analysis are acceptable, the batch is validated and the tank is declared ready for distribution

11.8.3.4.9 Hydrazine

The hydrazine system supplies a hydrazine nitrate and nitric acid solution for use in the purification cycle of the AP process. An air-operated pump will transfer this fluid from drums into a tank. Nitrogen blankets are used to prevent formation of explosive mixtures in the vapor space from the vessels in the hydrazine system. This system is not an IROFS and does not contain any IROFS components.

11.8.3.4.10 Manganese Nitrate

The manganese nitrate system supplies manganese nitrate to the oxalic mother liquor recovery unit. This fluid will be transferred from bottles into a tank and then mixed with nitric acid.

This system is not an IROFS, but contains an administrative sampling IROFS. The level in the preparation tanks is monitored after sampling to ensure that no leakage into or out of the tank has occurred and that the concentration is correct. If the results of the analysis are acceptable, the batch is validated and the tank is declared ready for distribution

11.8.3.4.11 Aluminum Nitrate

The aluminum nitrate system supplies aluminum nitrate to the purification cycle for solvent scrubbing. This system is not an IROFS, but contains an administrative sampling IROFS for monitoring concentration.

11.8.3.4.12 Zirconium Nitrate

The zirconium nitrate system provides a zirconium nitrate solution to the AP process to avoid fluoride corrosion of stainless steel vessels in the purification (KPA) cycle and acid recovery (KPC) units by reacting the fluoride with zirconium. This fluid will be pumped from a drum to the preparation tank. This system is not an IROFS, but contains redundant and independent seismic isolation valves and an administrative sampling IROFS. The level in the preparation tanks is monitored after sampling to ensure that no leakage into or out of the tank has occurred and that the concentration is correct. The transfer pump will be locked out by redundant and independent IROFS instrumentation until the sample is validated.

11.8.3.4.13 Silver Nitrate

The silver nitrate system supplies silver nitrate to the electrolyzers in the dechlorination and dissolution unit. Solid silver nitrate, demineralized water, and 13.6 N nitric acid are mixed in a preparation tank in the reagent processing area and pumped to a distribution tank in the AP area. Samples are taken, but they are not IROFS. This system is not an IROFS, but contains redundant and independent seismic isolation valves that are IROFS.

11.8.3.4.14 Sodium Sulfite

The sodium sulfite system supplies sodium sulfite to the dechlorination and dissolution unit for use as a washing solution. This fluid will be mixed with water in a preparation tank and pumped to a distribution tank in the AP area. Samples are taken, but they are not IROFS. This system is not an IROFS, but contains redundant and independent seismic isolation valves that are IROFS.

11.8.3.4.15 Sodium Nitrite

The sodium nitrite system supplies sodium nitrite to the AP process for treatment or destruction of azides in alkaline waste in the aqueous liquid waste reception unit. This fluid will be pumped from a drum to a distribution tank. Samples are performed, but they are not IROFS. This system is not an IROFS and does not contain any IROFS components.

11.8.3.4.16 Uranyl Nitrate

The uranyl nitrate system supplies depleted uranyl nitrate to the plutonium feed stream and uranium waste stream for reducing the isotopic composition of uranium in the plutonium feed material. This fluid will be pumped from a drum to a distribution tank in the AP area. This system is not an IROFS, but contains an administrative sampling IROFS for concentration. If the results of the analysis are acceptable, the batch is validated and the tank is declared ready for distribution.

11.8.4 Evaluation

11.8.4.1 Seismic Isolation Valves

11.8.4.1.1 Description

The primary function of the fluid systems is to provide the safe and reliable handling of the fluids utilized during plant operation, including upset and emergency transients, and maintenance functions.

[redacted]

[redacted]

No utility, gas, or reagent supplied from outside the BMF to a process or component inside the BMF supports an IROFS safety function and is not required to remain in service after a seismic event. A utility, gas, or reagent required to remain in service after a seismic event to support an IROFS function is supplied separately from independent, dedicated sources located inside of the BMF and is completely independent of the external portion of the system.

Seismic isolation valves are classified as FTS Category 1 or FTS Category 2 (Section 11.7.3.1 of this SER discusses FTS categories) and are designed in accordance with the codes and standards identified in Table 11.7-1 of the LA (MOX, 2010a). The applicable codes and standards for FTS Category 1 valves is American National Standards Institute/American Society of Mechanical Engineers (ANSI/ASME) B31.3, “Process Piping” (ANSI/ASME, 1996) (Category M restrictions are applicable) and ANSI/AMSE B31.3 for FTS Category 2.

11.8.4.1.2 Evaluation

The staff reviewed applicable Federal regulations, national codes and standards, NUREG-1718 (NRC, 2000), other industry and NRC staff guidance, and available operational history. The following evaluation discusses the staff's consideration and use of these documents.

As discussed previously, redundant isolation valves are provided to automatically isolate the fluid lines if earthquake conditions are detected. These valves have been designated as IROFS. The valves will be ordered, built, delivered, installed, and used in accordance with the quality level appropriate to each application (as defined by the facility's quality assurance program).

The staff's evaluation of the design basis for the seismic isolation valves consisted of a review of the applicant's proposed design basis against national codes and standards and industry practices. The staff also applied the guidance of NUREG-1718 (NRC, 2000) to the review.

The isolation valves will be designed to passively return to a closed, or isolated, position in the event of a failure of the valve actuator or the air supply or a loss of power. Based on the varying sizes, designs, materials of construction, methods of operation, and fail-safe design, the staff finds that the seismic isolation valves have the redundancy and diversity of components required for isolation and to prevent release of radioactive material.

The applicant will design fluid systems, including the seismic isolation valves and other components, with consideration to the process fluids contained therein and will specify materials with the appropriate corrosion resistance. The applicant will also consider proper surface finishes and decontamination characteristics for their particular application in a system. Based on these considerations and their alignment with the materials and design methodology requirements of the ASME Boiler and Pressure Vessel Code, Section VIII (ASME, 1998), the staff finds this design basis to be acceptable.

The seismic isolation valves will be designed and qualified to maintain functionality when subjected to severe natural phenomena as established in the ISA Summary (MOX, 2010b). The seismic isolation valves will actuate on a seismic event that is one-third of the design-basis earthquake ground motion (i.e., one-third of the safe-shutdown earthquake ground motion). This design basis meets the requirements for nuclear power generation facilities given in paragraph IV(a)(2)(i)(A) of Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." The staff finds this design basis to be acceptable, since the facility design for actuation of the seismic isolation valves is based on the ground motion found to be acceptable at nuclear power generation facilities.

11.8.4.2 Emergency Diesel Generator Fuel Oil System

11.8.4.2.1 Description

[redacted]

[redacted]

11.8.4.2.2 Codes and Standards

The applicant committed to using ANSI/ANS 59.51-1997 (ANSI/ANS, 1997) (endorsed by Regulatory Guide 1.137, "Fuel-Oil Systems for Standby Diesel Generators"). This standard addresses mechanical equipment associated with the fuel oil system. The purpose of this standard is to define those features of fuel oil systems required to ensure an adequate fuel supply to safety-related EDGs and to provide performance and design criteria to ensure that sufficient fuel is available for supply to the EDGs under all plant conditions. This standard applies to this design system with active components. Performance requirements state that the fuel oil system must be designed to maintain its integrity and to remain functional during and after all design-basis events.

The ANSI/ANS 59.51-1997 (ANSI/ANS, 1997) standard also interfaces with ANSI/IEEE Standard 387-1995 and the Class 1E Power systems ANSI/IEEE Standard 308, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations" (IEEE, 1991). This standard requires adherence to the single-failure criterion for light-water reactor safety-related fluid systems, as stated in ANSI/ANS 59.51-1997.

The applicant also committed to meet the requirements of NFPA 30 (NFPA, 1996) and NFPA 37 (NFPA, 1998a).

11.8.4.2.3 Evaluation

The staff reviewed applicable Federal regulations, national codes and standards, NUREG-1718 (NRC, 2000), other industry and NRC staff guidance, and available operational history. The following evaluation discusses the staff's consideration and use of these documents.

The staff reviewed national codes and standards, such as IEEE Standard 308 (IEEE, 1991), NFPA 37 (NFPA, 1998a), and NFPA 110, "Standard for Emergency and Standby Power Systems" (NFPA, 1999b). The staff reviewed the seismic design of the EDGs. The staff also reviewed the design basis and provisions for maintaining a clean, reliable source of fuel oil for the EDGs by comparing the proposed design to standard industry practices and historical diesel generator failure information published by the NRC.

The seismic design of the EDGFOS ensures that these systems will be able to safely shut down during normal operations and accident conditions. Based on the description of the system and the structures that will contain the system, the EDGFOS is adequately designed to maintain functionality when subjected to severe natural phenomena, such as tornadoes, tornado missiles, earthquakes, floods, and any other appropriate phenomena as established in the ISA Summary (MOX, 2010b).

The staff has reviewed the design and configuration of the EDGFOS system tanks and piping. The redundancy and diversity of the system and the seismic design, applied to the design as standard industry practices, will ensure a positive flow of fuel oil into the EDGs. This system will be designed with materials that will be corrosion resistance. The staff finds the design of the system tanks, piping, and pumping systems to be acceptable.

Section 11.5 of this SER presents the staff's review of the electrical capacity of the EDGs.

11.8.4.3 Emergency Scavenging Air System

11.8.4.3.1 Description

The ESAS will be an independent IROFS subsystem of the IAS. The purpose of the ESAS will be to provide scavenging air for dilution of hydrogen that may be produced by radiolysis. The ESAS will be automatically activated following the loss of the normal instrument air supply by starting two parallel air compressors or dryer skids. The ESAS will be designed to maintain the hydrogen concentration in the vapor space of process vessels at less than or equal to 1 percent. Each train of scavenging air contains sufficient air to maintain the hydrogen concentration in the vapor spaces of supplied vessels at less than or equal to 1 percent.

[redacted]

11.8.4.3.2 Codes and Standards

The applicant provided the design-basis codes and standards for the ESAS in Section 11.8.1.1.3 of the LA (MOX, 2010a). That section referenced other portions of the LA which provided the codes and standards for system piping and valves, seismic isolation valves, dryers, receiving tanks, I&C, and the electrical supply.

11.8.4.3.3 Evaluation

The ESAS is an IROFS designed to provide emergency scavenging air for dilution of hydrogen that may be produced by radiolysis. The ESAS is designed to prevent the system from exceeding a 1-percent hydrogen buildup in the tanks. The ESAS is independent of the IAS. In the event of a failure of any of these systems, the ESAS is designed to compensate for the loss of the IAS. A loss of instrument air will automatically activate the ESAS. The pressure and flow switches will close isolation valves to isolate the air system to prevent backflow.

The ESAS is designed to provide two trains, each capable of supplying 100 percent of the scavenging air requirements for 7 days.

The subsystem does not rely on emergency power and will be designed to operate during accident conditions. The staff concludes that, because of the passive nature of the subsystem, it can be safely shut down during any normal or accident conditions. In the event that normal scavenging air is lost (without a seismic event), the ESAS will start and provide scavenging air to the AP process equipment, as necessary, until the normal scavenging air is restored. During a seismic event, the normal scavenging air will be isolated. The ESAS will start on a low/no airflow indication from the normal scavenging air and provide scavenging air to all applicable AP process equipment.

Based on the provision of two 100-percent capacity trains of emergency scavenging air, as well as the design of the system air pressure control to switch supply banks on low system pressure, the staff finds that the subsystem has the adequate redundancy and diversity of components required to prevent release of radioactive material to the environment and that its design is adequate for safe operation of the system. The staff also agrees with the applicant that the ESAS is an independent IROFS because it is not physically connected to the normal scavenging air supply and has a separate nozzle to supply the vessels.

The design of the ESAS falls into FTS Category 2, which is defined in Section 11.7 of this SER. The staff concludes that these design criteria adequately address equipment design to resist corrosion, piping design, and layout.

Therefore, the staff concludes that the equipment will be made of materials with the proper surface finishes and decontamination characteristics for their particular application.

The NUREG-1718 (NRC, 2000) guidance indicates that equipment is to be adequately designed to maintain functionality when subjected to severe natural phenomena as established in the ISA. The ESAS will be seismically designed and qualified in accordance with Regulatory Guide 1.100, Revision 2, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants” (NRC, 1988), and IEEE Standard 344-1987, “IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations” (IEEE, 1987). Because the design conforms to Regulatory Guide 1.100 and the industry standard, the staff concludes that the applicant adequately addressed the features that apply to this IROFS.

11.8.5 Evaluation Finding

The applicant provided design information for the fluid systems that are identified as IROFS. Based on the staff’s review of the LA (MOX, 2010a), ISA Summary (MOX, 2010b), and supporting information provided by the applicant relevant to the fluid systems, the staff concludes that (1) in accordance with 10 CFR 70.61(e), each engineered or administrative control or control system needed to meet the performance requirements is designated as an IROFS and the applicant’s safety program will ensure that each IROFS will be available and reliable to perform its intended function when needed and (2) the design bases of the IROFS evaluated in this SER section will provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents.

REFERENCES

(ANSI/ANS, 1997) American National Standards Institute/American Nuclear Society, ANSI/ANS 59.51-1997, “Fuel Oil Systems for Safety-Related Emergency Diesel Generators,” 1997.

(ANSI/ASME, 1996) American National Standards Institute/American Society of Mechanical Engineers, ANSI/ASME B31.3, “Process Piping,” 1996 including 1998 addenda.

(ASME, 1998) American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section VIII, Division 1, “Rules for Construction of Division 1 Pressure Vessels” (Sections UG-125 through UG-136), 1998.

(MOX, 2010a) Shaw AREVA MOX Services, “License Application,” Aiken, SC, March 2010.

(MOX, 2010b) Shaw AREVA MOX Services, “Integrated Safety Analysis Summary,” Aiken, SC, March 2010.

(IEEE, 1987) Institute of Electrical and Electronics Engineers, Standard 344, “IEEE Recommended Practice for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations,” 1987.

(ANSI/IEEE, 1991) Institute of Electrical and Electronics Engineers, Standard 308, “IEEE Standard Criteria for Class IE Power Systems for Nuclear Power Generating Stations,” 1991.

(NFPA, 1996) National Fire Protection Association, NFPA 30, “Flammable and Combustible Liquids Code,” 1996.

(NFPA, 1998a) National Fire Protection Association, NFPA 37, “Standard for the Installation and Use of Stationary Combustion Engines and Gas Turbines,” 1998.

(NFPA, 1999b) National Fire Protection Association, NFPA 110, “Standard for Emergency and Standby Power Systems,” 1999.

(NRC, 1988) U.S. Nuclear Regulatory Commission, Regulatory Guide 1.100, Rev. 2, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants,” Washington, DC, 1988.

(NRC, 2000) U.S. Nuclear Regulatory Commission NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility,” Washington, DC, 2000.

(IEEE, 1995) Institute of Electrical and Electronics Engineers, Standard 387, “Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations”, 1995.

10 CFR Part 70, Domestic Licensing of Special Nuclear Material.

11.9 <u>Heavy Lift Cranes</u>

11.9.1 Conduct of Review

This chapter of the safety evaluation report contains the staff’s review of the heavy lift cranes for the mixed oxide (MOX) fuel fabrication facility (MFFF), as described by the applicant in Chapter 11.9 of the license application (LA) (MOX, 2009a). The purpose of this review was to determine whether heavy lift cranes are needed to meet the performance requirements and are designated as IROFS. The review will also determine whether heavy lift cranes identified as items relied on for safety (IROFS) will be available and reliable to perform their required safety function(s) when needed (NRC, 2000, Section 11.1).

For this review, the staff evaluated the information provided by the applicant for heavy lift cranes in Chapter 11.0. The staff also reviewed the design and operation of the heavy lift cranes in conjunction with the review of applicable portions of Chapters 4 and 5 of the Integrated Safety Analysis (ISA) Summary (MOX, 2009b), which discuss the crane operations and potential load-handling events.

The staff's safety evaluation and review of the heavy lift cranes examined the safety function(s) they perform; the description of each heavy lifting crane system and its major components; and the safety analysis of potential load-handling accidents, including accident prevention and mitigation of the resulting consequences. The staff used the guidance in NUREG-1718 (NRC, 2000) to perform and prepare its review.

11.9.2 Regulatory Requirements

The regulations applicable to material handling equipment and controls are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 70.61(e), specifically relating to the requirement that each engineered or administrative control or control system that is needed to meet the performance requirements be designated as an IROFS and relating to the safety program that ensures that each IROFS will be available and reliable to perform its intended function when needed

- 10 CFR 70.64, "Requirements for New Facilities or New Processes at Existing Facilities," specifically relating to the application of baseline design criteria and defense-in-depth practices to new facilities or new processes at existing facilities

11.9.3 Regulatory Acceptance Criteria

The review focused on the requirements and guidelines for heavy lift cranes identified as IROFS and related to the baseline design criteria and defense-in-depth principle. The design and operation of heavy lift cranes should fulfill all of the functional requirements determined from the ISA, and the heavy lift cranes should be available with adequate reliability to perform all of their intended safety functions when needed.

Section 11.4.8.2 of NUREG-1718 (NRC, 2000) provides the acceptance criteria related to heavy lift cranes identified as IROFS.

11.9.4 System Description

Heavy lift cranes are defined as those overhead load-handling systems that are designed to lift a load greater than the weight of a single fresh fuel assembly and its associated handling tool (i.e., greater than 1,800 pounds (lb), in accordance with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" (NRC, 1980). Heavy loads are divided into critical and non-critical categories. Critical loads are defined as those loads whose uncontrolled movement or release could result in unacceptable radiological dose consequences to plant workers, the individual outside the controlled area, or the environment. Cranes that handle critical loads are considered Type I. Other loads are considered noncritical, and the cranes that handle these loads are designated as Type II or Type III. Heavy lift cranes in the MFFF that have capacities greater than 1,800 lb are discussed in this section. MFFF cranes not defined as heavy lift cranes but with loads that, if dropped, could result in breaching a confinement system are also evaluated for load-handling events (see Section 11.6 of the SER).

Permanently installed heavy lift cranes are located within the protected structure of the BMF and are protected by the structure from design-basis natural phenomena events, except seismic events. These heavy lift cranes within the MFFF are Type II, which are not required to retain their load during normal operation, upset conditions, or design-basis natural phenomena events. Where heavy lift cranes can move over IROFS equipment, they are qualified to remain structurally intact under design-loading conditions, including the design earthquake. During process operations, material-handling controls protect against load drops over IROFS components.

Based on the above information, the applicant did not identify any MFFF heavy lift cranes that handle critical loads as defined above and stated that no MFFF heavy lift cranes have been identified as an IROFS. The MFFF heavy lift cranes and hoists, which are required to maintain structural integrity in seismic conditions, are designed and qualified in accordance with American Society of Mechanical Engineers (ASME) NUM-1-2000 or ASME NOG-1-1998.

11.9.5 Accident Sequences and Items Relied on for Safety

In general, the IROFS are design features, human actions, or other controls credited with reducing the likelihood of an event or mitigating the consequences of the event. As previously stated, the heavy lift cranes in the MFFF are not identified as IROFS in the ISA Summary and do not perform any critical safety functions.

11.9.6 Evaluation Findings

In Section 11.9 of the LA, the applicant provided design-basis information for the heavy lift cranes in the proposed facility. The applicant stated that “no MFFF heavy lift cranes have been identified as an item relied on for safety.” Based on the staff's review of the LA (MOX, 2009a) and ISA Summary (MOX, 2009b), the staff agrees with this finding and concludes, pursuant to 10 CFR 70.61(e), that regarding the heavy lift cranes of the MFFF, the applicant’s proposed equipment and facilities are adequate to protect health and minimize danger to life or property.

REFERENCES

(MOX, 2009a) Shaw AREVA MOX Services, “Mixed Oxide Fuel Fabrication Facility—License Application,” Aiken, SC, October 2009.

(MOX, 2009b) Shaw AREVA MOX Services, “Mixed Oxide Fuel Fabrication Facility—Integrated Safety Analysis Summary,” Aiken, SC, October 2009.

(NRC, 2000) U.S. Nuclear Regulatory Commission, NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility,” Washington, DC, August 2000.

(NRC, 1980) U.S. Nuclear Regulatory Commission, NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” Washington, DC, July 1980.

11.10 <u>Laboratory Description</u>

The laboratory consists of three major sections:

(1) mixed oxide fuel fabrication facility (MFFF) laboratory
(2) test line (LCT)
(3) 33–millimeter (mm) pneumatic transfer (LLP) system

11.10.1 Conduct of Review

This section of the safety evaluation report (SER) contains the staff's review of the mixed oxide (MOX) process (MP) safety analyses described by the applicant in Section 11.10 of the license application (LA) (MOX, 2010a), with supporting process safety information from Chapters 5, 8, and 11 of the LA, and Section 4.1 of the Integrated Safety Analysis (ISA) Summary (MOX, 2010b). The objective of this review is to determine whether the chemical process safety items relied on for safety (IROFS) and their design bases provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents. The staff coordinated the review of laboratory design bases and strategies with the review of radiation and chemical safety aspects of accident sequences described in the safety assessment of the design bases (see Chapter 5 of this SER), the review of fire safety aspects (see Chapter 7 of this SER), and the review of other plant systems (see other sections of Chapter 11 of this SER).

The staff evaluated the laboratory information in the LA and ISA Summary against the following regulations:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 70.64, "Requirements for New Facilities or New Processes at Existing Facilities," requires that baseline design criteria and defense-in-depth practices be incorporated into the design of new facilities or new processes at existing facilities. With respect to chemical protection, 10 CFR 70.64(a)(5) requires that the MFFF design provide for adequate protection against chemical risks produced from licensed material, facility conditions that affect the safety of licensed material, and hazardous chemicals produced from licensed material. Related to chemical protection, 10 CFR 70.64(a)(3) requires that the facility design provide for adequate protection against fires and explosions, such as those that could be initiated by or involve chemicals at the facility.

The review of the LA and ISA Summary focused on the design basis of chemical process IROFS, their components, and other related information. For each IROFS, the staff reviewed information provided by the applicant for the safety function, system description, and safety analysis. The review also included other design-basis considerations such as redundancy, independence, reliability, and quality. The staff used Chapter 8.0 of NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility" (NRC, 2000), as guidance in performing the review.

As stated in the memorandum of understanding between the U.S. Nuclear Regulatory Commission (NRC) and the Occupational Safety and Health Administration, "Worker Protection at NRC-Licensed Facilities," published in the *Federal Register*, Volume 53, No. 210, dated October 31, 1998, pp. 43950–43951, the NRC oversees chemical safety issues related to (1) radiation risk produced by radioactive materials, (2) chemical risk produced by radioactive materials, and (3) plant conditions that affect the safety and safe handling of radioactive materials. These types of chemical safety issues represent an increased radiation risk to the workers. However, the NRC does not oversee facility conditions that result in an occupational risk but do not affect the safe use of licensed material. The provisions of the memorandum of understanding applicable to the MFFF are codified in 10 CFR 70.64(a)(5).

As described in the following sections, the NRC staff reviewed the following areas of the LA and ISA Summary applicable to process safety:

- MFFF laboratory description
- hazardous chemicals and potential interactions affecting licensed materials
- MFFF laboratory accident sequences
- MFFF laboratory chemical accident consequences
- MFFF laboratory safety controls

11.10.1.1 System Description of the MFFF Laboratory

The MFFF laboratory is primarily used to perform chemical and physical analyses of samples coming from the MP production units, aqueous polishing (AP) production units, and the LCT.

The MFFF laboratory is located in the MP area. The main portion of the laboratory is located on the third floor of the MP building; the other portion is located on the intermediate level of the same building.

Analyses in the lab are performed for the following purposes:

- manufacturing control (process control)
- material control and accountability
- product quality control (specification analyses)
- criticality safety
- process safety
- subsequent waste disposition at the Savannah River Site

The following operations are also performed in the laboratory:

- laboratory liquid and solid waste management
- preparation of reagents used in the MFFF laboratory
- analysis of depleted uranyl nitrate samples
- temporary storage of scrap materials from the MFFF laboratory
- dissolution tests of alternate feedstock plutonium dioxide (PuO_2) powders
- dissolution operations occasionally performed for dissolution (KDB) and dissolution of chlorination feed bag prefilter residues (more than 40 grams (g) of plutonium)
- calibration
- document storage

Samples are transferred from the MP and AP areas to the MFFF laboratory in vials. Samples are transferred between the different analytical units of the laboratory in aliquot containers. An aliquot is a measured portion of a sample taken for analysis. Vials and aliquot containers specific to liquid and solid samples are not reusable. Transfers are either manual or pneumatic; however, most transfers are pneumatic via the LLP or the 76-mm pneumatic transfer system.

11.10.1.2 Test Line

The LCT is used to qualify the pellet fabrication process of the MOX process on a small scale, from powder preparation to grinding and pellet sorting.

The LCT is a small reproduction of the main stages of pellet fabrication. It reproduces the process equipment of the MP from powder preparation to grinding and pellet sorting. The primary processes executed in the LCT are micronizing powder, sieving powder, dosing powder, mixing powder, pressing powder into pellets, grinding sintered pellets, sampling and manually sorting pellets, and carrying out quality control measurements. The LCT is also used to determine physical powder characteristics (i.e., density, grain-size distribution, and flowability).

Powder packaged in vials within bags is transferred into the milling glovebox through a glove port. Four types of powder can be milled in this glovebox: primary blend (master blend), final blend, scrap powder, and PuO_2 powder. Additives (pore-former and lubricant) are also introduced into this glovebox.

Vials are weighed and identified before being emptied into a bowl. The bowl is docked and locked onto the ball mill drum. Milling is accomplished by rotating the drum about itself while inclining the whole system. The bowl is locked onto the drum during rotation and inclination, which facilitates the flow of powder into the drum. The milled powder is collected in a pot and transferred through the tunnel from the milling glovebox to the forced sieving glovebox where powder is forced through a perforated grid with a swing arm.

The powder is transferred in a vial through a tunnel to the dosing and mixing glovebox where the desired amount of each powder, including additives, is weighed and poured into a vial. The vial is attached to the mixer and mechanically shaken.

Also in the dosing and mixing glovebox, powder grain size distribution is determined by mechanical sieving, bulk and tap density are measured, and powder flowability is tested.

Mixed powder from the dosing and mixing glovebox is transferred to the pelletizing press through a tunnel and poured into the press hopper. With the aid of an impactor, the hopper fills the press shoe with powder. The powder is pressed into pellets that are manually removed and placed into small molybdenum boats.

The density of the pellets is measured before they are transferred out of the LCT unit to a furnace (PFE or PFF unit) for sintering. After sintering, the pellets are returned to the LCT to the grinding glovebox. Pellets are manually placed into a vibrating bowl that supplies the centerless grinder and then ground. This glovebox is also used for pellet quality control measurements and visual checking.

A pneumatic transfer departure station in this glovebox transports pellet samples to the laboratory for analysis.

The dust collection system uses a vacuum pipe in the protective cover of the grinding wheels and the glovebox atmosphere to control dust in the grinding glovebox.

A shielded cask on a hand-pushed trolley is used to transfer samples and products (pellets and powder) between the production stations and the LCT.
The LCT normal process inventory is 263 g of plutonium.

11.10.1.3 Pneumatic Transfer Line System

The LLP system transports liquid, pellet, or powder samples between AP liquid automatic and manual sampling points, analytical units, and the LCT.

The LLP system comprises pneumatic transfer tubes with a diameter of 33 mm that connect gloveboxes within the same or different units. The primary process executed by the LLP system is the transfer of sample vials. The LLP system also provides transport of empty vials to the sampling (KPG) unit and laboratory units.

The LLP is divided into nine separate networks as described in the following paragraphs. Each vial that is used in the LLP system has the external shape of a shuttle. Therefore, no shuttle is required for vial transfer. For transfer, a vial is inserted in one end of a tube, and the tube is closed. Transfer of the vial is enabled by an exhauster, which creates negative pressure within the tube between the vial and its destination. Switching devices, as required, direct the vial through the correct tube to the destination arrival station. Upon arrival at the destination, after the exhauster has been turned off, the vial is removed from the tube.

Network 1—AP Automatic Liquid Sampling Points to Laboratory

This network transports liquid samples from the automatic liquid sampling system of the AP process KPG gloveboxes to two laboratory analysis lines or to temporary storage. The two laboratory analysis lines are (1) high-plutonium content solution analysis and preparation (KLA) and low-plutonium content solution analysis and preparation (KLB). The temporary storage is AP liquid samples storage (KLG).

A separate line brings the empty vials from the vial delivery system to the KPG gloveboxes and to the room of the inactive local sampling (ILS) and active local sampling (ALS) departure stations.

The KPG departure stations receive sample vials filled by the automatic sample needles. A robot takes an empty vial in the arrival station and presents it to the filling head that is connected to the sampling needle. Then, the robot brings the vials into the cleaning, rinsing, and sending stations (the cleaning station is part of the KPG unit).

Network 2—AP Manual Liquid Sampling Points to the Laboratory

This network transports liquid samples from the ALS departure station to laboratory analysis line KLA. Network 2 also transports samples from the ILS departure station to the arrival point of the ILS unit in the laboratory area. An operator brings empty vials from the vial delivery system (Network 1) to the manual ILS sampling points and ALS sampling points. The ILS departure and arrival stations are located outside the gloveboxes.

Networks 3, 4 and 5—MP Laboratory Links

These networks transport samples between the MP laboratory lines (Networks 3 and 4). Network 3 also transports samples from the LCT to the receipt, weighing, and dispatching (LRD) unit. Network 3 samples contain MOX pellets, and Network 4 samples contain MOX powder. Network 5 transports samples between the MP laboratory stations for AP powder analysis.

Network 6—Liquid Analysis Laboratory

This network transports liquid samples between the liquid samples analyses laboratory stations and provides links between the MP and AP laboratory stations.

A manual station supplies two departure stations with empty vacuum-packed vials directly from the supplying device. The two departure stations are lines KLA and PuO_2 dissolution (KLE).

In the glovebox, a liquid sample is taken and placed in a vial. The empty, vacuum-packed vial that will be transferred in the pneumatic transfer system stands in the rotary sender/receiver. The filled vial containing the liquid sample is stuck by a double needle with the vacuum-packed vial. The pressure differential between the filled vial and the vacuum-packed vial transfers the sampled liquid to the vacuum-packed vial. The vacuum-packed vial is then ready for transfer.

Network 7—Spectrometer Supplying

This network supplies the two mass spectrometers in the mass spectrometer (LSR) unit with liquid preparations from the AP and MP preparation lines.

Network 8—AFS/PDCF Analysis

This network supplies the analytical units with alternate feedstock (AFS)/pit disassembly and conversion facility (PDCF) powders.

[redacted]

Network 9—AFS/PDCF Liquid Analysis and Liquid Waste Transfer

This network transfers AFS/PDCF powder solutions from the KLI1 line to the KLI2 line. The network also transfers the liquid preparation from the dissolution units (KLI) to the three spectrometers in the metallic impurities determination (KLJ) unit. The network transfers liquid waste from the KLI units to the powder density measurements and laboratory electrolyzers (KLO) unit. A manual station supplies the KLI1 and KLI2 departure stations with empty vacuum-packed vials directly from the supplying device.

In the dissolution enclosures (in the KLI unit), the liquid preparation is placed in a vial. The empty, vacuum-packed vial that will be transferred in the pneumatic transfer system stands in the rotary sender/receiver departure station. The filled vial is stuck by a double needle with the empty vacuum-packed vial. The pressure differential between the filled vial and the vacuum-packed vial transfers the sampled liquid to the vacuum-packed vial. The vacuum packed vial is then ready for transfer.

11.10.1.4 Laboratory Liquid Waste Receipt Unit

The laboratory liquid waste receipt (LGF) unit receives and manages the liquid wastes generated by the MFFF laboratory for subsequent recycling in the KDB unit, and for transfer to the aqueous liquid waste reception (KWD) unit or solvent liquid waste reception (KWS) unit.

Liquids recycled into the AP process via the KDB unit are not generally classified as liquid wastes. However, to facilitate the description of the processes in the LGF unit, they are referred to as liquid wastes throughout this section.

The LGF unit is an ancillary unit located in the AP process area of the MFFF. Electrical power and control equipment of the normal control system are installed in separate rooms.

[redacted]

[redacted]

[redacted]

[redacted]

Process Chemistry

The addition of reagents (i.e., aluminum nitrate) to the plutonium-receiving tank for fluorides complexation and/or 13.6 N nitric acid for acidity adjustment is infrequent and involves only small volumes. [redacted]

[redacted]

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
LGF TK 4000	230	[redacted]	[redacted]	[redacted]	[redacted]
LGF TK 5000	300				
LGF TK 6000	1,500				
LGF TK 7000	1,500				

Table 11.10-2 Chemical Quantities in LGF Unit during PDCF Operations

Item	Volume (L)	HNO_3 (moles)	$AgNO_3$ (moles)	TBP (L)	Diluent (L)
LGF TK 1000	190	522.5	2.9×10^{-3}		
LGF TK 2000	190	522.5	2.9×10^{-3}		
LGF TK 3000	100			22	78
LGF TK 4000	230	115			
LGF TK 5000	300	150			
LGF TK 6000	1,500	750			
LGF TK 7000	1,500	750			

11.10.2 Staff Review of Laboratory Safety

11.10.2.1 Laboratory Explosion (EXP 14)

The applicant identified two potential explosion events in the MFFF laboratory. One of the explosion events is related to hydrogen, and the other is produced by an unintended chemical reaction.

11.10.2.1.1 LAC Laboratory Explosion (EXP 14a)

The hydrogen explosion event is related to a test that determines the oxygen-to-metal ratio of MOX and LCT pellets. The test is carried on in a tubular electrical furnace in a glovebox of the fluorine and chlorine determination and oxygen-to-metal ratio determination (LAC) unit. The furnace is heated and fed with argon to perform the test. Once the furnace is at the operating temperature, pellet samples undergo oxidation using air as the scavenging gas. The furnace is purged with argon, and then fed with an argon-hydrogen (Ar-H) mixture for further reduction to obtain normal pellet stoichiometry. The operating temperature in the furnace is high enough to support spontaneous ignition of hydrogen in the presence of air. [redacted]

[redacted]

11.10.2.1.2 General Laboratory Explosion (EXP 14b)

[redacted]

11.10.3 Evaluation Findings

The NRC staff finds the descriptions of the MFFF laboratory to be adequate to facilitate an understanding of the operations and possible hazards. The safety strategy for the explosion scenarios satisfies the single-failure criterion. The single-failure criterion, in combination with management measures for EACs (as described in Chapter 15 of the LA) the, quality assurance requirements (as described in the MOX Project Quality Assurance Plan (MPQAP)), and the use of codes and standards for engineered IROFS give the NRC staff reasonable assurance that these high-consequence scenarios are highly unlikely. Therefore, the proposed safety strategy and IROFS comply with the performance requirements of 10 CFR 70.61, the baseline design criteria in 10 CFR 70.64(a)(3) and 10 CFR 70.64(a)(5), and the defense-in-depth practices in 10 CFR 70.64(b).

REFERENCES

(MOX, 2010a) Shaw AREVA MOX Services, "Mixed Oxide Fuel Fabrication Facility License Application," Aiken, SC, March 2010.

(MOX, 2010b) Shaw AREVA MOX Services, "Mixed Oxide Fuel Fabrication Facility Integrated Safety Analysis Summary," Aiken, SC, March 2010.

(NRC, 2000) U.S. Nuclear Regulatory Commission, NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility,” Washington, DC, August 2000.

10 CFR Part 70, Domestic Licensing of Special Nuclear Material.

Memorandum of Understanding Between the Nuclear Regulatory Commission and the Occupational Safety and Health Administration: Worker Protection at NRC-licensed Facilities, October 21, 1988, 53 FR 43950.

11.11 Civil Structural Systems

This section of the safety evaluation report (SER) contains the U.S. Nuclear Regulatory Commission (NRC) staff’s review of the civil structural systems for the Mixed Oxide Fuel Fabrication Facility (MFFF), as described in Section 1.1.2.1 of the license application (LA) (MOX, 2010a). This review will determine whether the design bases and design of the items relied on for safety (IROFS) provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents. The staff evaluated the information the applicant provided for the civil structural systems by reviewing Chapter 1 of the LA, supplementary information the applicant provided, and relevant design documents available at the applicant’s offices but not submitted by the applicant. The staff closely coordinated its review of the design bases and design of the civil structural systems with the review of the civil structural aspects of accident sequences described in the safety assessment of the design bases (see Chapter 5 of this SER).

11.11.1 Regulatory Requirements

The staff reviewed how the information in the LA addresses the following regulations:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 70.61(e), “Performance requirements.“ requires that each engineered or administrative control or control system that is needed to comply with 10 CFR 70.61 (b), (c) or (d) be designated as an IROFS and requires that the safety program ensuring that each IROFS will be available and reliable to perform its intended function when needed.

- 10 CFR 70.64, “Requirements for New Facilities or New Processes at Existing Facilities,” requires that new facility designs incorporate baseline design criteria and defense-in-depth practices. With respect to natural phenomena hazards (NPHs), 10 CFR 70.64(a)(2) requires that new facility designs adequately protect against such hazards, considering the most severe documented historical events for the site.

11.11.2 Regulatory Acceptance Criteria

This LA review focused on the design basis and design of the civil structural systems and other related information. For each civil structural system, the staff reviewed information the applicant provided for the safety function, system description, and safety analysis. The staff used Section 11.4 in NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility” (NRC, 2000), as guidance in performing the review.

The staff reviewed the following four general areas of LA Section 1.1.2, "General Facility Description:"

(1) classification of civil structural systems
(2) codes and standards
(3) structural design criteria
(4) structural analysis and design

11.11.3 System Description

The civil structural systems for the MFFF include the buildings, support structures, and facilities that house, support, confine, or contain various plant systems, components, and equipment associated with licensed nuclear materials, or hazardous chemicals associated with licensed nuclear materials, as well as support buildings.

11.11.3.1 Function

As described in LA Section 1.1.2.1.1, the safety functions for the civil structural systems would do the following:

- Support the IROFS during normal, severe, and extreme loading conditions.
- Provide confinement functions as part of secondary and tertiary confinement systems.
- Protect IROFS from the effects of normal, severe, and extreme environmental loads.
- Protect IROFS from the effects of design-basis internal and external fires by providing fire barriers.

11.11.3.2 Major Components

LA Section 1.1.2.1.2 identified the seismic Categories (SCs) I and II or conventional seismic (CS) civil structural systems. The major components for each of the seismic categories are as follows:

- SC-I structures include the MFFF (which includes the mixed oxide (MOX) processing area, the aqueous polishing area (BAP), and the shipping and receiving area (BSR)), the emergency fuel storage vault (UEF), and the emergency diesel generator building (BEG).
- SC-II structures include the safe haven buildings.

The remaining structures are CS structures. LA Section 1.1.2.1.3 contains a detailed description of each major component..

11.11.3.3 System Interfaces

Civil structural systems interface with the site and all facility systems because they protect and support structures, systems, and components (SSCs).

11.11.3.4 Classification of Civil Structural Systems

The classification outlined in LA Section 1.1.2.1.2 consists of three levels: (1) SC-I, (2) SC-II, and (3) CS structures. The design loadings considered for the civil structures in each category are as follows:

(1) SC-I—normal, severe, and extreme environmental loads, including the design-basis earthquake and tornado

(2) SC-II—normal, severe, and design-basis earthquake

(3) CS—normal, severe, and CS loads, as specified by the Uniform Building Code

11.11.3.5 Design Basis of the SC-I and SC-II Structures

11.11.3.5.1 Codes and Standards

This section contains a review of LA Section 1.1.2.1.6.3, “Codes and Standards for SC-I Structures,” and Section 1.1.2.1.7.7, “Codes and Standards for SC-II Structures,” of the LA.

The designs of the SC-I civil structural systems and the associated steel and concrete components conform to standard engineering practices. LA Section 1.1.2.1.6.3 contains a comprehensive list of the applicable codes and standards. The applicant would supplement American Concrete Institute (ACI) 349-97, “Code Requirements for Nuclear Safety-Related Concrete Structures” (ACI, 1997), and American Institute of Steel Construction (AISC) N690-1994, “Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities” (ANSI/AISC, 1994), with specific provisions. Section 1.1.2.1.6.2.2 of the LA contains more specific concrete, steel, and foundation structural design requirements for SC-I structures.

In LA Section 1.1.2.1.6.3, the applicant referenced the same codes and standards as those for the SC-I structures for the design of the SC-II civil structural systems, except for ACI-SP-175-98, “Concrete and Blast Effects” (ACI, 1998); ACI 349-97 (ACI, 1997); ACI 349.1R-91, “Reinforced Concrete Design for Thermal Effects on Nuclear Power Plant Structures” (ACI, 1996); and AISC N690-1994 (ANSI/AISC, 1994).

However, the applicant included ACI 318-99, “Building Code Requirements for Structural Concrete” (ACI, 1999), for SC-II civil structures. Furthermore, it used older versions of certain documents for the SC-II civil structural systems. It used American Society of Civil Engineers (ASCE) 8-90, “Specification for the Design of Cold-Formed Stainless Steel Structural Members, (ASCE, 1991)” instead of ASCE 8-91 (ASCE 1991), and the 1986 version of the American Iron and Steel Institute (AISI), “Specifications for the Design of Cold-Formed Steel Structural Members,” instead of that from 1996 (AISI, 1986).

The staff reviewed the codes and standards for the designs of SC-I and SC-II civil structural systems and concluded that the cited codes and standards are consensus standards that provide reasonable guidance consistent with the categorization assigned to the buildings and are consistent with Section 11.4.6.1 of NUREG-1718 .

11.11.3.5.2 Structural Design Criteria

This section reviews the structural design criteria and load combinations for the civil structural systems discussed in the LA.

Section 1.1.2.1.6.4, “Design Values for SC-I Structures,” and Section 1.1.2.1.7.8, “Design Values for SC-II Structures,” of the LA discussed the structural design loads for SC-I and SC-II structures. The applicant divided the design criteria and loads anticipated for the civil structures into three categories: (1) normal loads, (2) severe environmental loads, and (3) extreme environmental loads. The normal loads include dead, live, hydrostatic fluid pressure, lateral soil pressure, thermal, and component reaction loads. The severe environmental loads include wind and flood loads. The extreme environmental loads include seismic, tornado, and explosive loads and post-earthquake settlements. The only extreme environmental load considered in the design of the SC-II structures was the seismic load. The staff finds that the applicant’s evaluation satisfies the requirements of 10 CFR 70.64(a)(2), which states that structures must be designed to adequately protect against natural phenomena, considering the most severe documented historical events for the site.

11.11.3.5.2.1 Normal Loads

Dead Loads

Dead loads are gravity loads induced by the mass of the structure, permanent equipment, and any permanent hydrostatic loads with constant fluid levels. This definition for dead loads is consistent with ASCE 7-98, “Minimum Design Loads for Buildings and Other Structures” (ASCE, 1998), and is acceptable to the staff.

Actual equipment loads will be applied to the design of structural systems and components with a minimum uniform dead load of 2.4 kilopascal (kPa) (50 pounds per square foot (psf)) applied to each wall panel, the underside of elevated floor slabs and roof slabs, and platforms. The LA also indicates that the effect of differential settlement is considered in determining dead loads.

Live Loads

Live loads are loads produced by building use and occupancy. The live loads considered for the civil structures of the MFFF include floor, rain, snow and ice, transportation vehicle, and heavy floor, as well as crane, monorail, hoist, and elevator loads.

Floor Live Loads

The applicant established the minimum uniformly distributed live loads for the civil structures, in accordance with ASCE 7-98 (ASCE, 1998). Specifically, the floor live loads identified include the following:

Platform and work area	6.0 kPa (125 psf)
Light storage	6.0 kPa (125 psf)
Heavy storage	12.0 kPa (250 psf)
Heavy operation	12.0 kPa (250 psf)
Office	4.8 kPa (100 psf)
Computer room	7.2 kPa (150 psf)
Dining/meeting rooms	4.8 kPa (100 psf)
Laboratory	9.6 kPa (200 psf)
Toilet areas	4.8 kPa (100 psf)

Mechanical (utility) rooms	7.2 kPa (150 psf)
Electrical rooms	7.2 kPa (150 psf)
Stairs, fire escapes, and corridors	4.8 kPa (100 psf)
Roof	2.4 kPa (50 psf)
Transportation vehicle loads	14.4 kPa (300 psf) or forklift truck of 26.7 kilonewton (6 kips) capacity

The staff reviewed the design-basis floor live loads discussed in Section 1.1.2.1.6.4.1.1 of the LA and determined that the floor live loads and the roof loads are acceptable for the design of the facility civil structures.

Rain Loads

The applicant determined the design-basis rain loads for the civil structures in accordance with the requirements of ASCE 7-98 (ASCE, 1998). The design rain load for the roof system of the SC-I and SC-II structures is 2.4 kPa (50 psf), which is equivalent to more than 24.4 centimeters (cm) (9.6 inches (in.)) of standing water on the roof because of deflection of the roof or blockage of the primary roof drains. The LA further states that "parapets or other structures, which could potentially contribute to significant ponding, are not used on the roofs of SC-I structures" and that the rain load does not combine with the roof live load in the load combinations. The staff reviewed the design-basis rain load and determined that it is appropriate and acceptable.

Snow and Ice Loads

The applicant determined the design-basis value of snow and ice loads to be 0.48 kPa (10 psf). The applicant estimated this value based on the 100-year maximum ground snow and ice loads. The importance factor for the value of snow and ice loads is 1.2, found in ASCE 4-98, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary" (ASCE, 1999). The staff reviewed the applicant's design-basis value of ground snow and ice loads and found that it was based on acceptable methods from the requirements of ASCE.

Transportation Vehicle Loads and Heavy Floor Loads

The design-basis load for transportation vehicular truck traffic in designated building areas was determined in accordance with the standard loadings defined by the American Association of State Highway and Transportation Officials in its document, "Standard Specifications for Highway Bridges," issued in 1996 (AASHTO, 1996). The wheel loading design used the minimum truck loading of HS 20-44. The staff reviewed the design-basis transportation vehicle loads and heavy floor loads and determined that they are appropriate and acceptable.

Crane, Monorail, Hoist, and Elevator Loads

These design loads apply to structural members and components to support permanently installed cranes, monorails, hoists, and elevators. Section 1.1.2.1.6.4.1.1 of the LA states that the design-basis crane, monorail, hoist, and elevator loads would envelop the full-rated capacity of the cranes, monorails, hoists, and elevators, including impact loads and test load requirements. The effects of a crane load drop were evaluated in accordance with the guidance provided in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" (NRC, 1980). The staff reviewed the design-basis crane, monorail, hoist, and elevator loads and found that these design-basis loads are appropriate and acceptable.

Hydrostatic Fluid Pressure Loads

The LA indicates that hydrostatic fluid pressure loads are limited to containment curbs to contain postulated spills from postulated flooding of the pipe tunnel in the reagents processing building (BRP). The applicant classified the BRP as a CS structure. The staff reviewed the information presented and found that the consideration of hydrostatic fluid pressure loads for the design of civil structural systems is acceptable.

Lateral Soil Pressure Loads

Section 1.1.2.1.6.4.1.1 of the LA indicates that the lateral soil pressure loads on structures, elements of structures, or both, because of retaining soil, are determined based on the density of the soil and any surcharge load, plus the hydrostatic pressure caused by ground water or soil saturation.

The minimum lateral soil pressure loads on structures or elements of structures resulting from retaining soil are in accordance with those defined in ASCE 7-98 (ASCE, 1998). Earthquake-induced soil pressure on structures or embedded wall design is developed in accordance with ASCE 4-98 (ASCE, 1999). No hydrostatic pressure is expected, because the ground water table at the site is below the MFFF.

The staff reviewed the applicant's approach for determining lateral soil pressure loads and found that it is acceptable, because it is based on ASCE national standards.

Thermal Loads

Thermal loads consist of thermally induced forces and moments on the structural components of buildings. These loads would result from operating and environmental conditions. The thermally induced loads would be design dependent. Consequently, determination and consideration of these thermally induced loads would be an integral part of a design. For the design of civil structural systems, the applicant considered the effects of thermal expansion loads caused by axial restraint of the structural components, as well as loads resulting from thermal gradients. The applicant also indicated that it determined these thermally induced loads based on the most critical transient or steady-state condition. The staff reviewed the information and found that the applicant's consideration of thermal expansion loads and thermal gradient loads for the design of civil structural systems is consistent with national codes and standards and is acceptable.

Equipment Reaction Loads

The equipment reaction loads included those from pipes; heating, ventilation, and air conditioning (HVAC) ducts; conduits; and cable trays. These loads would be design dependent and are to be assessed during design. The applicant stated that it would determine the equipment reaction loads based on the most critical transient or a steady-state condition. The applicant further indicated that it would ensure that the final designs envelop the actual equipment reaction loads. The design allowance for the equipment reaction loads is a minimum uniform dead load of 2.4 kPa (50 psf) for each wall panel, the underside of elevated floor slabs and roof slabs, and platforms. The staff reviewed the information on equipment reaction loads in the design and found that the bounding or enveloping approach is acceptable.

11.11.3.5.2.2 Severe Environmental Loads

Wind Loads

ASCE 7-98 (ASCE, 1998, Figure 6-1b) identifies a design-basis wind of 161 kilometers per hour (km/h) (100 miles per hour (mph)) for the region. Information provided in Table 1.1.2-2 of the LA for the Savannah River Site (SRS) identifies a design-basis wind of 209 km/h (130 mph), which is higher than the value provided in ASCE 7-98. The LA also indicates that the wind loads calculated based on the design-basis wind are determined using the procedures provided in ASCE 7-98. The approach for determining the wind loads is acceptable to the staff because it is the same or similar to consensus standards the staff has previously approved.

The applicant also considered windborne missiles in the design of the civil structural systems. Table 1.1.2-2 of the LA contains the windborne missile criteria. Considering the effects of windborne missiles in the design is consistent with ASCE 7-98, which requires the inclusion of windborne debris in areas where the basic windspeed is equal to or greater than 193 km/h (120 mph). The inclusion of windborne missiles in the design is acceptable to the staff because it meets the guidance provided in ASCE 7-98.

Flood Loads

The maximum probable flood level for the site is at elevation 68.4 meters (m) (224.5 feet (ft)) above mean sea level, and the design-basis flood for the MFFF site for the annual recurrence frequency of 1×10^{-5} is at a water level of 63.4 m (207.9 ft) above mean sea level. The corresponding site grade level is approximately 83 m (272 ft) above mean sea level. Because the site grade level is much higher than the maximum probable flood level at the site, the facility is a flood-dry site and will be free from the adverse effects of the maximum probable flood. Consequently, the design of civil structural systems does not need to consider the loads resulting from floodwater. The applicant based its analysis of the maximum probable flood level on the surface hydrology of the region and the potential dam failure resulting from seismic events given in Section 1.3.4.2, "Floods," of the LA.

The staff reviewed the flood load discussion and concluded that the facility design is consistent with the design criteria of Regulatory Guide 3.40, "Design Basis Floods for Fuel Reprocessing Plants and for Plutonium Processing and Fuel Fabrication Plants," issued December 1977 (NRC, 1977). The staff also found that the approach used for conducting the flood analysis is consistent with that outlined in Section 5.1.3 of American National Standards Institute/American Nuclear Society (ANSI/ANS) 2.8-1992, "Determining Design Basis Flooding at Power Reactor Sites" (ANSI/ANS, 1992), which determined design-basis flooding at power reactor sites. The use of standards for design basis flooding at power reactors is acceptable to the staff for the MFFF.

11.11.3.5.2.3 Extreme Environmental Loads

Seismic Loads

Table 1.1.2-2 of the LA lists the design-basis earthquake ground motions based on probabilistic seismological studies specific to the SRS. The applicant developed its design-basis horizontal and vertical response spectra for the facility based on Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," issued December 1973 (NRC, 1973a)—horizontal and vertical spectrum shapes anchored at a peak ground acceleration of 0.20g. Section 1.3.1.5.5 of this SER contains the applicant's detailed design spectra.

The applicant determined the design seismic loads for the facility's SC-I structures by first conducting soil-structure interaction analyses using the design-basis earthquake accelerations in the three orthogonal directions (two horizontal and one vertical). The analyses of the soil-structure interaction used the Framatome ANP version of the SASSI computer code described in the LA. The applicant then calculated the design seismic loads for the various structural elements in the three directions through static analyses of three-dimensional finite-element structural modeling, using the maximum floor accelerations developed from the three-dimensional soil-structure analysis. Subsequently, the applicant determined the resultant design seismic loads (forces and moments) on each structural element by combining the design seismic loads for the structural element from the static analysis in three directions using the 100-40-40-percent rule, as described in Section 3.2.7.1.2 of ASCE 4-98 (ASCE, 1999). Section 11.1.1.3.3 of this SER contains further details of the soil-structure interaction analyses.

For other structures, the applicable seismic response was applied to the base of the finite element models. The applicant used guidance in Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," issued February 1976 (NRC, 1976), and ASCE 7-98 (ASCE, 1998) to combine modal responses and collinear responses from the individual earthquake components.

The staff reviewed the approach the applicant used to determine the seismic loads and found the approach to be acceptable. Based on the in office review (NRC, 2007a and NRC, 2007b) of the method for determining the seismic loads for the MOX fuel fabrication building (BMF), the staff found the applicant implemented the ASCE 100-40-40-percent rule appropriately to calculate the resultant design seismic loads.

<u>Tornado Loads for SC-I Structures</u>

Table 1.1.2-2 of the LA provides the design-basis tornado windspeed, atmospheric pressure change, and rate of pressure drop. In determining design tornado loads, the applicant used the procedure provided in ASCE 7-98 (ASCE, 1998).

The staff reviewed the applicant's approach and found it to be acceptable, based on the guidance provided in ASCE 7-98. The three types of tornado loads on the facility structures are described below.

<u>Tornado Wind Pressure Loads</u>

Table 1.1.2-2 of the LA defines the tornado windspeed. The applicant used Section 6 of ASCE 7-98 (ASCE, 1998) to convert the tornado wind velocity into effective structural pressure loads. ASCE 7-98 is an industry consensus standard that the staff has accepted. Therefore, its usage for converting the tornado wind velocity is acceptable to the staff.

<u>Tornado-Created Differential Pressure Loads</u>

Table 1.1.2-2 of the LA contains the definition of the tornado-created differential pressure loads. The applicant determined these pressure loads based on guidance provided in Section 3.3.2 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (NRC, 1993), and, therefore, they are acceptable to the staff.

<u>Tornado-Generated Missile Loads</u>

Table 1.1.2-2 of the LA presents the design basis for tornado-generated missiles. Consistent with NUREG-0800 (NRC, 1993), three objects are postulated in determining design-basis tornado-generated missiles: (1) a massive high-kinetic energy missile, which deforms on impact, (2) a rigid missile to test penetration resistance, and (3) a small rigid missile of a size sufficient to pass through any openings in protective barriers.

The applicant selected the design-basis tornado-generated missiles based on DOE-STD-1020-94, “Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities” (DOE, 1996a). The applicant used the ASCE Manual and Report No. 58, “Seismic Analysis of Safety-Related Nuclear Structures and Commentary” (ASCE, 1980), to determine the tornado impact loads on buildings, structures, and facilities. The staff reviewed the information provided for the design criteria on tornado-generated missiles and found it acceptable.

Explosive Loads for SC-I Structures

Section 1.1.2.1.6.4.1.3 of the LA indicates that the applicant used the results of SRS facility explosions, SRS transportation explosions, MFFF transportation hypothetical explosions, and hypothetical explosions of the BRP to determine the impact of the bounding explosion on the SC-I structures. The Integrated Safety Analysis (ISA) Summary (MOX, 2010b) identified two bounding explosion events: hydrogen and transportation explosions. The staff also reviewed the documents related to external explosion analysis and design during an onsite review and found that the applicant used the approach suggested in “Explosion Hazards and Evaluation,” by Baker, et al. (Baker, 1983) to estimate the effective explosive yield (explosive load). The applicant used Technical Manual, TM 5-1300, “Structures To Resist the Effects of Accidental Explosions” (DOD, 1969), to assess structural system responses as it committed to do in the LA.

The staff reviewed the information related to explosive loads and found the applicant’s determination of these loads and the approach it used to assess structural system responses appropriate and acceptable.

Settlements

Settlements at the site come from two potential sources—(1) compaction of soft soil materials, including soft zones, and (2) localized liquefaction. The applicant treated the potential effects of the soil compaction as a part of dead loads and considered the settlement effects associated with liquefaction in the design as an extreme environmental load.

The primary settlements estimated by numerical modeling ranged from 5.1–7.1 cm (2.0–2.8 in.) from DCS01-WRS-DS-CAL-G-00017-D, “Estimates of Static Settlement of MFFF Structure” (DCS, 2005a), and the MFFF LA. Including the secondary consolidation, the total estimated settlements of the BMF are approximately 6.4–8.4 cm (2.5–3.3 in.) with the differential settlement varying from 0.5–1.8 cm (0.2–0.7 in.) (DCS, 2005a).

The staff reviewed the information presented regarding the applicant’s settlement analysis and determined that the approach for estimating structural settlements is acceptable because it is based on current analysis techniques.

The applicant estimated the post-earthquake dynamic settlement of the potentially liquefiable soil based on the PC-3+ design-basis ground motion and the 1886 Charleston motion in DCS01-WRS-DS-NTE-G-00005-E, "MOX Fuel Fabrication Facility Site Geotechnical Report" (DCS, 2003). The settlement ranges from 0.66–3.73 cm (0.26–1.47 in.) for the design-basis ground motion and 1.55–5.64 cm (0.61–2.22 in.) for the 1886 Charleston motion. The applicant indicated that these dynamic settlements might occur in loose or soft strata below the ground water level, at a depth of 18.29 m (60 ft) or more. There are two significantly stiffer soil layers more than 12.19 m (40 ft) thick between the potentially liquefiable zones and the foundations. These two soil layers would tend to redistribute the estimated dynamic settlement such that no significant differential settlement would occur at the foundation level (DCS, 2003).

The staff reviewed the dynamic settlement information and concurred that the post-earthquake-induced dynamic settlements resulting from localized liquefaction would not create stability problems for the foundations for the SC-I structures.

Aircraft Crash Hazard

As discussed in Section 1.3.1.1 of this SER, the applicant identified airports within 97 km (60 mi) of the SRS and provided the relative distance of these airports to the SRS.

Section 3.5.1.6 of NUREG-0800 provides guidance for assessing aircraft hazards. This source contains proximity criteria that allow excluding consideration of aircraft hazards if the facility meets these proximity criteria. The LA referred to these criteria in its Table 5.3.1-9 for aircraft hazard screening. Information required to implement these proximity criteria includes (1) the distance between the facility and nearby airports, (2) the annual number of operations for each airport, (3) military training routes, and (4) the distance of the facility from Federal airways, holding patterns, and approach patterns.

The applicant summarized the aircraft hazard analysis for the MFFF in DCS-NRC-000085, "Clarification of Responses to NRC Request for Additional Information" (DCS, 2002), indicating that all nearby commercial airports were more than 16 km (10 mi) from the facility. The applicant determined that the annual operations for these airports, based on information for 1999 from the Federal Aviation Administration, were smaller than the proximity criterion provided in Section 3.5.1.6 of NUREG-0800. In addition, the nearest edge of a military training route was more than 8 km (5 mi) from the facility (DCS, 2002). Consequently, the applicant concluded that the risk for the activities associated with the nearby commercial airports and military training routes was smaller than the acceptance probability of 10^{-7}/year and, therefore, based on Section 3.5.1.6 of NUREG-0800, this hazard is not a design-basis concern.

One Federal airway and the edge of a second cross the SRS. The applicant estimated the probability of an aircraft crash into the BMF and BEG from these Federal airways, based on the flight information compiled by the Federal Aviation Administration, to be 2.74×10^{-8} and 1.38×10^{-9}, respectively, using the formula given in Section 3.5.1.6 of NUREG-0800.
The summary of the aircraft hazard analysis also indicated that Wackenhut Services, Inc. operates a heliport in B-Area, approximately 4.7 km (2.9 mi) from the MFFF. It uses two lightweight, multipurpose helicopters to support the security services at the SRS. The applicant calculated the helicopter crash probability using the equations from DOE-STD-3014-96, "Accident Analysis for Aircraft Crash into Hazardous Facilities" (DOE, 1996b). This standard references the methodology suggested in Section 3.5.1.6 of NUREG-0800. The estimated helicopter crash probability, based on the flight activities over a 5-year period, was 8.64×10^{-7} and 6.54×10^{-8} for the BMF and BEG, respectively. The helicopter crash probability for the BMF

was greater than the 10^{-7} annual probability of unacceptable radiological consequences indicated in Section 3.5.1.6 of NUREG-0800. To reduce the probability of unacceptable radiological consequences to an acceptable level, the analysis included the penetration resistance of the hardened exterior design for the BMF. Based on the LA, the exterior walls of the BMF are reinforced concrete, and a reinforced concrete security wall will enclose the BMF. Gabion stones will be placed in the 0.9-m (3-ft)-wide gap between the BMF's exterior walls and the security wall. Chelapati and Kennedy (NED, 1972) showed that the estimated probability of a small aircraft, such as a helicopter, penetrating a 0.3-m (1-ft)-thick concrete wall was 0.003. By including this probability, the applicant was able to show that the probability of a helicopter crash resulting in unacceptable radiological consequences was 2.6×10^{-9}.

The total aircraft crash probability (summation of those from the Federal airways and SRS helicopters) was 2.99×10^{-8} for the BMF and 6.67×10^{-8} for the BEG. Both probabilities were smaller than the annual probability of unacceptable radiological consequences. Consequently, the applicant concluded that aircraft crash hazards at the facility are not a design concern.

The staff reviewed the aircraft hazard analysis for the BMF and BEG and found that the approach used for the analysis was consistent with that suggested in Section 3.5.1.6 of NUREG-0800 and that the applicant's conclusion was acceptable.

Load Combinations

The load combinations used for the design of both SC-I and SC-II civil structures were consistent with those in Section 3.8.4 of NUREG-0800, except that tornado, tornado missile, and explosion loads were not considered in the load combinations for the SC-II structures. The staff reviewed the various load combinations presented in Section 1.1.2.1.6.4.2, "Structural Design Loading Combinations for SC–I Structures," and Section 1.1.2.1.7.8.2, "Loading Combinations for SC-II Structures," of the LA and determined that these load combinations are in accordance with those suggested in NUREG-0800 for the design of structures. Therefore, the load combinations are acceptable to the staff.

11.11.4 Structural Analysis and Design

To facilitate the review of the civil structural design for the MFFF, the staff adopted a vertical slice approach, selecting the design and analysis of the BMF for detailed review. The BMF is classified as an SC-I structure and an IROFS structure. Furthermore, the design approach the applicant used for the BMF is relatively more complicated than other civil structure systems for the MFFF. The staff has sufficient confidence that the applicant used acceptable methodologies or common practices for the design of other civil structural systems, by following a similar process, following the relevant codes and standards, and satisfying the design bases that the LA specifies and the related analyses to support the design.

The selected detailed review included six areas: (1) soil-structure interaction analysis, (2) foundation design analysis, (3) tornado missile barrier analysis and design, (4) external explosion analysis and design, (5) structural analysis, and (6) structural design of the reinforced concrete BMF.

The detailed review focuses on (1) the appropriateness of the data input used, (2) the acceptability of the assumptions made, (3) the acceptability of the methodologies used, and (4) the appropriateness of the results used for the design.

11.11.4.1 Soil-Structure Interaction Analysis

To determine the design seismic loads for the BMF to support the design, the applicant performed soil-structure interaction analyses discussed in DCS-01-XGA-DS-CAL-B-01069-1, “Soil-Structure Interaction Analysis of MOX Fuel Fabrication Building” (DCS, 2004). To support these analyses, the applicant generated synthetic time histories for the three components of the design-basis ground motion, in accordance with the guidelines in Section 3.7.1 of NUREG-0800. For example, the response spectra of the synthetic time histories enveloped the design response spectra at 76 predetermined frequencies and met the minimum power spectral density requirement at frequencies between 0.3 and 24 hertz (Hz). In addition, no response spectrum of the synthetic time history fell below the design response spectrum at more than 5 frequency points, nor by more than 10 percent. The cross-correlation coefficients between the three components of the applicant’s synthetic time histories were smaller than the limit value suggested in Section 3.7.1 of NUREG-0800.

The applicant used a simplified three-dimensional finite-element model that ignored the embedment and simulated intact slabs for the soil-structure interaction analyses of the BMF. The BMF consists of the MOX processing area, the BAP, and the BSR. The floor of the MOX processing area is 0.3 m (1.0 ft) above grade. The floor of the BAP is 5.3 m (17.5 ft) below the floor of the MOX processing area, and the floor of the BSR is 4.3 m (14 ft) below the floor of the MOX processing area. According to the LA, the embedment in the BAP and BSR is shallow compared to the plan dimensions of BMF. Therefore, the soil-structure interaction analyses ignored the embedment. Furthermore, in DCS-01-XGA-DS-CAL-B-01069-1 (DCS, 2004), the applicant assumed the base slabs for the three areas of the BMF were rigid. Because the applicant made these two assumptions based on the guidelines provided in ASCE 4-98 (ASCE, 1999), they are acceptable to the staff.

For the BEG, the applicant modeled the structure as a three-dimensional lumped-mass stick model in the soil-structure interaction analyses. The applicant developed the soil model for the soil-structure interaction analyses using the information from the soil exploration and site-response analysis. Three soil conditions (lower, best, and upper bound) accounting for material property uncertainties were considered in the soil-structure interaction analyses to develop the in-structure response spectra at each direction and for a given structural level.

The soil model included a sufficient number of idealized soil layers from the ground surface to the bedrock. An in-office review of the soil-structure interaction analysis-related documents showed that the thickness of each soil layer met the maximum thickness guideline commonly accepted in engineering practice (DCS, 2002), thus it is acceptable to the staff. The structural damping values used in the analysis were in accordance with Regulatory Guide 1.61, “Damping Values for Seismic Design of Nuclear Power Plants,” issued October 1973 (NRC, 1973b), for a safe-shutdown earthquake and are acceptable to the staff.

From the soil-structure interaction analyses, the applicant obtained response spectra at the foundation and each floor and roof level to develop acceleration profiles to design IROFS SSCs. The structural and floor response spectra resulting from the three soil conditions were broadened individually to account for variations in structural material properties and then enveloped for use in the structural design of the BMF. Section 1.1.2.1.6.4.1.3, “Extreme Environmental Loads,” of the LA discusses specific extents of spectrum-peak broadening. The staff found that the extents of spectrum-peak broadening are generally consistent with Regulatory Guide 1.122, “Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components,” issued February 1978 (NRC, 1978).

The applicant applied additional spectrum-peak broadening to the structural and floor response spectra to account for discrepancies in the structural natural frequencies of a fixed-base structure resulting from model differences in the soil-structure interaction analysis and the ANSYS structural analysis to design the BMF. The applicant's soil-structure interaction analysis used a simplified three-dimensional model comparable to the three-dimensional model the ANSYS analysis used (MOX, 2010a, 2010b; DCS, 2004).

The applicant determined the out-of-plane seismic responses of the flexible slabs and walls (having a fundamental vertical frequency smaller than 33 Hz) in separate analyses. In these separate analyses, the applicant modeled the slabs or walls as a one-degree-of-freedom system, using the SAP2000 computer code (DCS, 2004). The modeled one-degree-of-freedom system was a 12.2×12.2×0.6-m (40×40×2-ft) slab with fixed edges. The applicant subjected this one-degree-of-freedom system to the applicable floor motions generated from the results of the soil-structure interaction analysis.

At the conclusion of the soil-structure interaction analyses, the applicant extracted the results of the soil-structure interaction analyses of the BMF and presented them in a separate report, DCS01-XGA-DS-CAL-B-01072-0, "Seismic Floor Response Spectra for BMF and BEG" (MOX, 2009a), to facilitate design use. The report included a matrix of floor response spectra for design use and seismic movements at each floor elevation for assessing seismic anchor movements in the stress analysis of suspended systems.

The staff made the following determinations:

- The synthetic time histories used for the soil-structure interaction analyses are acceptable because they were developed in accordance with the guidelines provided in NUREG-0800.

- The approach used for the soil-structure interaction analyses for consideration of uncertainties associated with structural and soil material properties is acceptable because this approach is consistent with the design guidance in Regulatory Guide 1.122 (NRC, 1978).

- The assumptions made in the soil-structure interaction analyses regarding modeling the foundation are acceptable because these assumptions were developed based on the guidelines provided in ASCE 4-98 (ASCE, 1999).

- The approach used to broaden the spectral peaks of the structural and floor response spectra from the soil-structure interaction analyses is acceptable because this approach is consistent with Regulatory Guide 1.122.

11.11.4.1.1 Foundation Design Analysis

The LA indicated that load combinations used to assess the effects of overturning, sliding, and flotation on structural stability were consistent with those recommended in Section 3.8.5 of NUREG-0800. A review of Table 1.1.2-3 of the LA confirms that the load combinations and the minimum safety factors for each condition listed are consistent with those provided in NUREG-0800 and are acceptable to the staff.

The in-office review focused on analyses of sliding stability, torsion, and overturning of the BMF during a seismic event. In the sliding stability analysis, only the friction between the foundation mat and soil was considered. Lateral resisting forces from the below-grade portion of the BMF were not included in the analysis for conservatism. The seismic load input was determined using the 100-40-40-percent rule. The horizontal resisting force was calculated based on the total weight of the BMF and an uplift resulting from the 40-percent vertical seismic uplift with a 10-percent margin.

The torsional and overturning moments were calculated using a static three-dimensional lumped-mass stick model. The horizontal forces, based on mass and seismic acceleration values, are applied to the stick model to obtain torsional and overturning moments. Torsional moment arises from the eccentricity between the mass center and rigidity center of the model. A 5-percent offset was added to the stick model by offsetting the mass center to obtain the accidental torsion. The overturning moment was calculated from the inertial forces on the mass centers and its distance to the edge of the foundation. For calculating resisting overturning moment, the mass of the structure was adjusted by the seismic uplift forces using the 100-40-40-percent rule. The foundation analysis, including assumptions and methodology used is consistent with ASCE requirements and is acceptable.

The lateral displacement of foundation soil, as calculated by the SASSI computer code, is acceptable.

11.11.4.1.2 Tornado Missile Barrier Analysis and Design

The applicant's impact analyses considered the following three tornado-generated missiles in DCS01-XGA-DS-CAL-B01063-0, "Tornado Missile Barrier Analysis and Design" (DCS, 2005b):

(1) a 1,361-kilogram (kg) (3,000-pound (lb)) automobile with a 40-km/h (25-mph) horizontal impact speed

(2) a 34-kg (75-lb), 7.6-cm (3-in.) standard steel pipe with a 121-km/h (75-mph) horizontal and a 80-km/h (50-mph) vertical impact speed

(3) a 6.8-kg (15-lb), 5.1 × 10.2-cm (2 × 4-in.) timber plank with a 241-km/h (150-mph) horizontal and a 161-km/h (100-mph) vertical impact speed

These tornado missiles are consistent with the three tornado-generated missile design bases discussed in Section 1.1.2.1 of the LA (Table 1.1.2-2).

The applicant analyzed the local and global effects of these three missiles on the roof and wall of the BMF and BEG (DCS, 2005b), using the methodologies specified in Section 6.4 of ASCE Manual and Report No. 58 (ASCE, 1980). The analyses did not account for the cushion of 0.9-m (3-ft) gabion stones between the security barrier and the exterior building walls. In addition, the roof and wall panels were fixed on all four sides, with the impact loads acting at the center of the panels. With the assumption of fixed ends, the entire energy from the tornado-generated missiles is transformed into impact energy; the staff considers this assumption to be conservative. The applicant also increased the calculated perforation and scabbing thicknesses by a factor of 1.2, following the recommendation in the ASCE Manual and Report No. 58 (ASCE, 1980) to provide sufficient safety margin. The results of the analysis indicate that the civil structure systems, as designed, will be able to retain their integrity and functionality after a strike from a tornado-generated missile.

The staff reviewed the applicant's tornado-missile barrier analysis and found it acceptable because the applicant used the methods recommended in ASCE Manual and Report No. 58 (ASCE, 1980) for impact analysis, and the applicant obtained and interpreted the analysis results appropriately.

11.11.4.1.3 External Explosion Analysis and Design

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[redacted]

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[redacted]

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The staff reviewed the analyses related to the impact of the external explosion and has determined that they followed the acceptable procedures and that the results are acceptable.

11.11.4.1.4 Structural Analysis of the Mixed Oxide Fuel Fabrication Building

To account for the effect of static settlement on the BMF, the applicant estimated a set of static spring constants at various locations of the base slab, using the computer code FLAC, in DCS01-WRS-DS-CAL-G–00017D, "Estimates of Static Settlement of MFFF Structure" (DCS, 2005a). The applicant calculated the static soil spring constants by dividing the ground pressure estimated at a location immediately beneath the base slab with the estimated settlement at the same location (DCS, 2005a). Then it verified the developed static soil spring constants by applying these constants to the structural (ANSYS) modeling of the BMF to approximate the stress-strain response of foundation soils to structural loads in DCS01-XGA-DS-CAL-B–01070-0, "Vertical Soil Springs at Base Slabs of MOX Fuel Fabrication

Building" (DCS, 2005c). The staff found that the approaches for estimating and verifying the static soil spring constants for structural design and analysis are reasonable.

The applicant performed a structural analysis of the BMF using the computer codes ANSYS and SASSI. A detailed three-dimensional finite-element model was analyzed using the computer code ANSYS. This analysis used the design-bases inputs discussed in the LA site and design-specific parameters and information and soil-structure interaction analysis carried out by the SASSI computer code. The applicant used shell elements to model the slabs and walls and beam elements to model the interaction between the security shear wall and the exterior shear wall. The SASSI soil-structure interaction analysis included all frequencies that significantly contribute to the seismic response of the BMF. The staff reviewed the structural analysis of the BMF and found the input data, assumptions, and idealizations used for structural and soil-structure interaction analyses consistent with ASCE requirements are acceptable.

11.11.4.1.5 Structural Design of Reinforced Concrete Mixed Oxide Fuel Fabrication Building

The structural design reports, design drawings, computer simulation demonstration, and design-related details examined during the in-office review provided structural design details of the reinforced concrete foundations, security shear walls, exterior shear walls, frames, floor slabs, roof slabs, and tie-back steel beams, and the connections among these components. The applicant based the designs on the member forces calculated by computer codes ANSYS, SASSI, SAP2000, and other associated codes. The seismic forces used for the design are based on the simplified model of the structure used in the soil-structure interaction analysis. The design followed ACI 318-99, ACI-SP-175-98, ACI 349-97, and ACI 349.1R-91 (ACI, 1999, 1998, 1997, and 1996) for reinforced concrete components and the AISI specifications (AISI, 1996) for steel structures. The staff reviewed the structural design of the BMF and found the assumptions, idealizations, and codes and standards used for the design of the reinforced concrete and steel structures are acceptable.

11.11.5 Seismic Qualification of Civil Structures

The applicant conducted the seismic qualification for civil structures to demonstrate structural integrity under seismic loads through design and analysis. Section 1.1.2.1 of the LA provides the related requirements for seismic qualification. Sections 11.11.3.4 of this SER discuss the staff review of this section. The staff reviewed the codes and standards for the designs of SC-I and SC-II civil structural systems and concluded that the cited codes and standards are consensus standards that provide reasonable guidance consistent with the categorization assigned to the buildings and are consistent with Section 11.4.6.1 of NUREG-1718.

11.11.6 Natural Phenomena Accident Sequences

This section discusses the credible natural phenomena that could affect the MFFF during the period of facility operation. Natural phenomena could result in either the dispersion of radioactive material and hazardous chemicals, or a loss of subcritical conditions. Natural phenomena are also considered as initiators of other events, such as explosions or leaks.

The ISA addresses NPHs up to and including design-basis accidents. The design bases for applicable NPHs are based on the site description information. A comprehensive list of NPHs was evaluated and screened for applicability to MFFF operations for description. The resultant NPHs applicable to the MFFF, with unmitigated consequences that were determined to be not low, include the following:

- earthquake (including liquefaction)
- tornado (including tornado missiles)
- severe wind
- external fire (evaluated in Section 7.3.6.12 of the SER)
- rain, snow, and ice

The MFFF ISA evaluated the following NPH event groups:

- NPH-01, earthquake affecting the BMF, BEG/UEF, liquid waste reception (KWD) unit fluid transport system, hazardous material release

- NPH-02, tornado at the BMF, BEG/UEF, KWD unit, tornado-driven missiles, and a wind and atmospheric pressure change of 150 psf at a rate of 55 psf/second)

- NPH-03, severe winds affecting the BMF, BEG/UEF, waste transfer lines, extreme winds, and wind-driven missiles

- NPH-04, external fire starting from an NPH

- NPH-05, rain, snow, and ice affecting the BMF, BEG/UEF, and waste transfer lines

11.11.6.1 NPH-01, Earthquake Affecting the BMF, BEG/UEF, KWD Fluid Transport System, Hazardous Material Release

Earthquakes are postulated to occur as a natural phenomena event. Earthquakes can affect many SSCs simultaneously and, if unmitigated, could result in the dispersal of nuclear material or in a criticality accident. Thus, multifaceted event evaluations consider the impact of earthquakes. Equipment and structures where failure may directly or indirectly lead to an unacceptable dispersion of nuclear materials or to a criticality accident as a result of an earthquake are designed for the design earthquake (DE). Similarly, SSCs are designed for the DE if their failure during an earthquake could damage seismically designed equipment or prevent or limit its operation.

[redacted]

With respect to the discussions in Section 11.11.3.5.2.2, “Severe Environmental Loads”; Section 11.11.5, “Seismic Qualification of Civil Structures”; Section 11.7, “Fluid Transport Systems”; and Section 11.8, “Fluid Systems,” of this SER, the staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61(b) and (c), “Performance Requirements,” should a seismic event occur.

11.11.6.2 NPH-02, Tornado at the BMF, BEG/UEF, KWD, Tornado-Driven Missiles, and a Wind and Atmospheric Pressure Change of 7.18 KPa (150 psf) at a Rate of 2.63 KPa.Second (55 psf/second)

Tornadoes may occur in extreme weather, such as thunderstorms or hurricanes, and are postulated to occur on the MFFF site. Tornado loads include loads caused by tornado wind pressure, loads created by the tornado-generated differential pressure, and loads resulting from tornado-generated missiles. This event involves tornado winds that cause damage to IROFS SSCs, resulting in the failure of dynamic confinement systems caused by pressure differential, structural damage, or direct damage to SSCs, potentially resulting in radiological and chemical consequences to facility workers, site workers, individuals outside controlled areas, and the environment. The BMF structure, BEG and associated UEF structures, missile barriers, KWD high alpha liquid waste transfer lines, and tornado dampers provide protection to those IROFS SSCs from the effects of a tornado.

[redacted]

With respect to the discussions in Section 11.11.3.5.2.2, the staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61, in the event of a tornado.

11.11.6.3 NPH-03, Severe Winds Affecting the BMF, BEG/UEF, Waste Transfer Lines, Extreme Winds and Wind-Driven Missiles

Severe winds associated with thunderstorms or hurricanes are postulated to occur on the MFFF site. Severe wind loads include loads from straight wind and wind-driven missiles. This event involves severe straight winds that affect the BMF structure, the BEG and associated UEF structures, missile barriers, and waste transfer lines; such winds could lead to the failure of dynamic confinement systems caused by pressure differential, structural damage, or damage to SSCs that could result in radiological and chemical consequences to receptors.

[redacted]

With respect to the discussions in Section 11.11.3.5.2.2, the staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 in the event of severe winds.

11.11.6.4 NPH-05, Rain, Snow, Ice Affecting the BMF, BEG/UEF, and Waste Transfer Lines

Rain, snow, and ice are postulated to occur at the MFFF site during operation of the facility and are discussed below. The MFFF site grading design maintains flood levels, caused by runoff generated by locally intense rain, sufficiently below the building floor elevations. In addition, the anticipated maximum flood levels that could occur for the Savannah River Basin are well below the plateau elevations established for the MFFF project site.

The potential impacts of rain, snow, and ice are as follows:

- damage to the structures of the BMF and BEG and associated UEF that result in damage to SSCs within the structures
- direct damage to IROFS SSCs within the BMF and BEG/UEF
- damage to other SSCs (non-IROFS) within the BMF and BEG/UEF that cause them to fail in a manner that prevents IROFS from performing their safety functions
- damage to the KWD high alpha liquid waste transfer line leading to a radiological release

The safety strategy for this event is to mitigate the consequences of rain, snow, and ice by providing robust structures, in the case of the BMF and BEG/UEF structures, and by sufficiently burying waste transfer lines. [redacted]

With respect to the discussions in Section 11.11.3.5.2.1, "Normal Loads," the staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 in the event of rain, snow, or ice.

11.11.7 External Manmade Hazard Events

External manmade events are those events that are generated by external manmade hazards (EMMHs). EMMHs are those hazards that arise outside of the MFFF property boundary from the operation of nearby public, private, government, industrial, chemical, nuclear, and military facilities and transportation routes that could affect MFFF operations.

Section 11.11.7.1 discusses the event group EMMH-02, "External Explosion," which is related to the civil structural evaluation.

11.11.7.1 EMMH-02, External Explosion

An explosion originating outside the MFFF area at an SRS facility or along an SRS transportation route could be caused by a number of events, such as process upsets and transportation accidents.

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With respect to the discussions in Section 11.11.3.5.2.3, "Extreme Environmental Loads," and Section 11.11.4.1.3, "External Explosion Analysis and Design," the staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 in the event of an external explosion.

11.11.8 Explosion Events

11.11.8.1 EXP-15, Outside Explosion

Explosion events occurring outside the BMF that could affect MFFF operations or safety support systems are postulated to occur on the MFFF site in the following specific areas:

- BRP
- MFFF gas storage area (UGS)
- MFFF site roadways

[redacted]

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[redacted]

[redacted]

[redacted]

With respect to the discussions in Sections 11.11.3.5.1.3 and 11.11.4.1.3, the staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 in the event of an outside explosion.

11.11.8.2 *EXP-16, Miscellaneous Explosions*

Within the BAP, the MOX processing area, the BEG, and the BSR, there are some potential explosion hazards that either do not directly involve radiological material or involve only trace quantities of radiological material. [redacted]

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The staff concludes with reasonable assurance that the applicant has demonstrated that the facility will be in compliance with the performance requirements of 10 CFR 70.61 for miscellaneous explosions, since none of the potential events could cause a release of, or exposure to, radioactive material that would exceed exposures of low consequence.

11.11.9 Evaluation Findings

Section 1.1.2 of the LA provided design-basis and structural design information for civil structural systems for the MFFF. Based on the staff review of the LA and supporting information that the applicant provided relevant to civil structural systems, the staff finds that the applicant has met the baseline design criteria set forth in 10 CFR 70.64(a)(2). In addition, the staff concludes, pursuant to 10 CFR 70.23(b), that the design bases of the civil structural systems identified by the applicant will provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents. Furthermore, based on the staff's in-office design review of a selected civil structure (i.e., BMF), the staff concludes that the design of the facility was performed following the codes and standards specified in Section 1.1.2.1.6.3 of the LA and the design bases specified in Section 1.1.2.1.6.4 of the LA.

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12.0 HUMAN FACTORS ENGINEERING

This chapter of the safety evaluation report (SER) describes the review by the U.S. Nuclear Regulatory Commission (NRC) staff of human factors engineering (HFE) as described in Chapter 12 of the mixed oxide (MOX) fuel fabrication facility (MFFF) license application (LA) (MOX 2010a). The purpose of the review is to establish that HFE is applied to personnel activities identified as safety significant is consistent with the findings of the Integrated Safety Analysis (ISA) (MOX 2010b) and to determine whether an item relied on for safety (IROFS) has special or unique safety significance. The staff evaluated the information provided in Chapter 12 of the LA and supporting documentation (NRC 2009), in accordance with the guidance provided in NUREG-1718, "Standard Review Plan for Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility," Chapter 12, "Human Factors Engineering" (NRC, 2000).

12.1 Regulatory Requirements

The staff reviewed how the LA and ISA summary address the following regulatory requirements for HFE for personnel activities:

- Title 10 of the *Code of Federal Regulations* (10 CFR) § 70.61(e), "Performance requirements," which requires a safety program to ensure that each IROFS will be available and reliable to perform its intended function when needed

- 10 CFR § 70.62, "Safety Program and Integrated Safety Analysis," which requires a safety program and an ISA

- 10 CFR § 70.64(b)(2), "Requirements for new facilities or new processes at existing facilities," which requires features that enhance safety by reducing challenges to IROFS (defense-in-depth practices)

12.2 Regulatory Acceptance Criteria

NUREG-1718, Section 12.4.3, "Regulatory Acceptance Criteria" (NRC, 2000), contains the acceptance criteria to be used to support the HFE review. The criteria are divided into nine areas of review, designated as Criteria A through I:

A. a description of the safety-significant personnel actions, the associated human systems interfaces (HSIs), and the consequences of incorrectly performing or omitting actions for each personnel activity

B. the applicant's plans for HFE design review

C. operating experience review

D. function and task analysis

E. HSI design, inventory, and characterization

F. staffing

G. procedure development

H. training program development

I. human factors verification and validation (V&V)

In Section 12.3 below, each review area and its associated acceptance criteria are evaluated.

12.3 Regulatory Review and Analysis

The purpose of this review is to establish that HFE is applied to personnel activities identified as safety significant, consistent with the findings of the ISA and the determination of whether an IROFS has special or unique safety significance. A graded approach commensurate with the complexity and integration and operation of the control systems is appropriate. The application of HFE to personnel activities ensures that the potential for human error in facility operations is addressed during the design of the facility by facilitating correct, and inhibiting wrong, decisions by personnel and by providing the means for detecting and correcting or compensating for error.

The review was conducted in accordance with the review guidance in Chapter 12 of NUREG-1718 (NRC, 2000). The results are organized according to the review topics identified in Section 12.3 of NUREG-1718 and summarized in Section 12.4 of this SER.

12.3.1 Safety-Significant Personnel Actions

12.3.1.1 Review Criterion

Section 12.3 of NUREG-1718 states, “The scope of the review should be consistent with the results of the ISA and should include as appropriate, a description of the safety-significant personnel actions, the associated human systems interfaces (HSIs), and the consequences of incorrectly performing or omitting actions for each personnel activity” (NRC, 2000). Section 12.4.1 of NUREG-1718 states that the regulatory requirements for HFE for personnel activities are 10 CFR § 70.61(e) and 10 CFR § 70.64(b)(2). Section 12.4.3 provides Acceptance Criterion A for the review of personnel actions: “The applicant appropriately identified the personnel activities such that the reviewer can understand the actions, the HSIs involved, and the consequences.”

12.3.1.2 Evaluation

Chapter 12.0 of the LA states the following:

> HFE principles and practices are applied specifically to the MFFF active and passive Engineered IROFS (for maintainability, testing, and surveillance purposes) and to those personnel activities that are identified by the ISA as Enhanced Administrative Control (EAC) IROFS and Administrative Control (AC) IROFS (i.e., Administrative IROFS).

This provides a clear commitment to HFE for the most important activities and equipment in the MFFF. Chapter 12.0 of the LA also states:

> MOX Services will also review the operator actions identified as Defense-in-Depth in the Nuclear Criticality Safety Evaluations (NCSEs) and Nuclear Safety Evaluations (NSEs) using a graded approach as defined in the Human Factors Engineering Program Plan (HEPP) and the Human Factors Engineering Implementation Plan (HFIP).

This commitment provides for further HFE of activities that may also play a lesser role in ensuring the safety of the MFFF.

Chapter 1.0, "Overview," of the ISA Summary (Page 1-2) states the following:

> …the identified IROFS are the necessary and sufficient set of design features and administrative controls (activities of personnel) implemented in the design to satisfy the performance requirements of 10 CFR §70.61. To provide an additional safety margin and satisfy the requirements of 10 CFR §70.64(b), the MFFF employs defense-in-depth practices. These features ensure that multiple layers of risk reduction exist.

Section 5.3 of the ISA summary describes the results of the hazard and accident analyses performed to identify the facility IROFS. Section 5.3.3 has several subsections that describe the results of the consequence analysis for each of the analyzed events. These subsections describe the event and the safety strategy to protect against the event and list the IROFS identified as necessary to implement the safety strategy. They also discuss risk and identify defense-in-depth systems for that event. Tables in each of the Section 5.3.3 subsections list the IROFS, with separate tables for engineered IROFS and administrative IROFS. Many of the engineered IROFS contain HSIs, such as controls, displays, and alarms that are not separately identified in the administrative IROFS. As noted above, the applicant's HFE program addresses these engineered IROFS, as well as the administrative IROFS.

HEPP Section 1.1, "Scope," provides for the application of HFE in the design, construction, test and evaluation, startup, and operation of the MFFF. Section 1.3 states that the HFE program goal is to successfully integrate the human subsystem into the MFFF. The program is focused on HSI vis-à-vis engineering and administrative IROFS. HEIP Section 1.4 indicates that the HFE program applies to ISA-identified IROFS functions (personnel activities identified as IROFS and personnel activities that support safety, such as maintenance).

HFIP Section 1.3 describes the HFE program goals as the successful integration of the human subsystem into the MFFF design and modification within the constraints provided in the regulatory requirements of 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," and the MFFF project. Additional goals are to ensure that engineering provides an MFFF work environment that fosters effective procedures, work patterns, personnel safety and health, and that minimizes factors that degrade human performance and increase error potential. Additional human factors goals are identified that are similar to those in NUREG-0711, "Human Factors Engineering Program Review Model" (NRC, 2004), with some exceptions (addressed in the function allocation (SER Section 12.3.4.2) and HSI design sections (SER Section 12.3.5.1).

In summary, the staff has reviewed the applicant's LA and supporting documentation and finds that the review criterion on safety-significant personnel actions is acceptable. The commitments made in the LA are consistent with the guidance provided in NUREG-1718 and the regulatory requirements of 10 CFR § 70.61(e) and 10 CFR § 70.64(b)(2) and thus are acceptable to the staff.

12.3.2 Human Factors Engineering Planning

12.3.2.1 Review Criterion

Section 12.4.3 of NUREG-1718 (NRC, 2000) provides Acceptance Criterion B for the review of HFE planning:

B. HFE Design Review Planning

The applicant's approach for planning HFE design review includes:

i. Identification of appropriate goals and scope to ensure that HFE practices and guidelines are implemented during design, construction, and operation of the facility.

ii. Implementation by an HFE team that has the appropriate composition, experience, and organizational authority to ensure that HFE is considered in the design of HSI for personnel activities. The HFE team's responsibilities include ensuring the proper development, execution, oversight, and documentation of the HFE function. Depending on the identification of personnel activities, it may be appropriate for the HFE team to consist of a single individual.

iii. An HFE team that attains the HFE goals and scope through established processes and procedures and that tracks HFE issues.

iv. An HFE function that ensures that all aspects of the personnel activities including the HSI are developed, designed, and evaluated on the basis of a structured approach using HFE.

The following section addresses each of the four subcriteria separately.

12.3.2.2 Evaluation

<u>Review Criterion 12.4.3 B(i)</u>

LA Chapter 12.2 states, "The MOX Project HFE program documented in the HEPP includes identification of HFE programmatic goals, scope, a description of the various HFE processes used for HFE review; the MOX HFE team composition; and preparation of a human performance monitoring strategy."

HEPP Section 1.1 provides for the application of HFE in design, construction, test and evaluation, startup, and operation of MFFF. Section 1.3 states that the goal of the HFE program is to successfully integrate the human subsystem into the MFFF. The program is focused on HSI vis-à-vis engineering and administrative IROFS. HEIP Section 1.4 indicates that the HFE program applies only to ISA-identified IROFS functions: personnel activities identified as IROFS and personnel activities that support safety, such as maintenance.

HEIP Section 1.3, “HFE Program Goals,” defines the program goal as the successful integration of the human subsystem into the MFFF design and modification within the constraints provided in the regulatory requirements of 10 CFR Part 70 and the MFFF project:

- 10 CFR 70.61(e) requires a safety program to ensure that each IROFS will be available and reliable to perform its intended function when needed. Per the HEIP, this includes enhanced administrative IROFS and administrative IROFS.

- 10 CFR 70.64(b)(2) requires features that enhance safety by reducing challenges to IROFS. MFFF design and construction incorporate HFE principles and practices for personnel activities designed as IROFS to eliminate or reduce the possibility of challenges to the performance capabilities of operators and maintainers.

The ISA Summary provides the following definitions:

- Administrative Controls—A human action that is prohibited or required to maintain safe process conditions (i.e., a simple administrative control).

- Enhanced Administrative Controls—A procedurally required or prohibited human action, combined with a physical device that alerts the operator that the action is needed to maintain safe process conditions, or otherwise adds substantial assurance of the required human performance (i.e., augmented administrative control).

Additional goals of the HEIP are to ensure that the work environment fosters effective procedures, work patterns, personnel safety and health, and minimizes factors that degrade human performance and increase error potential. Human factors goals are identified that are similar to those described in NUREG-0711 (NRC, 2004) with some exceptions (these are addressed in the function allocation (SER Section 12.3.4.2) and HSI design (SER Section 12.3.5.1).

Review Criterion 12.4.3 B(ii)

Team Characteristics

Section 12.2.2 of the LA describes the MOX project team characteristics. The LA indicates that the team is composed of a core group from functional groups involved in the design of the HFE aspects of the plant. Additional expertise is added as needed (e.g., training and procedure writing). The team also has the authority to make design changes.

The HEIP presents additional information on the project team. HEIP Section 2 addresses HFE organizational placement. Organizationally, HFE is within the MOX Services I&C Section. It reports to the Electrical/I&C Group which reports directly to the vice president of engineering. HEIP Section 2.1 identifies the HFE team, which comprises a core group of persons from different functional groups. Once operational, the team will transition to Plant Operations to continue human performance monitoring.

The expertise of the group is identified in HEIP Table 3, "HFE Core Team." This table provides qualifications and expected contributions that are consistent with those in NUREG-0711.

Team Responsibility

Section 12.2.2 of the LA describes the MFFF project team responsibilities. The team is responsible for implementing the HEPP and HFIP and making design changes within their scope. The team ensures that HFE criteria are properly applied to the design.

HEIP Section 2 also addresses team responsibilities. The human factors (HF) engineer develops the HFE program and implementation plans (IPs). The HF engineer is the focal point for HFE matters, including the design of applicable HSIs, and has primary control, direction, and supervision over all technical HFE aspects of the project and the authority to ensure that all HFE program plans are accomplished. HEIP Section 2 identifies an extensive list of responsibilities.

<u>Review Criterion 12.4.3 B(iii)</u>

Section 12.2.3 of the LA describes the HFE team's processes and procedures. The team uses a structured approach to HFE, which comprises established HFE activities such as operating experience review (OER), function allocation, task analysis, HSI design, and V&V. The HEPP and the HEIP further define and amplify the HFE program and the applicant's approach to HFE design. These provide the HFE goals and scope, as discussed in the previous review criteria. They also provide and refer to various HFE processes and procedures as described below.

HEIP Section 1.6 provides an overview of the technical program and the issues tracking system. This section also includes summary reports for major HFE activities. Figure 2 of the HEIP gives an overview of the application of HFE through the design process.

LA Section 12.2.4, "Issue Tracking," discusses the process by which issues are tracked. The HEIP indicates that HSI issues are identified as human engineering discrepancies (HEDs), and they are not considered in isolation. Any HFE concern or identified issue is tracked via the MOX project tracking system per MOX Services Project Procedure PP9-28, "Human Engineering Discrepancy."

<u>Review Criterion 12.4.3 B(iv)</u>

Section 12.2.3 of the LA addresses the HFE function for ensuring a structured approach and describes the full range of all HFE activities. The HFPP, HEIP, and supporting engineering plans reinforce the LA description.

In summary, the staff has reviewed the applicant's LA and supporting documentation and finds that the review criterion for HFE goals and scope, HFE team, HFE approach and issues tracking, and use of a structured approach to HFE is acceptable. The commitments made in the LA are consistent with the guidance provided in NUREG-1718 (NRC, 2000) and the regulatory requirements of 10 CFR § 70.61(e) and 10 CFR § 70.64(b)(2) and are acceptable to the staff.

12.3.3 Operating Experience Review

12.3.3.1 Review Criterion

Section 12.4.3 of NUREG-1718 (NRC, 2000) provides Acceptance Criterion C for the OER:

C. Operating Experience Review (OER)

The applicant identified safety-related HFE events or potential events that have occurred in existing facilities that are similar to the proposed facility. The applicant:

i. Reviewed the HFE-related events or potential events for relevance;

ii. Analyzed the HSI technology employed for the relevant HFE events or potential events; and

iii. Conducted (or reviewed existing) operator interviews and surveys on the HSI technology for the relevant HFE events or potential events.

12.3.3.2 Evaluation

In LA Section 12.0 the applicant stated that the design of the MFFF is based on the designs of two successfully operating AREVA NC facilities at La Hague and MELOX (the reference plants) in France, with modifications to incorporate “lessons learned” from operating those facilities. Existing designs of facilities, equipment, or systems that are adapted or modified for use in the MFFF are reviewed to evaluate the efficacy of human factors design elements. The depth and rigor of the evaluation depend on a determination of the complexity and importance to safety of the component or system and the consequences of human error.

LA Section 12.5 also addresses the OER. This section indicates that an OER was performed that focused on lessons learned from the reference plants. The review included sources such as interviews with operations, maintenance, and systems engineering personnel who are familiar with the reference plants. Further, Section 12.5 states that insights gained from the OER have been incorporated into the MFFF design.

The HEPP, Attachment D, “Lessons Learned,” lists 30 documents that outline where and how OERs were conducted for the reference plants in France (the aqueous polishing plant and the MELOX plant at La Hague). The staff reviewed some examples of completed OERs from the reference plants as documented in the MOX Services document entitled “MFFF Processing Area—Lessons Learned from Experience at MELOX—Overall Summary.”

In the OER area, the HEIP refers to the DCS Lessons Learned Program, PP1-7, Revision 1, dated March 1, 2006, and to the project-level lessons learned coordinator. PP1-7 applies to all MFFF personnel and specifies that during their work all personnel should identify potential lessons learned, which should then be evaluated and necessary actions determined. OER from external sources and reference plants is also evaluated. The procedure has an OER form and specifies routing of identified issues for review. If appropriate, the issues are entered into the condition report system governed by MOX Services procedure PP3-6, which uses logs and tracking numbers. The system appears appropriately specified and detailed.

The HFIP, Attachment B, "Lessons Learned Decision Point Questions," contains questions that can aid in determining the use and applicability of operating experience and lessons learned from the reference plants. Section 3.9, "Documentation," of the HFE IP discusses the "HFE Lessons Learned Log," which is used for tracking identified items. Section 3.10 of the HFE IP notes that while much has been accomplished in the area of lessons learned, the HFE team will continue to monitor various sources of lessons learned from the reference plants that might apply to the MFFF design and operation.

The staff has reviewed the applicant's LA and supporting documentation and finds that the review criterion for the operating experience review is acceptable. The commitments made in the LA are consistent with the guidance provided in NUREG-1718 (NRC, 2000) and the regulatory requirements of 10 CFR § 70.61(e) and 10 CFR § 70.64(b)(2) and are acceptable to the staff.

12.3.4 Function and Task Analysis

12.3.4.1 Review Criterion

Section 12.4.3 of NUREG-1718 (NRC, 2000) provides Acceptance Criterion D for the review of function and task analysis:

D. Functional Allocation Analysis and Task Analysis

i. Functional allocation analysis: The functional allocation analysis is based on the OER. Personnel activities are functionally allocated to take advantage of human strengths and to avoid demands that are not compatible with human capabilities.

ii. Task analysis: The task analysis includes the task analysis scope, identification and analysis of critical tasks; detailed description of personnel demands (e.g., input, processing, and output); iterative nature of the analysis; and incorporation of job design issues. The task analysis addresses each operating mode for each personnel activity (e.g., startup, normal operations, emergency operations, and shutdown). The task analysis results support the functional allocation.

The following section addresses each of the two subcriteria separately.

12.3.4.2 Evaluation

Review Criterion 12.4.3 D(i)

The LA addresses function allocation in Section 12.3. The function allocation methodology begins with an analysis of functional requirements for meeting MFFF safety functions. Section 12.3 of the LA states, "Functional requirements analysis (FRA) is the identification of functions that must be performed to satisfy the MFFF safety objectives to prevent or mitigate the consequences of postulated accidents that could damage the facility or cause undue risk to the health and safety of the public." This section also lists the specific objectives of the FRA.

HEPP Section 4, "Functional Requirements Analysis (FRA) and FA," defines FRA as the identification of IROFS functions that must be performed to satisfy MFFF safety objectives. It further notes that the FRA identifies IROFS control actions required to achieve functional goals. The ISA will identify all safety functions, with particular attention focused on the nuclear safety evaluation and the nuclear criticality safety evaluation.

The "Purpose" section of the document "Functional Classification List," dated January 7, 2008, states that the list is a product of general safety principles, the basis-of-design documents, and the ISA. It is essentially a large table providing quality levels and seismic qualification levels by system. Section 2.2, "Approach," lists the facility's performance criteria, as follows:

- Confinement of nuclear and radiochemical material with the use of
    - static or physical barriers,
    - dynamic means (ventilation systems)
    - emergency power

- Prevention of a nuclear criticality incident with the use of
    - equipment geometry control
    - fissile material mass control
    - moderation control
    - criticality control
    - process control

- Other items
    - fire detection and suppression
    - monitoring and alarm

The applicant's approach to functional requirements provides an acceptable definition of functions that can be and have been used as input to the rest of the HFE process.

According to the LA, function allocation results in the assignment of control functions to personnel, systems, or a combination of the two. Most of the allocations are based on the reference plant design where the approach taken is to automate operations to the extent possible. Thus, the allocations are based largely on the operating experience of the reference plants.

The allocations are then evaluated as part of the ISA. The Section 12.3 of LA states the following:

> Personnel activities are functionally allocated to take advantage of human strengths and to avoid demands that are not compatible with human capabilities. PrHAs for each process unit or workshop are performed and include the OER review for the unit. This is where the allocation is made to either an appropriate engineering control or to an Administrative Control (personnel action). HFE

evaluations or analyses on proposed Administrative IROFS are conducted to consider that the human interaction requirements will be compatible with human capability, under stated conditions or hypotheses. The HFE evaluations support the ISA in evaluating operator actions and inactions, including errors of omission and commission.

The HEPP and HEIP describe the function allocation methodology in greater detail.

HEPP Section 4 states that the MFFF function allocation is based largely on the reference plant design. It is acceptable to base MFFF predecessor functions on successful operation experience as long as the functions have not been modified and no other plant or procedure modifications negatively impact them. The HEPP states that operations at the MELOX and La Hague MFFF reference plants clearly demonstrate that automatic manufacturing processes produce the highest yield, lowest product variability, and highest and most consistent levels of quality for manufactured products. At the same time, both the chances and the consequences of an error or incident are greatly reduced.

With respect to function allocation (FA) methodology, MFFF safety and reliability are enhanced by exploiting the strengths and weaknesses of personnel and system elements. HEIP Section 4 states that FA will be based on HFE principles, in addition to technological and economic considerations, using a structured, well-documented methodology that seeks to provide personnel with logical, coherent, meaningful tasks. The HEIP indicates that the analysis considers the effects of interface management effects on situation awareness and workload. The items identified in the MOX plan include all the function allocation considerations in Figure 4.1 in NUREG-0711 (NRC, 2004).

With respect to the role of the MFFF operators, HEPP Section 4 indicates that the allocation of tasks between human and machine subsystems for activities that support IROFS will support the basis design goals contained in DCS01 AAJ DS DOB C 40112, “Basis of Design for Instrumentation and Control,” to automate the MFFF operations to the fullest extent. HEIP Section 4 indicates that the main driver for function allocation is the allocation of the predecessor plant designs; thus, OER is the technical basis. The basic design goal of FA is high automation, as stated in the “Basis of Design for Instrumentation and Controls.” This is further elaborated in the “Basis for Instrumentation and Control Design” document, which contains statements on the benefits of automation referenced above. The model for the MFFF control systems will be the MELOX system, which is a fully automated manufacturing and processing control system.

The staff has reviewed the applicant’s LA and supporting documentation and finds that the review criterion for function allocation is acceptable. The commitments made in the LA are consistent with the guidance provided in NUREG-1718 (NRC, 2000) and the regulatory requirements of 10 CFR § 70.61(e) and 10 CFR § 70.64(b)(2) and are acceptable to the staff.

Review Criterion 12.4.3 D(ii)

Task Analysis Scope

LA Section 12.5 addresses the task analysis scope and includes personnel activities identified as administrative IROFS. The scope of the task analysis is appropriately consistent with the scope of the overall program as discussed in SER Section 12.3.2.

Identification and Analysis of Critical Tasks

LA Section 12.1 addresses the identification of critical tasks, which are designated in the ISA as administrative IROFS. HEIP Section 5 states that there are over 200 such tasks. These will be examined to establish procedures, determine if they can be accomplished, and discover if human error can be introduced into the procedure. A task inventory will be prepared to list the IROFS tasks that operators and maintainers are to perform, along with a description of each task in behavioral terms. This provides an acceptable approach to identifying and analyzing critical tasks.

Detailed Description of Personnel Demands (e.g., input, processing, and output)

This subsection addresses the general task analysis (TA) methodology and summarizes key aspects of the methodology. In describing the general methodology, LA Section 12.5 indicates that task descriptions include detailed descriptions of personnel demands, including input, processing, and output. The process begins with a gross level of analysis involving the development of detailed narratives describing the tasks that personnel must accomplish. More detailed evaluations will be made based on specific methods selected by the responsible engineer.

HEIP Section 5 states that TA is the identification of task requirements for accomplishing functions allocated to personnel. TA can be conducted using many different methodologies. The desired type of information needed and its application will aid in selecting the appropriate technique and how it will be used. These considerations also determine the completeness of the analysis needed. The highest priority tasks (administrative IROFS) are deconstructed into their individual steps. The analyst suggests ways to make the task more efficient or suggests new tasks that more effectively support the goals. It is also important to discover where human errors are likely to be made and to attempt to prevent error. This portion of the TA should identify ways to advise the operator of an error just made, possible consequences of the uncorrected error, and how to correct the error.

The HFE team will agree on what TA method to apply. The responsible engineer will conduct TA during final design (HEPP Section 5).

HEIP Section 5.2 addresses specific methodologies. Generally, the method starts with a gross-level analysis (narrative of what personnel must do) that becomes more detailed (input, process, and output needed). HEIP Section 5.2 refers to Table 5, which is the task requirements table developed from NUREG-0711 (NRC, 2004).

HEIP Section 5 identifies the following TA steps:

> Step 1—Information collection—After a task description is in place, the type of information needed to do the analysis has to be decided (refers to Table 5.1, Identification of Task Information). It includes rows such as task, cognitive task, identification of subtasks, grouping of subtasks, commonalities and interrelationships, importance of subtasks, frequency, sequencing, decisions, objectives, performance criteria, info required, knowledge employed, etc. Each row has a description of what the task information is. Table 5.2 identifies methods that can be used to collect information to be used in TA, including: observation, interview, focus group, existing documentation, checklist, questionnaire, and videotape.

Step 2—Data recording—There are a variety of ways that data can be recorded and presented. A list of many types of data to be recorded are presented and are consistent with the task requirements table in NUREG-0711.

Step 3—Data analysis—The final step is using the information to yield the basic data for design decisions. Five selected techniques are described in Table 5.4. These are: hierarchical TA, interface surveys, link analysis, operational sequence diagrams (OSDs), and timeline analysis.

HEIP Appendix F gives an example TA recording form. It is in tabular format, with rows corresponding to task steps and columns for action, location, controls and displays, system response, confirmation required and how to confirm, and error of omission or commission observed. There is also heading information for IROFS, purpose of the task, equipment involved, special tools, task start cue, task stop cue, and total time. The form includes a set of 14 questions, including considerations such as how often a task is accomplished, the skill needed, and whether a task depends on another task.

HEPP Section 5 indicates that during TA, checks are done to ensure that operations has appropriate instrumentation and controls (I&C) available to confirm proper operation of the automated systems under all conditions. In addition, the TA will address the following:

- minimum number of operators
- minimum skills needed
- allocation of monitoring and control tasks to achieve meaningful jobs and to address workload management

HEPP Section 5 lists the task considerations and includes information similar to that found in Table 5.1 of NUREG-0711 (NRC, 2004). HEIP Section 5.1 defines additional task considerations. To ensure dependable and consistent human performance, the following should be addressed:

- Conditions leading to overload (multitasking) or underload are important in examining IROFS performance.
- Incorporation of reasonable margins may accommodate the consequences of inadequate performance.
- Meaningful feedback on task performance is needed.
- Designing for error tolerance to minimize human errors includes considerations such as automating the task, changing the task-loading condition, applying buffers such as time delays, and addressing significant findings in design, procedures, and training.

HEPP Section 5 states that the TA will identify applicable operator time response requirements, where the ISA has credited human actions. The ability of the operator to respond in a timely manner should be assessed.

HEPP Section 5 indicates that problem tasks (those that cannot be performed well, are confusing, or create a safety concern) will be identified and written as an HED for resolution.

HEIP Section 5 also indicates that the product of the TA will be documented in a summary report.

The methodology described by the applicant should result in a detailed description of personnel demands that includes input, processing, and output requirements for the tasks being analyzed, and meets the staff's review criterion.

Iterative Nature of the Analysis

LA Section 12.5 indicates that TA is performed iteratively and becomes progressively more detailed over the design cycle. HEIP Section 5.2 also describes the iterative aspect of tasks analysis. This meets the staff's review criterion.

Incorporation of Job Design Issues

LA Section 12.5 states that the TA methodology will address job design issues. (See also the job design discussion under "Detailed Description of Personnel Demands" above.) This meets the staff's review criterion.

TA Addresses Each Operating Mode for Each Personnel Activity

LA Section 12.5 indicates that personnel activities identified as IROFS are identified for each operating mode, including startup, normal operations, emergency operations, and shutdown. This meets the staff's review criterion.

TA Results Support the FA

LA Section 12.5 indicates that TA will confirm the results of the function analysis or the results may dictate a change in allocation. This meets the staff's review criterion.

The staff has reviewed the applicant's LA and supporting documentation and finds that it meets the review criterion for TA's. The commitments made in the LA are consistent with the guidance provided in NUREG-1718 (NRC, 2000) and the regulatory requirements of 10 CFR 70.61(e) and 10 CFR 70.64(b)(2) and are acceptable to the staff.

12.3.5 Human Systems Interface Design, Inventory, and Characterization

12.3.5.1 Review Criterion

Section 12.4 of NUREG-1718 provides Acceptance Criterion E for the review of HSI design. (Note that this criterion has several subparts. Brief titles have been added in brackets following each of the subparts for ease of reference in the evaluation that follows.)

E. HSI Design, Inventory, and Characterization

> The HSI design incorporates the functional allocation analysis and TA into the detailed design of safety-significant HSI components (e.g., alarms, displays, controls, and operator aids) through the systematic application

of HFE [HSI Design Inputs]. The HSI design includes the overall work environment, the work space layout (e.g., control room and remote shutdown facility layouts), the control panel and console design, the control and display device layout, and information and control interface design details [HSI Design Scope]. The HSI design process ensures the application of HFE to the HSI required to perform personnel activities [HSI Design Process]. The HSI design process excludes the development of extraneous controls and displays [Extraneous HSIs]. The HSI design documentation includes a complete HSI inventory and the basis for the HSI characterization [HSI Documentation].

The following section evaluates each of the five subcriteria separately.

12.3.5.2 Evaluation

<u>HSI Design Inputs</u>

LA Section 12.6 indicates that the "HSI design is derived from the existing and proven design of the reference plants HSIs, modified for both cultural calibration purposes and U.S. safety requirements." Additional inputs to the HSI design process come from a variety of sources, including the following:

- Analyses of personnel task requirements performed in the earlier stages of the design process are used to identify the requirements for the HSIs. These analyses include the OER, the FRA and FA, and the TA, along with the evaluations of staffing, qualifications, and job analyses.

- System requirements are interpreted as constraints imposed by the overall I&C system and are considered throughout the HSI design process.

- Regulatory requirements are identified as inputs to the HSI design process.

- The applicant's human factors design guideline, which is consistent with the staff's HSI design review guidance in NUREG-0700, "Human-System Interface Design Review Guidelines" (NRC, 2002).

- Other requirements that may be identified and are input to the HSI design function allocation and task analyses.

This information provides acceptable inputs to the HSI design process.

<u>HSI Design Scope</u>

LA Section 12.6 identifies the scope of the HSI design effort, including the HSI of the work environment, the work space layout, control panel and console design, control and display device layout, and information and control interface design. The HSI design avoids extraneous controls and displays and minimizes the incorporation of information, displays, controls, and features that unnecessarily complicate operator activities.

HEPP Section 7 indicates that, for those HSI design and characterization elements not carried over from the reference facilities or for elements that were substantially changed for "Americanization," the HSI design process translates function and task requirements into HSI characteristics.

The process is applied to those HSIs involving IROFS and will also be applied to non-IROF HSIs where practical. The HSI design scope is appropriately consistent with the scope of the overall HFE program.

<u>HSI Design Process</u>

LA Section 12.6 describes the general process by which HSIs are designed. Inputs are used to develop the human factors design guide (HFDG). The HFDG provides HFE guidance used for both the design and evaluation of HSIs. The HFDG is intended to provide MFFF staff with an easy-to-use resource for human factors guidance. It addresses a broad range of human factors topics that pertain to automation, maintenance, human-machine interface, workplace design, documentation, system security, safety, the environment, and anthropometry. The HFDG will be subject to updates and revision as the need arises. Because the HFDG is an approved project document, it is revised, reviewed, approved, and maintained in accordance with approved project records management procedures.

As noted in the review of the preceding criterion, the HSI design process will apply to those HSIs involving IROFS and will also be applied to non-IROF HSIs where practicable.

The staff reviewed the HFE guidance provided in the HEIP pertaining to alarm system, displays, controls, communications, and labeling. In general, the staff finds the guidance provided to be consistent with the HFE principles described in NUREG-0700 (NRC, 2002) and acceptable to the staff.

<u>Extraneous HSIs</u>

LA Section 12.6 indicates that the HSI design avoids extraneous controls and displays and minimizes the incorporation of information, displays, controls, and features that unnecessarily complicate operator activities. This meets the staff's review criterion.

<u>HSI Design Documentation</u>

LA Section 12.6 indicates that the design documentation includes a complete HSI inventory and HSI design basis. HEPP Section 7 provides additional detail concerning HSI documentation. The documentation will include the basis for HSI requirements and design characteristics, including the results of tests and analyses performed.

The staff has reviewed the applicant's LA and supporting documentation and finds that the review criterion on HIS design is acceptable. The commitments made in the LA are consistent with the guidance provided in NUREG-1718 (NRC, 2000) and the regulatory requirements of 10 CFR § 70.61(e) and 10 CFR § 70.64(b)(2) and are acceptable to the staff.

12.3.6 Staffing

12.3.6.1 Review Criterion

Section 12.4.3 of NUREG-1718 (NRC, 2000) provides Acceptance Criterion F for the review of staffing:

F. Staffing

Staffing is based on a review of the number and qualifications of personnel for each personnel activity during all plant operating conditions. The applicant conducts this review in a systematic manner that incorporates the functional allocation and task analysis results. Categories of personnel are based on the types of personnel activities. Staffing considerations include issues identified in the OER, functional allocation, HSI design, procedure development, and V&V.

12.3.6.2 Evaluation

Section 12.7 of the LA states that the MFFF organization will comprise five major groups: Business, Engineering, Licensing, Quality, and Plant Operations. Plant Operations is divided into the following subgroups: Operations, Maintenance, and Technical Support groups. The initial staffing levels are estimated and established based on experience with the reference plants that have been in operation for the last few decades in France and discussions with the NRC-licensed U.S. fuel assembly manufacturers. The LA also states that staffing will be updated based on results from the HFE program, including OER, TA, HSI design, procedure development, and V&V. The evaluation will also consider the number and complexity of tasks determined from the startup and test phase of the MFFF.

HEIP Section 6.2, “Objectives,” states that there will be a systematic analysis applying the results of the HFE elements of FA and TA to the number and qualifications of personnel. These analyses will also consider the other HFE elements (OER, HSI design, procedures, and V&V).

HEIP Section 6.1, “Purpose,” states that ultimately the operations manager and the operations group will make the final staffing and qualification decisions based on input from the HFE program elements. The HEIP also discusses the roles of different staff positions, including plant management, operators, shift supervisor, and maintenance personnel.

HEIP Section 6.6, “Role Developments,” states that the operations group defines the responsibilities and qualifications for individuals to assume their positions. It outlines “core competencies” that will be needed. Figure 6.1 of HEIP Section 6.6 shows the expected MFFF organization for full production operation, which has a staff of 787 personnel.

LA Section 12.7.1.2 states that shift staff teamwork and communications is based on the reference plants and NRC-licensed fuel assembly manufacturers. The applicant noted in a response to a request for additional information that staffing will probably be modified when more details become available. Generally, the operations manager meets with the staff to provide top-level guidance for shift work. The aqueous polishing manager and the MOX processing manager develop workbooks containing instructions for their shifts. The operators will have workbooks with specific instructions for the shift and also for recording the results of the process. During shift changeover, approximately 30 minutes of overlap are allowed for the operating shift to brief and update the oncoming shift.

The staff has reviewed the applicant's LA and supporting documentation and finds that the review criterion on staffing is acceptable. The commitments made in the LA are consistent with the guidance provided in NUREG-1718 (NRC, 2000) and the regulatory requirements of 10 CFR § 70.61(e) and 10 CFR § 70.64(b)(2) and are acceptable to the staff.

12.3.7 Procedure Development

12.3.7.1 Review Criterion

Section 12.4.3 of NUREG-1718 (NRC, 2000) provides Acceptance Criterion G for the review of procedure development:

> G. Procedure Development
>
> The applicant's procedure development for personnel activities incorporates HFE principles and criteria, along with all other design requirements, to develop procedures that are technically accurate, comprehensive, explicit, easy to utilize, and validated consistent with the acceptance criteria in Section 15.5.4 of this SRP. Because procedures are considered an essential component of the HSI design, they are derived from the same design process and analyses as the other components of the HSI (for example, displays, controls, operator aids) and subject to the same evaluation processes. Procedures include, as needed to support the personnel activity: generic technical guidance, plant and system operations, abnormal and emergency operations, tests (for example, preoperational, startup, and surveillance), and alarm response.

12.3.7.2 Evaluation

<u>Types of Procedures</u>

LA Section 12.8, "Procedure Development," states that procedures are essential to MFFF safety because they will guide personnel interactions with plant systems and responses to a variety of process-related events. Section 12.8 also states that the MFFF will include the following types of procedures: generic technical guidance, plant and system operations, abnormal and emergency operations, tests (e.g., preoperational, startup, and surveillance), and alarm response. LA Section 12.8 refers to MOX Project Quality Assurance Plan (MPQAP) for a more detailed description of MFFF procedures. The operating procedures include production procedures (such as normal, off-normal, temporary, shutdown, and alarm response), maintenance procedures (such as preventive and corrective maintenance, calibration, surveillance, functional testing, and work control), and emergency procedures.

Procedures will be developed for the following activities:

- operations—normal, off-normal, emergency, alarm response, startup, and shutdown
- maintenance

- surveillance and testing
- management control processes

In HEIP Section 9.12, the HFE list of MFFF procedures provides examples of the categories of procedures that will be developed.

The ISA results are used in identifying necessary procedures. Also, Section 9.3, "Basis Documentation," of the IP states that procedures are derived from management basis documents, design criteria, vendor information, engineering standards, drawings, and specifications.

In summary, the staff has reviewed the applicant's LA and supporting documentation and finds that the review criterion for procedures planned is acceptable.

<u>HFE for Procedures</u>

LA Section 12.8 states that "MFFF procedures for IROFS, including administrative control IROFS, incorporate HFE principles and other design criteria to develop procedures that are technically accurate, comprehensive, easy to utilize, and validated." Section 12.8 further states that "the scope of the procedures for HFE review will include Administrative IROFS covering emergency operating procedures; procedures for startup, operation, and shutdown; procedures for recovery from a "frozen" process; alarm response; and possible abnormal conditions." LA Section 12.8.1 states, "Procedures will incorporate appropriate HFE principles, practices and guidance criteria (NUREG-0700) into the text format and presentation to aid legibility, readability, and comprehension. Procedures will be technically accurate, comprehensive, explicit and validated."

HEIP Section 9 describes the procedure development process and the interaction of procedure development with the HFE program. Section 9.4 notes that TA and HSI design elements provide information needed by the procedure writers.

HEIP Section 9.13 describes how the operations staff is actively involved in the development of the procedures. Section 9.11 provides the required elements and format to be used for the procedures. Section 9.8, "Responsibilities," states that the MFFF HFE team will review the writers' style guides and the IROFS procedures.

The planned HFE for procedures at the MFFF is acceptable.

<u>Procedure V&V</u>

LA Section 12.8 states, "All applicable procedures will be verified and validated for correctness and that they can be carried out as required. Changes or modifications to procedures will be again verified." Section 12.8.1 states, that the HFE team will verify the hardcopy administrative control procedures (IROFS) of ISA-required administrative controls and validate the administrative procedures by observing operator walk-through or talk-through.

The MPQAP states that the procedure preparation methodology will ensure that plant procedures are validated through field tests. The MPQAP also states that "Operating and administrative procedures are reviewed and approved by management responsible and accountable for the associated operation. Prior to initial use or after major revisions, production and maintenance procedures are verified and validated."

Section 8 of the HEPP states, that all applicable procedures will be verified and validated to ensure that, they can be carried out as required. Section 9.8 of the HEIP states, that the HFE team will verify and validate the IROFS procedures.

The staff finds the planned V&V for procedures for the MFFF acceptable.

Overall, the HFE aspects of the procedure development program are acceptable. The commitments made in the LA are consistent with the guidance provided in NUREG-1718 (NRC, 2000) and the regulatory requirements of 10 CFR § 70.61(e) and 10 CFR § 70.64(b)(2) and are acceptable to the staff.

12.3.8 Training Program Development

12.3.8.1 Review Criterion

Section 12.4.3 of NUREG-1718 (NRC, 2000) provides Acceptance Criterion H for the review of training program development:

> H. Training Program Development
>
> The applicant's training program development addresses all personnel activities. The training program development indicates how the knowledge and skill requirements of personnel will be evaluated, how the training program development is coordinated with the other activities of the HFE design process, and how the training program will be implemented in an effective manner consistent with human factors principles and practices. The training program development should address the areas of review and acceptance criteria described in Section 15.4.4 of this SRP and should result in a training program that provides personnel with the qualifications commensurate with the personnel activities.

12.3.8.2 Evaluation

Section 12.9 of the LA states that the operator training program for the MFFF will address active and passive engineered IROFS and administrative control IROFS. Section 12.9 also states, "All personnel (except visitors) that have a need for MFFF plant access will be provided a General Employee Training (GET), along with more specified training according to position requirements. All the various functional groups of employees will be required to be trained in the aspects of their job responsibilities." The "Training and Qualification section of the MPQAP explains the systematic approach to training that is used at the MFFF and discusses the organization and management of training, the analysis of functional areas requiring training/qualification, the position training requirements, the use of learning objectives, organization of instruction, evaluation of trainee learning, systematic evaluation of training effectiveness, and retraining.

HEPP Section 9, "Training," discusses training programs and stresses the importance of training for safe and reliable operation of the MFFF. It also notes that personnel can perform only those functions for which training requirements are met. Personnel are trained for both normal operations and emergencies. Sections 9.1 through 9.5 describe a systematic approach to training that includes learning objectives, a comprehensive outline of knowledge, skills, and abilities, design and implementation of training, evaluation of trainee mastery, and evaluation and revision of training.

HEPP Section 9.4 states that trainees will be evaluated using methods that include written and oral tests, walk-throughs, and evaluation of on-the-job performance.

The HEIP provides more detailed information on training. Sections 10.3 through 10.9 describe the systematic approach to training to be used at the MFFF, as follows:

10.3 Training Program Development
10.4 General Approach Outline
10.5 Organization and Management of Training
10.6 Learning Objectives
10.7 Control of Training Program
10.8 Systematic Evaluation of Training Effectiveness and Modification of Training
10.9 Periodic Retraining and Continuing Assurance

Section 10.3 states that the training program development process will be coordinated with other HFE program activities. Section 10.6 states that learning objectives are derived from the HFE program elements of FRA, FA, TA, HSI design, and plant procedures. Section 10.11, "Personnel Qualifications," notes that minimum qualifications will be commensurate with assigned function responsibility. Section 10.13 states that the operations and maintenance organization is provided with training to ensure that its personnel can safely perform their assigned roles and work functions while meeting regulatory and product quality requirements.

The staff finds that the MFFF LA and supporting documentation meet this review criterion for the HFE aspects of training program development. The commitments made in the LA are consistent with the guidance provided in NUREG-1718 (NRC, 2000) and the regulatory requirements of 10 CFR § 70.61(e) and 10 CFR § 70.64(b)(2) and are acceptable to the staff.

12.3.9 Human Factors Verification and Validation

12.3.9.1 Review Criterion

Section 12.4.3 of NUREG-1718 (NRC, 2000) provides Acceptance Criterion I for the review of human factors V&V:

I. Verification & Validation

V&V confirms that the design incorporates HFE to HSI in a manner that enables the successful completion of personnel activities. The V&V should be applied to personnel activities and HSI design. The V&V process should consist of the following:

i. <u>HSI task support verification</u>: HSI components are appropriately provided for personnel activities through HSI task support verification. The verification shows that each HSI identified the task analysis and that the HSI design is appropriately provided, yet minimizes the incorporation of information, displays, controls, and decorative features that unnecessarily complicate personnel activities.

ii. HFE design verification: The HFE design verification shows that each HSI identified for a personnel activity incorporated HFE into the design. Deviations from accepted HFE principles and guidelines should be justified or documented for resolution/correction. If all HSI components are not addressed by HFE design verification, then an alternative multidimensional sampling methodology should be used to assure comprehensive consideration of the safety significance of HSI components. The sample size should be sufficient to identify a range of significant safety issues.

iii. Integrated system validation: The applicant commits to a performance-based evaluation of the integrated design to ensure that the HFE/HSI supports safe operation of the plant. Integrated system validation is performed after HFE problems identified in HFE design activities are resolved or corrected because these may negatively affect performance and, therefore, validation results. Validation is performed by evaluating personnel activities using appropriate measurement tools. All personnel activities should be tested and found to be adequately supported in the design, including personnel activities outside the control room.

iv. Human factors issue resolution verification: The applicant verifies that HFE issues identified during the design process were addressed and resolved. Issue resolution verification should be documented in the HFE issue tracking system established by the HFE team. Issues that cannot be resolved until the HSI design is constructed, installed, and tested should be identified and incorporated into the final HFE/HSI design verification.

v. Final HFE/HSI design verification: The applicant should commit to performing a final HFE/HSI design verification if the applicant cannot demonstrate that it has fully evaluated the actual installation of the final HSI design in the plant through the V&V activities described above. Final HFE/HSI design verification should demonstrate that in-plant HFE design implementation conforms to the HFE design as modified V&V activities.

V&V activities should be performed in the order listed above, as necessary. However, the applicant may find that it is necessary to iterate in order to address design corrections and modifications that occur during V&V.

The following section separately evaluates these six subcriteria (the sixth corresponds to the final unnumbered paragraph of the above criterion in NUREG-1718).

12.3.9.2 Evaluation

Review Criterion 12.4.3 I(i)

LA Section 12.10.1 addresses task support verification and indicates that it will verify that the HSI design is appropriately provided yet minimizes the incorporation of alarms, information, displays, and control capabilities that unnecessarily complicate personnel actions.

LA Section 12.10 indicates that HSI task support verification is an evaluation to verify that the HSI supports personnel task requirements as defined by task analyses. HSI task support verification determines whether the HSI provides all alarms, information, and control capabilities required for personnel tasks. The criteria for task support verification will come from earlier task analyses of HSI requirements for performance of personnel tasks. HEDs are identified for (1) personnel task requirements that are not fully supported by the HSI and (2) the presence of HSI components that may not be needed to support personnel tasks or that may impede personnel tasks. HFIP Sections 11.8.1, "Inventory and Characterization," and 11.8.2, "Task Support Verification," describe the task support verification methodology in greater detail.
In summary, the applicant has described an acceptable approach to HSI task support verification that meets the guidance provided in Review Criterion 12.4.3 I(i). The commitments made in the LA are consistent with the guidance provided in NUREG-1718 (NRC, 2000) and the regulatory requirements of 10 CFR 70.61(e) and 10 CFR 70.64(b)(2) and are acceptable to the staff.

Review Criterion 12.4.3 I(ii)

LA Section 12.10 indicates that HFE design verification is a static evaluation to verify that the HSI is designed to accommodate human capabilities and limitations as reflected in HFE guidelines, primarily using those guidelines provided in NUREG-0700 (NRC, 2002). HEDs are identified if the design is inconsistent with HFE guidelines. LA Section 12.10.2 describes the methodology for HFE design verification.

The HEPP indicates that the HSI design is compared with the MFFF style guide to perform HFE design verification.

HFIP Section 11.8.3 describes the HSI design verification methodology. The methodology described is essentially taken directly from NUREG-0711 (NRC, 2004).

In summary, the applicant has described an acceptable approach to HSI task support verification that meets the guidance provided in Review Criterion 12.4.3 I(ii). The commitments made in the LA are consistent with the guidance provided in NUREG-1718 (NRC, 2000) and the regulatory requirements of 10 CFR § 70.61(e) and 10 CFR § 70.64(b)(2) and are acceptable to the staff.

Review Criterion 12.4.3 I(iii)

LA Section 12.10.3 indicates that integrated system validation (ISV) is an evaluation using performance-based tests to determine whether an integrated system design (i.e., hardware, software, and personnel elements) meets performance requirements and acceptably supports safe operation of the MFFF. ISV will be conducted after significant HEDs identified in verification reviews have been resolved.

LA Section 12.10.3 identifies detailed ISV objectives, which include validation that the tasks associated with administrative IROFS can be accomplished within time and performance criteria. Other objectives are identified as well. Together, these objectives should lead to a comprehensive evaluation of the integrated system that meets the staff's review criterion.

The applicant's ISV methodology has not been developed yet. Instead, the LA commits to preparing an ISV methodology with general considerations of what a methodology should address. The ISV considerations identified in the LA include:

- The participants will be MFFF personnel.
- The scenarios to be used will include the administrative IROFS.
- Since no simulator will be available to conduct ISV scenarios, alternative methodologies, such as walk-throughs and observation of operation actions are the methods most likely to be used.
- Performance measures will be developed to validate that the operator understands the state of the system process, can navigate process screens on the Supervisory Control and Data Acquisition (SCADA) units, knows how to carry out emergency procedures, and has all the information and controls needed to accomplish the task.
- ISV acceptance criteria will include the requirement that personnel actions associated with administrative IROFS are 100 percent correct.
- HEDs are identified if performance criteria are not met.

In summary, the applicant has described a high-level approach to ISV. The commitments associated with this approach are made in the LA and are consistent with the guidance provided in NUREG-1718 (NRC, 2000) and the regulatory requirements of 10 CFR § 70.61(e) and 10 CFR § 70.64(b)(2) and are acceptable to the staff.

<u>Review Criterion 12.4.3 I(iv)</u>

LA Section 12.10.4 indicates that the resolution of HEDs identified during the design process will be verified prior to plant operation. Any significant HFE issues are reviewed, addressed, and documented. LA Section 12.10 indicates that HED resolution is an evaluation to provide reasonable assurance that the HEDs identified during the V&V activities have been acceptably assessed and resolved. HED resolution is an activity that should be performed iteratively with V&V. The MFFF process lead engineer or process responsible engineer may address and resolve issues identified during a V&V activity before conducting other V&V activities.

This commitment to use this methodology should ensure that identified HFE issues are resolved.

In summary, the applicant has described an acceptable approach to HFE issue resolution verification that meets the guidance provided in Review Criterion 12.4.3 I(iv). The commitments made in the LA are consistent with the guidance provided in NUREG-1718 (NRC, 2000) and the regulatory requirements of 10 CFR § 70.61(e) and 10 CFR § 70.64(b)(2) and are acceptable to the staff.

Review Criterion 12.4.3 I(v)

LA Section 12.10.5 indicates that the final (i.e., as-built) HSIs, procedures, and training program are compared with the detailed design description and HFDG to verify that they conform to the design that resulted from the HFE design process activities. The LA further indicates that aspects of the design not addressed during the V&V are evaluated later using an appropriate strategy or method (e.g., main control room noise and HVAC). Identified discrepancies are either corrected or justified.

Section 11 of the HEPP presents additional information, including the following:

- The design implementation verifies aspects of the design that are either partially verified or unverified prior to operation at the site.
- The final as-built HSIs, procedures, and training program are compared with the detailed design description to verify that they conform to the design that resulted from the HFE design process activities.
- HFE-related issues documented in the HFE issues tracking system and the MFFF project Action Tracker will be verified as having been adequately addressed.

HEIP Section 12.4, “Design Implementation Methods,” provides more detail on the implementation of the design.

This methodology should demonstrate that the final HSI conforms to the verified and validated design and that the design features not assessed in earlier V&V activities have been evaluated.

In summary, the applicant has described an acceptable approach to final HFE/HSI design verification that meets the guidance provided in Review Criterion 12.4.3 I(v). The commitments made in the LA are consistent with the guidance provided in NUREG-1718 (NRC, 2000) and the regulatory requirements of 10 CFR § 70.61(e) and 10 CFR § 70.64(b)(2) and are acceptable to the staff.

Review Criterion 12.4.3 I(vi)

LA Section 12.10 states that the preferred order is HSI task support verification, HFE design verification, and ISV, although iteration may be necessary. The HEPP and HFIP describe the order of V&V activities consistent with the criterion. However, Section 11.2 of the HFIP notes that it may be necessary to iterate portions of some activities in order to address design corrections and modifications that arise during V&V. This type of iteration is necessary and acceptable.

In summary, the applicant has described an acceptable approach to scheduling V&V activities that meets the guidance provided in Review Criterion 12.4.3 I(vi). The commitments made in the LA are consistent with the guidance provided in NUREG-1718 (NRC, 2000) and the regulatory requirements of 10 CFR § 70.61(e) and 10 CFR § 70.64(b)(2) and are acceptable to the staff.

12.4 Overall Evaluation Findings

The staff reviewed the application of HFE to personnel activities described in the LA to possess and use radioactive material at the MFFF according to Chapter 12 of NUREG-1718 (NRC, 2000). The staff evaluated the LA and related HFE plans against the guidance contained in NUREG-1718 and the regulatory requirements related to HFE.

The staff concludes that the applicant included commitments that applied HFE to personnel activities identified as IROFS, consistent with the results of the ISA, and that its personnel activities meet the requirements associated with human factors given in 10 CFR Part 70.

REFERENCES

10 CFR Part 70 *U.S. Code of Federal Regulations*, "Domestic Licensing of Special Nuclear Material, Part 70, Chapter I, Title 10, "Energy."

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(MOX, 2010b) Duke Cogema Stone & Webster, "Mixed Oxide Fuel Fabrication Facility Integrated Safety Analysis Summary," Aiken, SC, March 2010.

(NRC, 2009) Memorandum from David Tiktinsky (NRC) to Margie Kotzalas, In-Office Review Summary—Human Factors, June 26, 2009.

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(NRC, 2004) U.S. Nuclear Regulatory Commission, NUREG-0711, Revision 2, "Human Factors Engineering Program Review Model," Washington, DC, February 2005.

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13.0 SAFEGUARDS AND SECURITY

13.1 Physical Protection

The purpose of the review was to determine that the applicant has committed to having a physical protection system that provides high assurance that activities involving Special Nuclear Material (SNM) are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.

MOX Services submitted a Physical Protection Plan (PPP) (MOX, 2010a), a Training and Qualification Plan (T&QP) (MOX, 2010b), and a Safeguards Contingency Response Plan (SCRP) (MOX, 2010c) as part of its application to possess and use radioactive material at the MFFF. All three of these plans were submitted as classified documents.

13.1.1 Regulatory Requirements

The regulatory requirements for the MFFF are:

- 10 CFR 73.45, Performance capabilities for fixed site physical protection systems and 10 CFR Part 73.46, Fixed site physical protection systems, subsystems, components, and procedures.

- 10 CFR Part 73, Appendix B—General Criteria for Security Personnel, Appendix C-Nuclear Power Plant Safeguards Contingency Plans, and Appendix H-Weapons Qualification Criteria.

13.1.2 Regulatory Guidance

The regulatory guidance used by the staff in the review of the physical security related submittals was:

- Regulatory Guide (RG) 5.70 Guidance for the Application of the Theft and Diversion of Category I Special Nuclear Material Design-Basis Threat in the design, development, and implementation of a physical security program that meets 10 CFR 73.45 and 73.46 requirements (NRC, 2007) (Document is classified CONFIDENTIAL-NSI).

- NUREG 1718, Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility (NRC, 2000a).

- NUREG/CR-6667, Standard Review Plan for Safeguards Contingency Response Plans for Category I Fuel Facilities (NRC, 2000b).

- NUREG/CR-6668, Standard Review Plan for Training and Qualification Plans for Security Personnel at Category I Fuel Facilities (NRC, 2000c).

13.1.3 Staff Review and Analysis

MOX Services submitted a PPP, T&QP, and a SCRP as part of its application to possess and use radioactive material at the MFFF. All three of these plans were classified documents.

The plans stated above referenced a MFFF Final Design Vulnerability Assessment Study (FDVAS) dated April 15, 2008, (MOX, 2008), and Penetration Delay Analysis of the MFFF Barriers Report generated by Sandia National Laboratories, undated, (MOX). Both reports were used to justify the use of alternate security methods as authorized in 10 CFR 73.46(a). The staff reviewed both documents submitted by the Applicant.

In the plans, the applicant requested the use of alternate methods for numerous areas at the MFFF. The results of the review and analysis of the alternate methods are classified SECRET and are not publicly available.

For information purposes, MOX Services was provided with the Category I (NRC, 2002) and Design Basis Threat (DBT) Interim Compensatory Measures Orders (ICMO) to assist in the design of the Physical Protection System (NRC, 2003). The ICMOs were incorporated into the PPP, SCRP, and T&QP that were submitted as part of the license application process. The Orders will be formally issued in the future. The licensee will then be required to review the applicable plans to ensure that the Orders are incorporated as required and are adequately addressed. Since the Orders have not been officially issued, the staff cannot assess the degree to which the Order requirements have been incorporated into the security plans until after the Orders are officially issued.

13.1.4 Evaluation Findings

The NRC staff's review of the PPP, for the protection of Strategic Special Nuclear Materials (SSNM) contains information that has been marked by the Applicant as "CONFIDENTIAL-NSI" (SECRET when Appendix C is attached) pursuant to 10 CFR 2.390. NRC staff reviewed the applicant's PPP for fixed site physical protection of SSNM. The methods, alternate methods, and procedures outlined in the PPP satisfy or sometimes exceed the performance objectives, systems capabilities, and reporting requirements specified in 10 CFR 73.45 and 73.46. The PPP for the facility is acceptable and provides reasonable assurance that the requirements for the physical protection of SSNM will be met.

The NRC staff's review of the applicant's SCRP contains information that has been marked by the applicant as "CONFIDENTIAL-NSI" pursuant to 10 CFR 2.390. The methods, alternate methods, and procedures outlined in the SCRP satisfy or sometimes exceed the performance objectives, systems capabilities, and reporting requirements specified in 10 CFR 73.45, 73.46, and Part 73 Appendix C. The SCRP for the facility is acceptable and provides reasonable assurance that the requirements for the physical protection of SSNM will be met.

The NRC staff's review of the applicant's T&QP, contains information that has been marked by the applicant as "CONFIDENTIAL-NSI" pursuant to 10 CFR 2.390. The methods, alternate methods, and procedures outlined in the T&QP satisfy or sometimes exceed the performance objectives, systems capabilities, and reporting requirements specified in 10 CFR 73.45, 73.46, and part 73 Appendix B. The T&QP for the facility is acceptable and provides reasonable assurance that the requirements for the physical protection of SSNM will be met.

REFERENCES

(NRC, 2000a) U.S. Nuclear Regulatory Commission, NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility," Washington, DC, August 2000.

(NRC, 2000b), U.S. Nuclear Regulatory Commission, NUREG/CR-6667, Standard review plan for safeguards contingency response plans for Category I fuel facilities, Washington, D.C, May 2000.

(NRC, 2000c), U.S. Nuclear Regulatory Commission, NUREG/CR-6668, Standard review plan for training and qualification plans for security personnel at Category I fuel facilities, Washington, D.C., May 2000

(NRC, 2002) Interim Compensatory Measure Order number EA 02-CAT I, Washington, DC, August 21, 2002. (Document is CONFIDENTIAL-NSI)

(NRC, 2003) Interim Compensatory Measure Order number EA 03-087, Design Basis Threat, Washington, DC, April 29, 2003. (Document is CONFIDENTIAL-NSI)

(MOX, 2008) MFFF Final Design Vulnerability Assessment Study (FDVAS), Aiken, SC April 15, 2008 (Document is SECRET),

(MOX) Penetration Delay Analysis of the MFFF Barriers Report, Sandia National Laboratories, Albuquerque, NM, undated, (Document is CONFIDENTIAL-NSI)

(MOX, 2010a) Revision 3 of the Physical Protection Plan (PPP), Aiken, SC, May, 2010 with page changes dated September 30, 2010, (Document is CONFIDENTIAL-NSI).

(MOX, 2010b) Revision 2 of the Training and Qualification Plan (T&QP), Aiken, SC, May, 2010 with page changes dated September 30, 2010, (Document is CONFIDENTIAL-NSI).

(MOX, 2010c) Revision 2 of the Safeguards Contingency Response Plan (SCRP), Aiken, SC, May 2010, (Document is CONFIDENTIAL-NSI).

(NRC, 2007) U.S. Nuclear Regulatory Commission, Regulatory Guide (RG) 5.70 Guidance for the application of the theft and diversion of Category I special nuclear material design-basis threat in the design, development, and implementation of a physical security program that meets 10 CFR 73.45 and 73.46 requirements, Washington, D.C, September 2007, (Document is CONFIDENTIAL-NSI).

10 CFR Part 73, Physical Protection of Plants and Materials

13.2 <u>Material Control and Accounting</u>

The purpose of this review was to verify that the applicant, Shaw AREVA MOX Services, LLC (MOX Services), provided sufficient information in the Mixed Oxide Fuel Fabrication Facility (MFFF) Fundamental Nuclear Material Control Plan (FNMCP) (MOX, 2010) for the staff of the U.S. Nuclear Regulatory Commission (NRC) to determine that the material control and accounting (MC&A) program meets the applicable regulatory requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 74, “Material Control and Accounting of Special Nuclear Material.” Specifically, the review is to ensure that the applicant’s FNMCP describes how an MC&A system will be established, implemented, and maintained, and to ensure that the MC&A system is adequate to protect against, detect, and respond to the loss or theft of strategic special nuclear material (SSNM) by achieving the following five performance objectives stated in 10 CFR 74.51(a):

A. Prompt investigation of anomalies potentially indicative of SSNM losses;

B. Timely detection of the possible abrupt loss of five or more formula kilograms (FKG) of SSNM from an individual unit process;

C. Rapid determination of whether an actual loss of five or more FKG occurred;

D. Ongoing confirmation of the presence of SSNM in assigned locations; and

E. Timely generation of information to aid in the recovery of SSNM in the event of an actual loss.

13.2.1 Regulatory Requirements

The NRC specifies the requirements for MC&A, including provisions for reports and regulatory inspections, in 10 CFR Part 74.

13.2.2 Regulatory Acceptance Criteria

The FNMCP must meet the regulatory requirements specified in 10 CFR Part 74. The NRC staff used the following guidance to determine if the applicant's FNMCP meets the requirements in 10 CFR Part 74:

- NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility" (NRC, 2000)
- NUREG-1280, "Standard Format and Content Acceptance Criteria for the Material Control and Accounting (MC&A) Reform Amendment: 10 CFR Part 74, Subpart E," Revision 1 (NRC, 1995)
- NUREG/BR-0006, Revision 7, "Instructions for Completing Nuclear Material Transaction Reports (DOE/NRC Forms 741 and 740M)" (NRC 2008)
- NUREG/BR-0007, Revision 6, "Instructions for the Preparation and Distribution of Material Status Reports (DOE/NRC Forms 742 and 742C)" (NRC 2009)
- NUREG/BR-0096, "Instructions and Guidance for Completing Physical Inventory Summary Reports" (NRC, 1992)

13.2.3 Staff Review and Analysis

The staff reviewed and evaluated information provided by the applicant in the MFFF FNMCP (MOX, 2010) for the proposed MC&A program. The staff reviewed the following aspects of the MC&A program to ensure that the applicant's FNMCP meets the five performance objectives outlined above:

- Process monitoring
- Item monitoring
- Alarm resolution
- Qualtiy assurance and accounting

13.2.3.1 Process Monitoring

For each unit process, the applicant must establish a production quality control program capable of monitoring the status of material in process. To that end, the FNMCP identifies the two main operation processes and segments them into process control units and related measurement points that allow for monitoring the status of licensed materials in process. The staff concludes the description of the process subdivision is acceptable because it satisfies test criteria for the category of material being processed, and meets the unit detection requirements and associated measurement points and parameters in accordance with 10 CFR 74.53.

The applicant described adequate material control tests for each process unit that has at least a 95 percent chance of detecting potential abrupt material losses of five FKG. The plan also calls for the evaluation and update of the action thresholds on a semiannual basis and outlines the ability to detect material losses involving credible substitute materials. The staff concludes that the applicant's description of the material control test is adequate per 10 CFR 74.53(b)(1) because one is provided for each process control unit, and a system of material control tests is confirmed for detecting abrupt loss of SSNM from single process units or locations within the facility.

The applicant also identified credible material substitutions and methods of testing for substitution or of controlling the substitute material to prevent or detect attempts at substitutions. The applicant also provided a basis for material and location categorization, including the classification of inaccessible locations and a listing of material types exempted from the abrupt loss detection tests, with respective locations and bases for exemption. Staff confirmed that the applicant identified the nuclear material locations within the facility classified as inaccessible and provided supporting rationale per 10 CFR 74.53(b)(2).

The description of the process monitoring program includes trend analysis techniques and decision criteria to indicate trickling material diversions. The staff concludes that the applicant's trend analysis is adequate because it meets the requirements of 10 CFR 74.53(b)(3) for monitoring sequences of process differences from material control tests, including the decision criteria for ascertaining when a significant trend exists.

The staff evaluated the applicant’s process monitoring program and found that this program contains adequate system features to detect abrupt losses and monitor the status of material in process, including process subdivisions and measurement points, material control tests, material classification, location categorization, material substitution, trend analysis, material exemptions, and research and development operations. The staff found that the applicant is capable of implementing a process monitoring program that permits effective functioning of the MC&A system and ensures that MC&A program performance will conform to the requirements in 10 CFR 74.53, “Process Monitoring.”

13.2.3.2 Item Monitoring

The applicant must establish a process to verify the presence and integrity of SSNM items on a statistical sampling basis. To that end, the FNMCP identifies and describes an item monitoring program that establishes the capability to provide timely plantwide detection of the loss of items that total two FKG of plutonium, with 99-percent power of detection. The staff concludes that the applicant can detect, in a timely manner, the loss of items that total two FKG of plutonium or more, and verify the presence and integrity of selected nuclear material items on an established periodic detection basis because it has established an adequate item monitoring program, per the requirements of 10 CFR 74.55(a).

The applicant described an identification system that includes attributes to ensure unique item identification. The description also includes the system features that precludes falsification or makes prompt detection of such attempts. Staff also determined that the applicant's item identification system is adequate because it possesses control measures that ensure unique item identification, precludes or make prompt detection of such attempts highly probable by using numbered containers, printed labels, and tamper-safe seal numbers for unique identification, per the requirements of 10 CFR 74.55(a).

It is also determined that the applicant's tamper-safing procedures are acceptable to ensure the continuing validity of previously measured and attested to nuclear material values assigned to unique items, and the personnel access controls, surveillance and records procedures for entrance and exit of personnel to and from control access areas. This meets the requirements of 10 CFR 74.55(a) for storage of nuclear material items.

To establish the capability to detect a five FKG loss in item forms, the applicant described a program for item monitoring tests to verify the presence and integrity of selected items, including the appropriate power of detection, verification frequency, and relative attractiveness of the material type in the item. In addition, the applicant described the statistical method used to establish the sample population and how the presence of selected items will be verified and confirmed through item monitoring tests. Staff determined that the applicant's item verification tests and related statistical sampling basis are consistent with the requirements specified in 10 CFR 74.55(b).

The applicant provided a basis for item classification and exempted items from item control tests, and a sound tamper-safing system employed to ensure the continuing validity of previously measured and attested to values assigned to unique items. Also included was a description of facility accessibility associated with personnel access controls, surveillance procedures, and item accounting and control procedures for items placed in and removed from secure storage. Staff determined that the applicant's methods used to classify nuclear material items are adequate because the methods are consistent with the material categories for the related process monitoring program, per the requirements of 10 CFR 74.55.

To achieve the detection capability, the applicant identified an item measurement system to quantify the content of items, including confirmatory measurements, an item sampling technique, and a verification method for verifying the presence and integrity of licensed nuclear materials. The applicant also applied facility design features in the form of a highly automated remote-controlled process and manufacturing facility features, including its Manufacturing Management and Information System. Staff determined that the applicant's description of the item accounting and control procedures for items placed in and removed from secure storage is acceptable because it will quantify the contents, and included the confirmatory measurements used to quantitatively verify the nuclear material content of items, including the controls that prevent or detect material substitution attempts.

The staff has reviewed the elements of this item monitoring program and found that the applicant's program is capable of providing timely plantwide detection of the loss of items and verifying the presence and integrity of nuclear material items at a required frequency. The staff also found that the applicant provided an adequate item monitoring program with real-time status of nuclear materials, a system of item identification and classification, tamper-safing procedures, material accessibility, item accounting and control procedures, item measurements, sample items, and item verification tests, as required in 10 CFR 74.55, "Item Monitoring."

13.2.3.3 Alarm Resolution

The applicant must establish an alarm resolution program that is capable of:

- Resolving the nature and causes of any MC&A alarm within approved time periods;
- Notifying the NRC of any MC&A alarms that remain unresolved beyond the time periods;
- Determining the amount of actual SSNM lost and taking corrective actions;
- Providing an ability to rapidly assess the validity of alleged thefts; and
- Taking appropriate actions when the abrupt loss detection estimate exceeds 2 kg of plutonium.

The FNMCP identifies and describes features of an alarm resolution program that is capable of resolving the nature and cause of any MC&A alarm within an approved time period and notifying the NRC of any unresolved alarm beyond the time period specified for its resolution. The applicant described the alarm resolution procedures that are applied to the various types of alarms and process units. The FNMCP provides a listing of credible causes of possible alarms by process unit and details of the resolution procedures by which specific causes could be identified, as required by 10 CFR 74.57(b).

The applicant described an alarm reporting program to notify the NRC of any MC&A alarms that remain unresolved beyond the time period specified for its resolution. The applicant also described the responsibility and assignment within the facility's organization for reporting unresolved alarms, and the types of information that will be provided to the NRC, including types of resolved alarms and related innocent causes as specified in 10 CFR 74.57(c) and (f)(2).

Decision rules by which a particular cause or combination of causes is considered acceptable for an alarm declaration, response time allotted to resolve each alarm type, and actions in response to the listing of item discrepancies are appropriately identified and described in the applicant's FNMCP to meet the requirements of 10 CFR 74.57(d) and (f)(1).

The applicant described a program to rapidly assess the validity of any suspected thefts of licensed nuclear materials, using a maintained item control system to determine the identity, quantity, and location of materials in item form, and implemented protective measures to prevent loss, misplacement, or accidental destruction of inventory and item location records. This program description satisfies the requirement in 10 CFR 74.57(e) to respond rapidly to alleged thefts.

The staff has reviewed the elements of this alarm resolution program and found that the applicant's program demonstrates that it will have the ability to respond to and resolve MC&A alarms of the potential loss of nuclear materials. This program covers alarm resolution procedures, decision rules, response time, item discrepancies, alarm reporting responsibilities, and the handling of any alleged thefts. This alarm resolution program conforms to the requirements contained in 10 CFR 74.57 "Alarm Resolution," and is acceptable.

13.2.3.4 Quality Assurance and Accounting

The applicant must establish quality assurance and accounting capabilities that address the following 11 elements: (1) MC&A management structure, (2) personnel qualification and training, (3) measurement systems, (4) measurement control, (5) physical inventory, (6) records systems and record maintenance (accounting), (7) material shipping and receiving, (8) scrap material control, (9) prevention of human error, (10) independent assessment, and (11) SSNM custodianship responsibilities. In its FNMCP, the applicant identified detailed methods and measures with respect to the overall MC&A quality assurance and accounting programs, which includes the 11 elements just described.

(1) The management structure described by the applicant includes the establishment and implementation of an organization that permits the effective functioning of the MC&A program; independence of responsibilities that have potentially conflicting goals; and adequate review, approval, and use of MC&A core procedures that are critical to the effectiveness of the MC&A system. Therefore, the applicant’s management structure satisfies 10 CFR 74.59(b).

(2) The applicant provided a personnel qualification and training program to ensure the competency of key MC&A personnel, including job requirements, minimum qualification, requalification needs, training program structure, and training objectives. The training program is also extended to operators and analysts involved with MC&A activities in order to perform their functions correctly with a minimum of errors. Therefore, the applicant’s personnel qualification and training program satisfies 10 CFR 74.59(c).

(3) The measurement system described in the FNMCP identifies all measurement points and the types of nuclear materials, and appropriate measurement methods for the types of materials and the components of each measurement system involved at each measurement point . This will ensure that all material information in accounting records is based on measured values, that the standard deviation associated with each measurement quantity is estimated, and that necessary data are provided for performing material control tests. Therefore, the applicant’s measurement system satisfies 10 CFR 74.59(d).

(4) The applicant described a measurement control program that is applied to measurement systems used for inventory, shipper-receiver measurement, monitoring cumulative shipper-receiver differences, standard error of the inventory difference estimator, and detection and response measurements. Therefore, the applicant’s measurement controls satisfies 10 CFR 74.59(e).

(5) The applicant provided the necessary requirements for scheduling, performing, and evaluating physical inventories, including an implementation of policies, practices, and procedures designed to ensure the quality of physical inventories, and methods for controlling and maintaining records and documentation associated with the inventory program. Therefore, the applicant’s physical inventory program satisfies 10 CFR 74.59(f).

(6) The applicant described the auditable records system, including recordkeeping policies, the type of data and information routinely recorded, and control measures and record maintenance to ensure record integrity, redundancy, and protection. Therefore, the applicant’s accounting system satisfies 10 CFR 74.59(g).

(7) In the FNMCP, the applicant established a program to accurately quantify in a timely manner the material contents of shipments and receipts, including procedures for the verification and confirmation measurements of shipments and receipts and for the review, evaluation, investigation, and resolution of significant shipper-receiver differences. Therefore, the applicant's internal control program satisfies 10 CFR 74.59(h)(1).

(8) The applicant also established a control program to process internally generated scrap materials, including identification of scrap quantities with regard to source, storage, and disposition, potential onsite or offsite recovery capacity, and measurements and inventory control via scrap control procedures. Therefore, the applicant's internal control program satisfies 10 CFR 74.59(h)(2).

(9) The applicant in its FNMCP establishes a human errors program to incorporate checks and balances in the MC&A system to control the rate of human errors in the MC&A accounting information, including procedures used to control the frequency and types of human errors in MC&A data. Therefore, the applicant's internal control program satisfies 10 CFR 74.59(h)(3).

(10) The applicant described an audit and independent assessment program to review the overall effectiveness of the MC&A system and past performance of its MC&A program relative to the performance objectives and system capabilities defined respectively in 10 CFR 74.51(a) and (b). Therefore, the applicant's internal control program satisfies 10 CFR 74.59(h)(4).

(11) The applicant provided a material custodianship program that identifies a sufficient number of areas to ensure an effectively executed custodial responsibility, as well as defined duties and authorities, minimum number of alternates, and maintenance of a listing of custodians and alternates. Therefore, the applicant's internal control program satisfies 10 CFR 74.59(h)(5).

The staff has reviewed the 11 elements of these quality assurance and accounting programs and found that these programs are appropriate and acceptable with regard to the requirements contained in 10 CFR 74.59, "Quality Assurance and Accounting Requirements."

13.2.3.5 Items Addressed by Compliance Plan

In its FNMCP, the applicant described the compliance item commitments in each affected section of the plan, the actions to be taken to achieve compliance, and the schedule for completing those actions. The compliance items pertain mainly to the development of implementing procedures associated with measurements, measurement control, item control, inventory, shipments and receipts, and basic software user guides for its computerized accounting programs. The staff determined that the commitments in the compliance plan are adequate and will be verified prior to granting authorization for possession of special nuclear material.

13.2.4 Evaluation Findings

The staff reviewed the MFFF license application to possess and use strategic special nuclear materials and special nuclear materials according to Section 13.2 of NUREG-1718 (NRC, 2000) and 10 CFR Part 74. Based on the review of the license application, the staff concludes that the applicant provided an acceptably robust FNMCP for the facility operations that will meet the applicable 10 CFR Part 74 requirements. The FNMCP describes acceptable methods for achieving the performance objectives in 10 CFR 74.51(a) and the system capabilities of 10 CFR 74.51(b). As a result, the staff has determined that the applicant meets the requirements in the area of MC&A to operate the facility under 10 CFR Part 74.

REFERENCES

(MOX, 2010) Shaw AREVA MOX Services, "Fundamental Nuclear Material Control Plan," Aiken, SC, May 2010.

(NRC, 1992) U.S. Nuclear Regulatory Commission, NUREG/BR-0096, "Instructions and Guidance for Completing Physical Inventory Summary Reports," Washington, DC, 1992.

(NRC, 1995) U.S. Nuclear Regulatory Commission, NUREG-1280, Revision 1, "Standard Format and Content Acceptance Criteria for the Material Control and Accounting (MC&A) Reform Amendment: 10 CFR Part 74, Subpart E," Washington, DC, April 1995.

(NRC, 2000) U.S. Nuclear Regulatory Commission, NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility," Washington, DC, August 2000.

(NRC, 2008) U.S. Nuclear Regulatory Commission, NUREG/BR-0006, Revision 7, "Instructions for Completing Nuclear Material Transaction Reports (DOE/NRC Forms 741 and 740M)," Washington, DC, June 2008.

(NRC, 2009) U.S. Nuclear Regulatory Commission, NUREG/BR-0007, Revision 6, "Instructions for the Preparation and Distribution of Material Status Reports (DOE/NRC Forms 742 and 742C)," Washington, DC, January 2009.

(10 CFR) Title 10 of the *Code of Federal Regulations* Part 74, "Material Control and Accounting of Special Nuclear Material."

14.0 EMERGENCY MANAGEMENT

14.1 Regulatory Requirements

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 70.22(i)(1), Shaw AREVA MOX Services (MOX Services) is required to develop an emergency plan or conduct an evaluation to demonstrate that an emergency plan is not required. The applicant is not required to submit an emergency plan if “an evaluation demonstrates that the maximum dose, resulting from a release of radioactive material, to a member of the public off site would not exceed 1 rem effective dose equivalent or an intake of 2 milligrams (mg) of soluble uranium.” The applicant chose to submit an evaluation rather than submit an emergency plan and submitted this revised evaluation on September 14, 2009 (MOX, 2009).

14.2 Facility Description

The mixed oxide (MOX) fuel fabrication facility (MFFF) is located near the center of the Savannah River Site (SRS), which extends over 803 square kilometers (km^2) (310 square miles) in southwestern South Carolina. The nearest public residence is located in the town of Jackson, South Carolina., outside the SRS boundary, approximately 9.7 km (6 miles) northwest of the facility. The SRS boundary is controlled with barriers, sign postings, and a security force. Several public transportation corridors, including public roads, rail lines, and recreational traffic on the Savannah River, allow unobstructed travel through the SRS, but only one of these corridors is within a 5-mile radius of the MFFF. During an emergency, these roads are evacuated and controlled. Figures 2-1 to 2-3 of the evaluation document (MOX, 2009) provide both a regional map of the SRS and a local map of the MFFF facility. The MFFF site comprises an area of approximately 0.17 km^2 (41 acres), with 0.07 km^2 (17 acres) developed and the remaining 0.10 km^2 (24 acres) landscaped.

The MFFF processes plutonium recovered from decommissioned weapons and other U.S. Department of Energy (DOE) sources. Impurities are removed in a liquid extraction process, which results in the development of plutonium dioxide. This material is blended with uranium dioxide to fabricate fuel for commercial reactors. Throughout the liquid extraction process and fuel fabrication, the plutonium and uranium compounds must be handled with care to avoid internal exposure. [redacted]

The MFFF implements an effluent control system to minimize the environmental release of radioactive materials. The ventilation system for each process area is designed with multiple filters to remove hazardous chemicals and radioactive materials. Off-gas scrubbers, high-efficiency particulate air (HEPA) filters, and effluent monitors are used to minimize radioactive releases. After passing through filtration, effluent gases are exhausted through the plant stack, which extends approximately 36.6 m (120 feet) above the surrounding landscape.

The emergency evaluation describes the MFFF location, population, and plant layout. The site has effluent controls, including a stack, which limit the release of radioactive material and reduce the potential for public exposure. The evaluation was prepared by the applicant to demonstrate compliance with the regulatory requirements in 10 CFR 70.22(i)(1)(i). The staff reviewed the emergency evaluation in accordance with the staff review guidance in Chapter 14.3.1 of NUREG-1718 (NRC, 2000). Details of the evaluation and the staff’s review follow. The facility processes large quantities of MOX material, but its location within the SRS facility and effluent control systems support the evaluation in accordance with regulatory requirements in and Chapter 14.2.1 of NUREG-1718.

14.3 Types of Accidents

14.3.1 Radiation Exposure

The applicant evaluated credible accident events to members of the public located on the SRS site boundary which is 6.05km (5 miles) from the MFFF. The evaluation consisted of a hazard analysis to identify general events that could lead to a significant exposure to radioactive material. Event sequences identified in the Integrated Safety Analysis Summary (MOX 2010) were evaluated under accident conditions (i.e., assuming the accidents occurred). The mitigating factors listed in 10 CFR § 70.22(i)(2) were used to demonstrate that postulated events would fall below the 10 CFR § 70.22(i)(1)(i) thresholds for radiation exposures (0.01 sieverts (1 rem)) and the chemical exposures (20 mg of soluble uranium) at the SRS boundary.

Internal exposures are the primary concern at the MFFF because the uranium and plutonium compounds are dominated by alpha radiation. The distance to the SRS site boundary would limit any direct radiation from material on site. MOX Services evaluated the release of plutonium and uranium compounds due to fires, spills, criticality, and explosions. The bounding events were determined based on radiological consequences identified in the hazards analyses, when considering 10 CFR § 70.22(i)1(i).

The bounding sequences for the four general accident types are addressed individually. Fires are limited to a single fire zone because of the combustible material available, fire detection, and fire barriers. Load-handling events (spills) involve a single container in a process unit and are further limited by restrictions on the lift height of containers. Criticality events are limited by the mass of material allowed in process units such as tanks, gloveboxes, and filters. Explosion releases are limited by the amount of material involved in a single container and the airborne release fractions for the type of material involved. For each of these event types, HEPA filtration through the heating, ventilation, and air conditioning (HVAC) system and the applicant's response resulting from early detection by radiation monitors would reduce the release.

The MFFF used a dispersion model to evaluate the X/Q dispersion factor and postulated doses at the SRS boundary. This evaluation determined the dose to an individual at the SRS boundary to be approximately 0.001 Sieverts (0.1 rem), well below the 0.01 Sieverts (1-rem) regulatory limit in 10 CFR § 70.22(i)(1)(i). This dispersion factor and dose estimate was confirmed by the applicant using RASCAL, as demonstrated in Enclosure 2 of the evaluation (MOX, 2009). Events evaluated in the ISA but not described in the ISA Summary, due to low-consequences as defined by 10 CFR § 70.61, "Performance Requirements," would not exceed the 0.01 Sieverts (1-rem) limit due to dispersion during propagation to the site boundary.

14.3.2 Uranium Exposure

Uranium exposure to a member of the public on the site boundary was evaluated for uranium dioxide and uranyl nitrate. The primary release mechanism was due to spills and fires. The applicant determined the maximum source term that would be released for each uranium compound. Once the source term was established, MOX Services analyzed the uranium consequence and dispersion at the SRS boundary using ALOHA9 and MACCS210 computer codes. For spills, the source term was estimated based on evaporation from a puddle formed near a leak. For fires, the maximum amount of material involved, the airborne release fractions, and respirable fractions were used to estimate the source term. [redacted] Such a fire event evaluated over 8 hours was calculated to result in an intake of 0.1 mg at the site boundary, well below the 2-mg regulatory limit in 10 CFR 70.22(i)(1)(i). The source term will be reduced further if the10 CFR 70.22(i)(2) criteria, such as limited tank inventory, leak detection, leak isolation, and operator action, is included in the evaluation. The calculations of the source terms are based on published airborne release and respirable fractions and a conservative quantity of material available for the vent. The dispersion factor is also reasonable because of the large distance from the MFFF and the site boundary (i.e., 5 miles). Based on these considerations, the staff finds that the evaluation provides reasonable assurance that a uranium release event would not result in a uranium exposure on the site boundary exceeding the 10 CFR 70.22(i)(1)(i) threshold of 2 mg.

14.3.3 Criticality

The MFFF evaluated a worst-case criticality accident, which would produce 1.0×10^{19} fissions with an initial burst of 1.0×10^{18} fissions and continuing to pulse for 8 hours. The dose was calculated based on exposure to material boiled from solution and the release of fission products. The criticality was assumed to occur in the largest MFFF process tank with standard variables drawn from NUREG/CR-6410, “Nuclear Fuel Cycle Facility Accident Analysis Handbook” (NRC, 1998). The inhalation and submersion doses were summed to obtain the dose to a member of the public on the site boundary of 0.000252 Sieverts (0.0252 rem). The airborne release fraction and dispersion for nonvolatiles would result in negligible uranium exposure at the site boundary. This evaluation did not credit criticality detection alarm systems or the HVAC HEPA filter system which would further minimize criticality impacts. The direct radiation would be rapidly attenuated, and dispersion would minimize internal exposure as confirmed by the applicant in Table 9-1 of the evaluation (MOX, 2009). Based on these considerations, the staff finds that the evaluation provides reasonable assurance that a criticality event would not exceed the 10 CFR 70.22(i)(1)(i) threshold on the site boundary.

14.3.4 Additional Considerations

The applicant also evaluated the impact of natural phenomena and external manmade events in its ISA Summary. The facility has been designed and built to meet the baseline design criteria as required in 10 CFR 70.64(a), “Requirements for New Facilities or New Processes at Existing Facilities.” The staff agrees with the applicant’s conclusion that the events addressed in the emergency evaluation (MOX, 2009) bound these accident scenarios.

14.4 Interaction with Savannah River Site and Offsite Officials

The applicant’s description of the internal emergency response plan includes participation in the SRS emergency response plan. The interaction between these two plans is defined by a work task agreement, which both parties have reviewed and approved. MOX Services remains responsible for contacting the NRC and DOE in case of an event, but interactions with State and local officials are conducted through the SRS Emergency Duty Officer who oversees the SRS Operations Center. The MFFF emergency preparedness program incorporates plans for radiation monitoring, repair and recovery efforts, search and rescue, and initial medical response. Additional resources are available upon request from the SRS. As part of the work task agreement, MFFF personnel are trained in emergency response, which includes participating in SRS drills and exercises. The applicant stated in the evaluation that they will work with the SRS Operations Center to verify that the SRS emergency plan incorporates MFFF emergencies. The MFFF maintains emergency procedures and coordinates with SRS emergency requirements to comply with NRC regulations while also complying with DOE requirements. Based on the review of the evaluation, the MFFF the internal emergency plan and procedure for interaction with the SRS Operations Center provides the staff reasonable assurance that the MFFF has adequate emergency capability in compliance with 10 CFR § 70.64(a)(6).

14.5 Evaluation

The staff has reviewed MOX Services dose calculations and determined that (1) the calculated dose to the offsite public is reasonable and conservative, (2) the dose to the offsite public is less than 0.01 Sieverts (1 rem) effective dose equivalent or an intake of 2 mg of soluble uranium, (3) the accident scenarios chosen by the applicant are credible and bounding based on regulatory requirements; and (4) no formal emergency plan is required in accordance with 10 CFR § 70.22(i)(1)(i).

The staff notes that although an NRC-approved emergency plan is not required by 10 CFR 70.22, “Contents of Applications,” the MFFF has committed to maintaining an emergency plan, implementing procedures, and emergency response organization for internal use. The staff considers this commitment prudent and acceptable. Based on this evaluation, the staff concludes that the MFFF has demonstrated reasonable assurance of compliance with 10 CFR 70.22(i)(1)(i), 10 CFR 70.64(a)(6), and the performance criteria in NUREG-1718, Sections 14.4.1 through 14.4.3.1.4 (NRC, 2000).

REFERENCES

(MOX, 2009a) Shaw AREVA MOX Services, “Emergency Plan Evaluation,” Aiken, SC, September 14, 2009.

(MOX, 2010) Shaw AREVA MOX Services (MOX Services). “Mixed Oxide Fuel Fabrication Facility Integrated Safety Analysis Summary,” Aiken, SC, March 2010.

(NRC, 2000) U.S. Nuclear Regulatory Commission, NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility,” Washington, DC, August 2000).

(NRC, 1998) U.S. Nuclear Regulatory Commission, NUREG/CR-6410, “Nuclear Fuel Cycle Facility Accident Analysis Handbook,” Washington, DC, March 1998.

10 CFR Part 70 – Domestic Licensing of Special Nuclear Material.

15.0 Management Measures

Management measures are functions that MOX Services performs, generally on a continuing basis, that are applied to items relied on for safety (IROFS), to ensure the items are available and reliable to perform their safety functions when needed. Management measures shall be implemented to ensure compliance with the performance requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 70.61, "Performance requirements," and the degree to which they will be applied will be a function of the item's importance in terms of meeting performance requirements, as evaluated in the integrated safety analysis (ISA). This chapter addresses each of the management measures included in the definition of management measures in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," including (1) configuration management (CM), (2) maintenance, (3) training and qualifications, (4) procedures, (5) audits and assessments, (6) incident investigations, (7) records management, and (8) other quality assurance (QA) elements.

The purpose of this review is to verify whether the MOX Services license application (MOX, 2010a) provided conclusive information to demonstrate that the management measures applied to IROFS, as documented in the ISA Summary, provide adequate assurance that the IROFS will be available and reliable and will function according to the performance requirements of 10 CFR § 70.61.

Quality level definitions and the requirements for applying graded QA to IROFS are found in the MOX Services Project Quality Assurance Plan (MPQAP), which the staff of the U.S. Nuclear Regulatory Commission (NRC) has reviewed and approved. Revision 8 of the MPQAP was reviewed and accepted by the staff, as documented in letter dated October 19, 2009 (ML092790580).

15.1 Regulatory Requirements

The requirements in 10 CFR Part 70 specify fuel cycle facility management measures, as follows:

- 10 CFR § 70.4, "Definitions," states that management measures include CM, maintenance, training and qualifications, procedures, audits and assessments, incident investigations, records management, and other QA elements.

- 10 CFR § 70.22(a)(8), Contents of applications," requires that each application for a license contain proposed procedures to protect health and minimize danger to life or property.

- 10 CFR § 70.62(a)(3), "Safety program and integrated safety analysis," states that records must be kept for all IROFS failures, describes required data to be reported, and sets time requirements for updating the records.

- 10 CFR § 70.62(d) requires an applicant to establish management measures for engineered and administrative controls (ACs) and control systems that are identified as IROFS, pursuant to 10 CFR 70.61(e), so that they are available and reliable to perform their functions when needed.

- 10 CFR § 70.64(a)(1), "Requirements for new facilities or new processes at existing facilities," states that new facilities or new processes at existing facilities shall develop and implement designs in accordance with management measures, to provide adequate assurance that IROFS will be available and reliable to perform their safety function when needed.

- 10 CFR § 70.64(a)(1) states that appropriate records of IROFS must be maintained by, or under the control of, the licensee throughout the life of the facility.

- 10 CFR § 70.64(a)(8) states that the design of IROFS must provide for inspection, testing, and maintenance adequate to ensure their availability and reliability to perform their function when needed.

- Facility changes and change processes are required to conform to 10 CFR § 70.72, "Facility Changes and Change Process."

- 10 CFR § 70.74(a) and (b), "Additional reporting requirements," require incident investigation and reporting.

- In addition, an applicant to possess and use special nuclear material (SNM) in a plutonium processing and fuel fabrication facility such as the mixed oxide (MOX) fuel fabrication facility (MFFF) is required, pursuant to 10 CFR § 70.22(f), to describe the QA program to be applied to the design, fabrication, construction, testing, and operation of the structures, systems, and components (SSCs) of the facility.

- 10 CFR Part 21, "Reporting of Defects and Noncompliance," contains additional pertinent regulatory requirements for identifying, controlling, and reporting defects with a facility, activity, or basic component supplied to a facility licensed under 10 CFR Part 70.

15.2 Regulatory Acceptance Criteria

Section 15, "Management Measures," of NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility" (NRC, 2000), contains the acceptance criteria for the NRC review of MOX Service's management measures program.

15.3 Staff Review and Analysis

15.3.0 Management Measures

The applicant established management measures to ensure that facility IROFS would be available and reliable to perform their safety function when needed and to ensure that work is conducted efficiently and in a manner that protects workers, the public, and the environment. The applicant describes its management measures as a framework of administrative and programmatic measures that includes CM, maintenance, training and qualification, procedures, audits and assessments, incident investigations, and records management.

The applicant commits to implementing management measures in accordance with a QA program established in accordance with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

The applicant commits to applying management measures to IROFS to ensure that they are available and reliable upon demand. The applicant assigns management measures for training based on four types of IROFS classifications and the risk-reduction level attributed to the IROFS. The types of IROFS classifications are as follows:

(1) passive engineered controls (PECs), which are devices that use only fixed physical design features to maintain safe process conditions and require no human action

(2) active engineered controls (AECs), which are physical devices that use active sensors, electrical components, or moving parts to maintain safe process conditions and which require no human action

(3) enhanced ACs (EACs), which are procedurally required or prohibited human actions, combined with a physical device that alerts the operator that the action is needed to maintain safe process conditions, or otherwise adds substantial assurance of the required human performance

(4) ACs, which are procedurally required or prohibited human actions needed to maintain safe process conditions

The applicant identifies the specific elements of the various management measures programs assigned to each IROFS classification in the license application. This illustrates how the various management measures elements apply to the different IROFS classifications (i.e., PECs, EACs, AECs, and ACs). The applicant describes the application of EACs and states that, for the EAC, the specific management measures for the physical device are covered under the AEC classification.

15.3.1 Quality Assurance

In accordance with Option A of NUREG-1718, the applicant elects to implement and maintain a QA program in conformance with the applicable requirements of Parts I and II of ASME-NQA-1-1994, as revised by the ASME NQA-1a-1995 (ASME, 1995) Addenda or equivalent. The MOX Services QA program is described in the MPQAP, which the NRC approved in a safety evaluation report dated October 1, 2001 (NRC, 2001). The MPQAP establishes the QA requirements to control quality-affecting activities related to the design, construction, and operation of the MFFF. By letter dated October 19, 2009 (NRC, 2009), the staff documented its review and approval of the latest revision to the MPQAP, Revision 8.

The applicant commits to submitting any change that would reduce the commitments of the NRC-approved QA program, along with written justification for the change, to the NRC for acceptance before implementing it. The applicant states that it will update the MPQAP, as necessary, during testing, operation, and deactivation of the MFFF. The applicant commits to implementing the requirements of 10 CFR Part 21 for design, construction, procurement, testing, and operations of Quality Level 1 SSCs (i.e., IROFS). Section 4 of the MPQAP, "Procurement Document Control," requires that 10 CFR Part 21 be invoked for procurements of IROFS for the MFFF, unless the procurement is for a commercial grade item.

15.3.2 Configuration Management

15.3.2.1 Configuration Management Policy

The applicant states that the CM program for MFFF will ensure that IROFS are designed and operated within the design basis. As stated in the application, the MFFF CM program will identify and control the preparation and review of documentation associated with IROFS, control changes to IROFS, and maintain the physical configuration of the facility consistent with the approved design.

The applicant states that it accomplishes configuration control during design through the use of procedures for controlling design activities. These design control procedures will encompass activities related to the preparation, review, verification (where appropriate), approval, release, and distribution of the design for use. The applicant states that changes to the approved design are subject to a review to ensure consistency with the design bases of the IROFS.

The applicant has established quality level classifications for MFFF SSCs and associated documents, as documented in Section 2.2 of the MPQAP. As described in the MPQAP, quality levels are assigned to SSCs commensurate with their safety significance and a combination of the likelihood and consequences of design basis events. Quality Level 1 (QL-1) SSCs are IROFS credited in the Integrated Safety Analysis with a required function to prevent or mitigate design basis events such that high-consequence events are made highly unlikely; intermediate-consequence events are made unlikely; or to prevent criticality. Quality Level 2 (QL-2) SSCs are not relied on to satisfy the performance requirements of 10 CFR 70.61; they perform functions such as maintaining public and worker radiological exposure within normal operating limits; monitoring and alerting personnel of changes to facility conditions, such as criticality; managing radioactive waste; and protecting IROFS from potentially harmful physical interactions. Quality Level 3 (QL-3) SSCs have no safety function but their performance may be important to ensuring operational or mission-critical goals are achieved. Finally, Quality Level 4 (QL-4) SSCs are those SSCs that are not designated as QL-1, QL-2, or QL-3, and controls on those SSCs do not impact the regulatory basis of the MFFF.

The applicant states that it will accomplish CM through design review and verification, which ensure that design documents are consistent and that design requirements for IROFS are met. The applicant states that changes identified during construction or testing must be approved by the Engineering Department through a documented engineering change process or an approved nonconformance report before implementation to ensure that testing is successfully accomplished and that configuration is maintained.

The applicant commits to conducting initial and periodic assessments of the CM system to determine its effectiveness and to correct deficiencies. The applicant states that it will conduct audits and assessments of the CM program to ensure that the program meets its goals and that the design is consistent with the design bases. The applicant also states that it will perform the corrective action process in accordance with the MPQAP and associated procedures, in the event that any problems are identified. The applicant will develop prompt corrective actions in response to audit or assessment findings or as a result of incident investigations.

The applicant states that it will maintain CM as the project progresses from design and construction to operations and will establish procedures to define the turnover responsibilities and processes.

The applicant demonstrates that it has established design requirements and associated design bases and that the appropriate organizational unit maintains them. The applicant states that the Functional Area Manager will approve procedures and revisions thereto for facility modifications made during the operations phase. Change procedures will ensure quality in the facility modification program and will include technical and quality requirements necessary to implement a modification, as well as the requirements for initiating, approving, monitoring, designing, verifying, and documenting modifications. The applicant also states that it will write the facility modification procedure to ensure that policies are formulated and maintained to satisfy the MPQAP, as applicable.

The applicant describes its compliance with the provisions of 10 CFR § 70.72 and commits to ensuring that each change to the MFFF or to activities of personnel during operations will have an evaluation performed in accordance with the requirements of 10 CFR § 70.72, as applicable. The applicant also states that it will evaluate each modification for any required changes or additions to the facility's procedures, personnel training, testing program, or regulatory documents, as applicable.

The applicant demonstrates that the CM system will ensure that the appropriate technical, management, and safety reviews are performed in support of changes to the MFFF and its IROFS. In describing its change control process, the applicant states that it will require that the impacts of any change (i.e., new design or operation) or modification to the facility or to activities of personnel (e.g., site SSCs, computer programs, processes, operating procedures, management measures) that involve or could affect the ISA, be evaluated and documented. The change control process also requires that, before implementing any change, the applicant shall demonstrate that it does not affect the safety basis, in accordance with 10 CFR § 70.72. Changes that affect the safety basis require NRC approval before implementation.

In the application, the applicant states that it will evaluate and document each modification to the facility or to activities of personnel for radiation exposure to minimize worker exposures as part of the facility's as-low-as-reasonably-achievable program (ALARA), criticality and worker safety requirements, or restrictions. The applicant states that it may also evaluate modifications in terms of cost; lessons learned from similar completed modifications; QA requirements; potential operability, maintainability, or constructability concerns; postmodification testing requirements; environmental considerations; human factors; and the ISA, as applicable.

The applicant commits to post-modification testing of items in addition to established periodic performance monitoring and maintenance functions. The applicant states that, upon completion of a modification to an SSC, the modification's responsible manager, or designee, will confirm that applicable testing has been completed to ensure correct operation of the system(s) affected by the modification. The applicant adds that the responsible manager will also ensure that documentation regarding the modification is complete. Documents such as the revised process description, checklists for operation, and flowsheets will be made available to operations and maintenance departments before the startup of the modified system to ensure that operators are able to operate the modified system safely. The applicant also states that it will complete the appropriate training on the modification before a system is placed in operation and will distribute a formal notice of the modification completion to appropriate managers. The applicant will complete drawings incorporating the modification, in accordance with MFFF design control procedures, and will retain identifiable records related to the modification, in accordance with the MFFF records management procedures.

15.3.2.2 Implementation of Configuration Management

As stated in the application, during the design phase of the project, the applicant will base CM on the design control provisions and associated procedural controls over design documents to establish and maintain the technical baseline. The applicant states that it will identify documents that provide design input, analysis, or results specifically for IROFS, including the ISA, with the appropriate quality level. The applicant states that these design documents will undergo interdisciplinary review during the initial issue and during each subsequent revision.

During the construction phase of the project, the applicant states that it systematically reviews and verifies changes to drawings and specifications issued for construction, procurement, or fabrication; evaluates changes for impact to the ISA; and approves the changes before implementing them. The applicant commits to verifying proper implementation of such changes by the QA organization.

The applicant states that it will implement measures to ensure that the quality of MFFF IROFS is not compromised by planned changes (modifications). These measures will include assigning responsibility for the design of and modifications to facility IROFS to the Plant Manager. These measures will also include performing the design and implementation of modifications so as to ensure quality is maintained in a manner commensurate with the remainder of the system that is being modified, or as dictated by applicable regulations.

15.3.2.3 Organization

The applicant describes the organizational structure and staffing interfaces of the CM system. The President of MOX Services is responsible for the overall implementation of the CM program, including development and approval of plans and policies necessary to provide overall program direction. The Vice President—Engineering administers the CM program during design, and the engineering organization includes engineering disciplines. Discipline engineers have primary technical responsibility for the work performed within their disciplines. Responsibility for interdisciplinary reviews lies with the responsible managers. Reviews are also conducted, as appropriate, by construction management, operations, environmental safety and health, QA, and support services personnel.

The applicant commits to an acceptable method of controlling and storing documents within the CM system. The applicant states that the MFFF design control process interfaces with the document control and records management process through procedures. The applicant's document control program includes provisions for the inclusion of documents in the MOX Services electronic document management system (Documentum), maintenance and distribution of documents, document retention, tracking of document change status, and document retrieval.

The Vice President—Construction is responsible for CM during construction and establishes and maintains processes and procedures used during construction of the facility.

The Plant Manager is responsible for ensuring the implementation of CM during operational testing, operation, and deactivation of the MFFF.

The various MOX Services departments and subcontractors perform quality-related activities, and the primary MOX Services subcontractors work to the MPQAP. Some MOX Services subcontractors will develop and implement their respective QA programs in a manner that is consistent with the requirements of the MPQAP for activities determined to be within the scope of the MPQAP. The interfaces between subcontractors and MOX Services or among subcontractors will be documented. MOX Services and subcontracted personnel have the responsibility to identify quality problems. Disagreements that cannot be resolved will be elevated to the next level of management for resolution and, if necessary, through successive layers of management until resolution is achieved.

15.3.2.4 Scope of Configuration Management Program

The applicant clearly defines the IROFS to be included in the scope of the CM program. The MFFF CM program includes the IROFS identified by the ISA and any items that may affect the safety function of the IROFS. The applicant also shows that the CM system will consistently capture documents that are relevant and important to safety as the project evolves from design and construction through operations. The applicant states that calculations, safety analyses, design criteria, engineering drawings, system descriptions, technical documents, operating procedures, and specifications that establish design and safety requirements for IROFS are also subject to CM. During the design phase, these documents are maintained under CM upon initial approval.

The applicant's design process leading to drawings and other statements of requirements proceeds logically from the MFFF design basis. The applicant states that the number of documents included in the CM program will increase throughout the design process as drawings and specifications related to IROFS are prepared and issued for procurement, fabrication, or construction. The related documents are included in CM.

The applicant states that, during construction, initial startup, and operations, the scope of documents under CM will continue to increase and will include, as appropriate: vendor data; test data; inspection data; initial startup, test, operating, and administrative procedures, as applicable to IROFS; and nonconformance reports. These documents will be generated through functional interfaces with QA, maintenance, and personnel training and qualification. The applicant commits to establishing CM procedures that evaluate, implement, and track changes to IROFS, as well as processes, equipment, computer programs, and activities of personnel that affect IROFS.

The applicant states that CM is implemented through or related to other management measures. The applicant identifies key interfaces and the relationship of CM to other management measures, including QA, records management, maintenance, training and qualifications, audits and assessments, and procedures.

The applicant states that the MFFF QA program establishes the framework for CM and other management measures for IROFS and items that affect the function of the IROFS.

The applicant commits to generating and processing records associated with IROFS, in accordance with the applicable requirements of the QA program. The applicant also commits to providing evidence of the conduct of activities associated with the CM of those IROFS.

The applicant commits to the establishment of maintenance requirements as part of the design basis, which is controlled under CM. The applicant will maintain records sufficient to provide evidence of compliance with preventative and corrective maintenance schedules for IROFS.

The applicant states that it will control personnel training and qualification in accordance with approved project procedures. Personnel qualifications and training to specific processes and procedures are management measures that support the safe design, operation, maintenance, and testing of IROFS. The applicant commits to developing and implementing procedures for work activities that are themselves IROFS (i.e., ACs) and to training and qualifying personnel to these procedures. The applicant also states that training and qualification requirements and documentation of training may be considered part of the design basis and be controlled under CM.

The applicant describes the interface between CM and audits, assessments, and incident investigations and applies its audit and assessment activities to the CM program, which includes the control of design requirements and the implementation of those requirements. The applicant states that corrective actions identified as a result of the management measures of audits, assessments, and incident investigations may result in changes to design features, ACs, or other management measures (e.g., operating procedures). The applicant commits to using the MFFF QA program and procedures to evaluate changes to maintain CM and to conducting periodic assessments of the CM program, in accordance with the audit and assessment program. The audit and assessment program includes requirements to perform both document assessments and physical assessments (walkdowns) to check the adequacy of the CM system and to document assessment and follow-up activities.

The applicant states that it will use operating, administrative, maintenance, and emergency procedures to conduct various operations associated with IROFS and will review these procedures as part of the CM program to identify potential impacts to the design basis. The applicant also states that work activities that are themselves IROFS (i.e., ACs) will be contained in procedures.

15.3.2.5 Change Control

The applicant fully describes the activities that comprise its CM program. According to the applicant, CM includes those activities conducted under design control provisions to ensure that design and construction documentation is prepared, reviewed, and approved in accordance with a systematic process. This process includes interdisciplinary reviews appropriate to ensure consistency between the design and the design bases of IROFS. During construction, it also includes those activities that ensure that construction is consistent with design documents. Finally, it includes activities that provide for operation of the IROFS in accordance with the limits and constraints established in the ISA and that provide for the control of changes to the facility in accordance with 10 CFR § 70.72.

The applicant states that CM also includes records to demonstrate that personnel conducting activities that are IROFS are appropriately qualified and trained to perform that work.

The applicant commits to applying the MFFF document control system to the control of implementing documents as a means of controlling documents within the CM system. The applicant states that implementing documents include those documents that support CM by ensuring that only reviewed and approved procedures, specifications, and drawings are used for procurement, construction, installation, testing, operation, and maintenance of IROFS, as appropriate.

The applicant demonstrates that the CM system provides for keeping design requirements and the safety assessment of the ISA current and ensures that suitable hazard and accident analysis methods are available to evaluate safety margins of proposed changes. The applicant states that it uses procedures to control changes to the design documents, and the change process includes an appropriate level of technical, management, and safety review and approval before implementation. During the design phase of the project, the method of controlling changes is the design control process described in the implementing procedures. This process includes conducting interdisciplinary reviews and design reviews and verifications that constitute a primary mechanism for ensuring that the design is consistent with the design bases. During both the construction and operations phases of the facility, the applicant will use appropriate reviews to ensure consistency with the design bases of IROFS and the ISA, respectively, to ensure that the design is constructed and operated or modified within the limits of the design basis.

The applicant commits to performing a systematic review of the design bases when making changes to the design, to ensure consistency. In the event that changes reflect design or operational changes from the established design bases, the applicant commits to properly modifying, reviewing, and approving the ISA before the change is implemented. The applicant states that it will make approved changes available to personnel through the established document control function.

During design, the applicant commits to using the interdisciplinary review process as the method of ensuring consistency between documents, including consistency between design changes and the safety analyses. The applicant asserts that interdisciplinary reviews ensure that design changes: (1) do not affect the ISA, (2) are accounted for in subsequent changes to the ISA, or (3) are not approved or implemented. Before issuance of the license, MOX Services commits to notifying the NRC of potential changes that reduce the level of commitments or margin of safety in the design bases of IROFS.

When the project enters the construction phase, the applicant will document, review, approve, and post changes to documents issued for construction, fabrication, and procurement against each affected design document. Vendor drawings and data will also undergo an interdisciplinary review to ensure compliance with procurement specifications and drawings and to incorporate interface requirements into facility documents.

During construction, the applicant will continue to evaluate design changes against the approved design bases. The applicant expects changes to the design as detailed design and construction activities progress and states that it will use a systematic process, consistent with that described above, to evaluate changes in the design against the design bases of IROFS and the ISA.

Upon issuance of the MFFF Possession and Use License, the applicant states that the configuration change process will fully implement the provisions of 10 CFR § 70.72, including reporting changes made without prior NRC approval, as required by 10 CFR § 70.72(d)(2) and (3). The applicant also states that it will submit any change that requires Commission approval as a license amendment request, as required by 10 CFR § 70.72(d)(1), and that it will not implement the change without prior NRC approval.

During the operations phase, the applicant commits to documenting, reviewing, and approving changes to the design before implementation. The applicant also commits to using a change process that fully implements the provisions of 10 CFR § 70.72. The applicant states that it will make responsible facility personnel aware of design changes and modifications that may affect the performance of their duties.

The applicant assigns specific personnel the responsibility for maintaining the design bases and requirements. The applicant states that, upon acceptance by Operations, the Plant Manager will be responsible for the design of and modifications to IROFS and for designing and implementing modifications so as to ensure that quality is maintained in the remainder of the system, or as dictated by applicable regulations.

The applicant commits to applying CM controls incorporated into the original design and modifications throughout operations to facilitate deactivation of the facility.

The applicant describes its technical management review and approval procedure. The applicant states that the Functional Area Manager approves the administrative instructions for modifications contained in a facility administrative procedure, including revisions. The applicant states that the modification procedure contains (1) the technical and quality requirements that shall be met to implement a modification and (2) requirements for initiating, approving, monitoring, designing, verifying, and documenting modifications. The applicant maintains that the facility modification procedure will be written to ensure that policies are formulated and maintained to satisfy the MPQAPas applicable and that QA is ensured.

The applicant commits to performing an evaluation of each change to the facility or to activities of personnel, in accordance with the requirements of 10 CFR § 70.72, as applicable. The applicant also commits to the evaluation of modifications to identify any required changes or additions to the facility's procedures, personnel training, testing program, or regulatory documents.

The applicant states that it will evaluate and document the impacts of changes (e.g., new design or operation, or modification to the facility or to activities of personnel, IROFS, computer programs, processes, operating procedures, management measures) that involve or could affect the ISA. The applicant also maintains that, before implementing any change, it will demonstrate that the change does not affect the safety basis, in accordance with 10 CFR § 70.72.

15.3.2.5.1 Identification of Changes

The applicant states that design bases and design requirements that are derived from the design bases will be established and maintained by the engineering organization during design and construction and by the Plant Manager during operations. The applicant will document the design bases in licensing bases documents and in design documents such as calculations, safety analysis, engineering drawings, system descriptions, technical documents, and specifications. The applicant commits to controlling design documents under the design control provisions of the CM program.

The applicant describes the quality levels and CM controls assigned to IROFS. The applicant has designated all IROFS as Quality Level 1 and commits to performing interdisciplinary reviews and design review and verification activities for design documents associated with IROFS, as well as for analyses constituting the ISA. The applicant summarizes IROFS in the ISA Summary and commits to evaluating changes to the design to ensure consistency with the design bases.

The applicant demonstrates that suitable design control analysis methods are available to evaluate the safety margins of proposed changes and describes the methods applied to control computer codes used for such evaluations. The applicant states that it will subject computer codes used in safety analyses and the design of IROFS to the same design control measures as IROFS and ISA analyses, with additional requirements, as appropriate, for software control, verification, and validation.

The applicant describes personnel responsibilities for maintaining design bases and requirements. The applicant states that qualified individuals will prepare design documents (e.g., calculations, specifications, procedures, or drawings) and will specify and include the appropriate codes, standards, and license requirements within the design documents. The applicant states that these individuals will note any deviations or changes from such standards within the design documentation package.

The applicant identifies its process for the review and approval of design documents. After the preparation of design documents by qualified individuals, each design document is reviewed by another individual qualified in the same discipline. The applicant states that design inputs will be sufficiently detailed so as to permit verification of the document. The manager having overall responsibility for the specific design function will then approve the document and will record the entire review process in accordance with approved procedures. The applicant's procedures will include provisions to ensure that design documents specify the appropriate quality standards, including quantitative or qualitative acceptance criteria. The QA Manager will conduct audits on the design control process using independent technically qualified individuals to augment the QA audit team.

During the review of design documents, the applicant commits to emphasizing conformance with applicable codes, standards, and license application design commitments. The applicant grants full and independent authority to engineering personnel assigned to perform document reviews such that review personnel may withhold approval of design documents until questions concerning the work have been resolved.

The applicant will accomplish the design verification function through design reviews, alternative calculations, or qualification testing. The applicant requires that (1) the bases for a design, such as analytical models, theories, examples, tables, codes, and computer programs, be referenced in the design document, and (2) the application of such bases be verified during check and review. The applicant states that the responsible qualified individual will review and approve model tests when such tests are required to prove the adequacy of a concept or a design. The applicant commits to applying design verification testing to demonstrate adequate performance under conditions that simulate the most adverse design conditions. The applicant states that tests used for design verification will meet the design requirements.

The applicant will use qualified individuals other than those who prepared the design to verify it. MFFF personnel from the same organization as those who prepared the design may verify it; the supervisor of the individual who prepared the design may verify it, provided that the supervisor did not specify a singular design approach or rule out certain design considerations and did not establish the design inputs.

The applicant commits to accomplishing independent design verification before use of the design document (or information contained therein) by other organizations for design work or to support other activities, such as procurement, construction, or installation. The applicant states that, when this is not practical because of time constraints, it will identify and control the unverified portion of the document; however, the applicant commits to completing all design verification activities before relying on an item to perform its function. The applicant requires that the review and approval of changes to design and procurement documents be commensurate with the original approval requirements. This requirement applies to all changes, including field changes.

15.3.2.5.2 Review and Approval of Changes

The applicant demonstrates that the CM system will maintain strict consistency among design requirements, physical configuration, and facility documentation. The applicant will accomplish configuration control during design through the use of design control procedures. These procedures include controls for design preparation, review (including interdisciplinary review and preparation of nuclear safety evaluations (NSEs) and nuclear criticality safety (NCS) evaluations (NCSEs), as applicable), verification, approval, and release and distribution for use. The applicant will assess engineering documents for quality level classification and will review changes to the approved design to ensure consistency with the design bases of IROFS.

The applicant will apply design verification in the CM program to ensure that design documents are consistent and that design requirements for IROFS are met. The applicant states that the construction and quality control organizations will conduct in-process verifications during construction, and the startup and quality organizations will verify configuration during testing to demonstrate the performance of IROFS.

The applicant states that the MPQAP will require the use of procedures to ensure that work is accomplished in accordance with the requirements and guidelines imposed by applicable specifications, drawings, codes, standards, regulations, QA criteria, and site characteristics.

The applicant will incorporate acceptance criteria established by the designer into the instructions, procedures, and drawings used to perform work at the MFFF. The applicant commits to maintaining documentation, such as test results and inspection records, to demonstrate the proper performance of work activities. The applicant also states that MFFF procedures will provide for review, audit, approval, and documentation of activities affecting the quality of items to ensure that applicable criteria have been met.

The applicant establishes measures to review procedures to ensure that current maintenance, operations, and other facility procedures reflect any modifications to facility IROFS. The applicant states that qualified personnel knowledgeable in the QA disciplines will review MFFF maintenance, modification, and inspection procedures to determine the need for inspection, identification of inspection personnel, and documentation of inspection results. The review will also verify that applicable procedures have identified the necessary inspection requirements, methods, and acceptance criteria. The applicant commits to reviewing facility procedures on a frequency based on the age and use of the procedure to determine if changes are necessary or desirable and to ensuring that procedures are kept current with the facility configuration. The applicant states that procedure reviews will be conducted by individuals knowledgeable in the area(s) affected by each procedure.

15.3.2.5.3 Implementation of Changes

The applicant describes its process for tracking, implementing, documenting, and distributing changes to design requirements and facility documentation, including the placement of documents into a document control center and plans to disseminate these changes to affected functions within the facility. The applicant states that, after the appropriate parties have properly prepared, reviewed, and approved design documents, the responsible engineer will send them to document control for distribution. After the document is entered into Documentum, it will be electronically routed (distributed) to employees identified on the record submittal form.

The applicant describes its process for identifying, authorizing, and implementing changes to design requirements and facility documentation in the event that it identifies deficiencies that affect the design of IROFS. The applicant states that it documents and resolves such deficiencies in accordance with approved corrective action program (CAP) procedures. In accordance with the CAP, the deficiency report documenting the inadequacy is forwarded for appropriate review to the responsible manager, who coordinates further review of the problem and revises the design documents affected by the deficiency. Where required, the responsible manager will forward the report to engineers in other areas to enable them to coordinate necessary revisions to their affected documents.

The applicant states that design interfaces will be maintained by communication among the Functional Area Managers. The applicant describes the methods used to accomplish effective communication among design interfaces. The responsible engineer or authorized representative will review design documents. Project interface meetings will provide the primary working interface among the MFFF organizations and will be scheduled and held to coordinate design, procurement, construction, and preoperational testing of the facility. In addition to document review activities and project interface meetings, the applicant will maintain design interfaces by using procedures to establish policies for the transmittal and control of nonconformance reports.

The applicant commits to establishing measures for MFFF operations to ensure responsible facility personnel are made aware of design changes and modifications that may affect the performance of their duties.

15.3.2.6 Document Control

15.3.2.6.1 Storage of Documents

The applicant commits to establishing procedures to control the preparation, issuance, and revision of documents, such as manuals, instructions, drawings, procedures, specifications, procurement documents, and supplier-provided documents. The applicant also commits to establishing measures to ensure that documents, including revisions thereto, will be adequately reviewed, approved, and released for use by authorized personnel.

In the MFFF electronic document control and storage system, the applicant states that approved documents included in the CM program will be stored in Documentum, which is a tool capable of reporting the status of documents. The applicant commits to storing records not suitable for storage in Documentum in accordance with the requirements of MPQAP Section 6, “Document Control.”

The applicant states that document control procedures will require documents to be transmitted and received in a timely manner at appropriate locations (including the location where the prescribed activity will be performed) to ensure that controlled copies of documents and their revisions are distributed to and used by the persons performing the activities.

As stated in the application, the MFFF will retain superseded documents within Documentum and control them through document control. The applicant commits to generating indexes of current documents using Documentum.

15.3.2.6.2 Identification of Documents

The applicant describes procedures that it will use to implement the document control program at the MFFF. The applicant states that it will implement approved procedures to track and retrieve current documents, historical records, and other information included in the CM program by attributes such as document number, document subject, component number, component name, and status. The applicant also states that the MFFF document control system will be capable of generating indices of controlled documents, which will be uniquely numbered (including revision numbers).

The applicant commits to maintaining controlled documents until they are cancelled or superseded, after which the applicant commits to maintaining the documents as records for the life of the project or until termination of the license, whichever occurs later. The applicant commits to distributing controlled documents in hard-copy format when needed, in accordance with applicable procedures (e.g., when the electronic document management system (EDMS) is not available).

The applicant defines documents that will be controlled at the MFFF. These documents will include design requirements; the ISA, through the controlled copies of supporting analyses; NSEs and NCSEs; drawings; specifications; calculations; technical reports; project procedures; QA documents; maintenance documents; audit and assessment reports; operating procedures; emergency response plans; and system modification documents.

15.3.2.7 Audits and Assessments

The applicant commits to performing initial assessment(s) of the CM program as part of system turnover upon entering the operations phase. The applicant further commits to performing periodic assessments of the CM and design control program to determine the system's effectiveness and to correct deficiencies. The applicant states that assessments will include a review of the adequacy of documentation and will be scheduled, conducted, and documented in accordance with approved procedures.

As stated in Section 15.2.1, "Configuration Management Policy," of the license application and reiterated in Section 15.2.7, the applicant commits to ensuring that the system meets its goals and that the design is consistent with the design bases through periodic audits and assessments of the CM program and of the design. The applicant states that it will perform incident investigations in accordance with the MPQAP and associated CAP procedures in the event problems are encountered. When needed as a result of incident investigations or in response to adverse audit or assessment results, the applicant commits to developing prompt corrective actions in accordance with CAP procedures.

15.3.3 Maintenance

The applicant describes the maintenance and functional testing programs that will be implemented for the operations phase of the facility in Section 15.3 of the application. The applicant states that it will develop the maintenance program using information from sources such as equipment suppliers, reference plants, and lessons learned from other appropriate facilities. The preventive and corrective maintenance activities, surveillance activities, and performance trending, as discussed in this section, will provide reasonable and continuing assurance that IROFS will be available and reliable to perform their safety functions commensurate with the risk levels identified in the ISA.

The applicant commits to providing and implementing measures that ensure that: (1) the safe and reliable operation of IROFS is continued, (2) the quality of the IROFS is not compromised by planned changes (modifications) or maintenance activities, and (3) quality will be maintained, in accordance with the quality requirements of the system under modification or as required by applicable regulations. As stated in the application, the Plant Manager will be responsible for the design of, and any modifications to, IROFS, as well as for maintenance activities performed during operations. The Plant Manager will also be responsible for ensuring the operational readiness of IROFS during maintenance.

The applicant will develop and maintain IROFS so as to maximize their availability and reliability. The applicant commits to performing planned and scheduled maintenance of IROFS to ensure that they remain in a condition of readiness to perform their planned and designed functions when required. As stated in the application, the applicant will perform maintenance activities in accordance with approved procedures that meet the applicable requirements of the MPQAP. However, planning, scheduling, coordinating, and tracking work activities to completion, maintaining data analysis records, and trending equipment performance will be the responsibility of a work management group that will be compiled by the applicant. The work management group will also be responsible for the assessments of any recommendations or corrective actions identified by the Incident Investigations Program.

15.3.3.1 Maintenance Categories

The applicant's maintenance activities are categorized into four general areas or programs: (1) surveillance and monitoring, (2) preventive maintenance (PM), (3) corrective maintenance, and (4) functional testing. The applicant commits to performing audits and assessments of the maintenance activities to ensure the effectiveness of the maintenance function.

15.3.3.1.1 Surveillance and Monitoring

The applicant states the general purpose of the surveillance and monitoring program is to measure the degree to which IROFS meet performance specifications and detect degradation and adverse trends of IROFS. The applicant states that it will use data sources such as surveillances, periodic and diagnostic test results, plant computer information, operator rounds, walkdowns, as-found conditions, failure trending, and predictive maintenance to select parameters to be monitored. As stated by the applicant, these parameters will be selected based upon their ability to detect the predominant failure modes of the critical components.

Surveillances, as stated by the applicant, may consist of measurements, inspections, functional tests, and calibration checks. The applicant will conduct surveillances at specified intervals and will trend results. The applicant states that PM frequencies will be adjusted and appropriate corrective actions implemented when trending identifies the degradation of IROFS. Incident investigations may be used, as stated in the application, to identify the root causes of failures that are related to the type or frequency of maintenance performed. The lessons learned from such investigations will be factored into the surveillance and monitoring and PM programs, as appropriate.

The applicant states that it will establish criteria to monitor plant performance, IROFS functions, and component parameters. The applicant commits to establishing maintenance procedures that include appropriate compensatory measures for surveillance tests of IROFS that can be performed only while equipment is out of service.

The applicant commits to maintaining records identifying the current surveillance schedule, performance criteria, and test results for IROFS, in accordance with the record management system.

15.3.3.1.2 Preventive Maintenance

The applicant provides a description of the PM program, including the commitment to conduct preplanned and scheduled periodic refurbishment, partial or complete overhauls, or replacement of IROFS, as necessary, to ensure the continued safety function of IROFS, even with unplanned outages. The applicant states that it will consider the results of surveillance and monitoring activities, in addition to any failure history, during PM planning.

As part of PM activities, the applicant states that it will address instrumentation calibration and testing through procedures and calibration standards traceable to the national standards system. The applicant further states that it will provide compensatory measures during testing performed on IROFS that are not redundant to ensure that the IROFS function until they are returned to service.

The applicant states that it will determine initial PM frequencies and procedures through the use of applicable industry experience, vendor-recommended intervals, and data derived from the reference facilities. Should it choose to deviate from those industry standards or vendor recommendations, the applicant commits to documenting the rationale for the deviation. In addition, the applicant states that feedback from PM and corrective maintenance, the results of incident investigations, and identified root causes, as appropriate, will be used to modify the frequency or scope of PM. The applicant states that, in determining the PM frequencies, it will consider the need to appropriately balance the objective of preventing failures through maintenance against the objective of minimizing the unavailability of IROFS because of PM.

After conducting PM on IROFS, and before returning IROFS to operational status, the applicant commits to performing necessary functional testing as described in Section 15.3.1.4 of the application, "Functional Tests," to ensure IROFS will perform their intended safety function.

The applicant commits to maintaining records pertaining to PM in accordance with the records management system. As stated by the applicant, it will evaluate the results of PM activities related to IROFS through the CM system by safety disciplines to determine any impact on the ISA and the need for updates.

15.3.3.1.3 Corrective Maintenance

The applicant describes the corrective maintenance program as the repair or replacement of equipment that has unexpectedly degraded or failed. The applicant's corrective maintenance program provides a planned, systematic, controlled, and documented approach for repair and replacement activities associated with IROFS.

After conducting corrective maintenance on IROFS, and before returning them to operational status, the applicant commits to performing necessary functional testing, as described in Section 15.3.1.4 of the application, to ensure IROFS will perform their intended safety function.

As stated by the applicant, it will evaluate the results of corrective maintenance activities related to IROFS through the CM system by safety disciplines to determine any impact on the ISA and the need for updates.

15.3.3.1.4 Functional Tests

The applicant states that it will implement a test control program incorporating plant procedures for test control and will provide for applicable compensatory measures during testing, in accordance with the limiting conditions for operations.

The applicant divided the operational testing program structure into two major testing programs—the preoperational testing program (defined below) and the operational testing program (defined below), each of which contains two testing categories. The preoperational testing program contains functional and initial startup testing, while the operational testing program includes periodic and special testing.

The applicant defines the objectives of the preoperational and operational testing program as ensuring that IROFS: (1) have been adequately designed and constructed, (2) meet licensing requirements, (3) do not adversely affect the health and safety of workers or the public, and (4) can be operated in a dependable manner so as to perform their intended function. In addition, the applicant states that the programs will ensure that operating, emergency, and surveillance procedures are correct.

The applicant states that the facility operating, emergency, and surveillance procedures will be progressively use-tested throughout the testing program and will also be used in the development of preoperational and startup testing procedures, to the extent practicable. In addition, the preoperational use of procedures will serve to familiarize personnel with plant operations during the testing phases and also will ensure the adequacy of the procedures under actual or simulated operating conditions.

Preoperational Testing Program

The applicant defines preoperational testing as testing performed following construction turnover to determine facility parameters and to verify the ability of IROFS to meet performance requirements. As stated in the application, the applicant will complete MFFF preoperational functional tests related to IROFS before the introduction of SNM to the facility to verify that those IROFS that are essential to the safe operation of the plant are capable of performing as intended. The applicant states that any tests or portions thereof that are not required to be completed before the introduction of the SNM will be specified in the test plans.

Functional Testing

As stated in the application, the applicant will perform functional testing, as appropriate, (1) following initial installation, (2) as part of periodic surveillance testing, and (3) after PM or corrective maintenance or calibration, to ensure that the item is capable of performing its safety function when required.

Initial Startup Testing

The applicant defines the period during which it will perform initial startup testing. The applicant states that initial startup testing will begin during the introduction of SNM to the facility and will end with the start of operations. The applicant states that the purpose of initial startup testing is to ensure the safe processing of SNM and the verification of parameters assumed in the ISA.

Operational Testing Program

The operational testing program, as described by the applicant, consists of periodic testing and special testing. Periodic testing will be conducted at the facility to monitor facility parameters and verify the continuing integrity and capability of IROFS. Special testing is defined by the applicant as any testing that does not fall under any of the other testing programs and is conducted on a nonrecurring basis.

The applicant states that the Maintenance Manager will have overall responsibility for the development and conduct of the operational testing program. The Operations Manager and Licensing Manager, in conjunction with the Maintenance Manager, will ensure that testing commitments and applicable regulatory requirements are met.

Periodic Testing

Periodic testing, as described by the applicant, will verify that the facility (1) complies with regulatory and licensing requirements, (2) does not endanger health and minimizes danger to life or property, and (3) is capable of operating so as to perform its intended function. The applicant states that the periodic testing program will apply during preoperational and operational stages of the facility, and the applicant commits to performing periodic testing and surveillances associated with the Quality Level 1 and Quality Level 2 SSCs in accordance with written procedures.

The applicant states that it will establish a periodic testing schedule to ensure that required testing is performed properly, in a timely manner, and consistent with the limiting conditions for operations, as identified in the Operating Limits Manual. The applicant further states that it will schedule periodic testing, such that the plant's safety will not be dependent on IROFS that are not tested. In cases where the testing is not performed within the specified timeframe, the applicant commits to providing appropriate compensatory measures.

Special Testing

The applicant describes special testing as testing that is not a facility preoperational test, periodic test, postmodification test, or postmaintenance test. The applicant states that it will conduct special testing to determine facility parameters or to verify the capability of IROFS to meet performance requirements. The applicant states that, at the discretion of the plant manager, any test may be conducted as a special test.

The applicant identifies some of the purposes of special testing as acquisition of particular data for special analysis; determination of information relating to facility incidents; verification that required corrective actions reasonably produce expected results and do not adversely affect the safety of operations; and confirmation that facility modifications reasonably produce expected results and do not adversely affect systems, equipment, or personnel by causing them to function outside established design conditions.

15.3.3.2 Measuring and Test Equipment

The Measuring and Test Equipment/Calibration Program, as described by the applicant, will be used to calibrate and maintain active engineered components used as IROFS. The applicant states that this program will identify the processes and plans to maintain and control calibration instruments and calibrations used at the MFFF. The applicant states that the program will also describe how instrument maintenance activities will take place.

15.3.3.3 Work Control

The applicant describes the maintenance work control process as a coordinated and structured process that integrates production activities and requirements. The work control process, as structured, seeks to minimize challenges to safety and production requirements, maximize work efficiency, and maintain consistency when making modifications. As stated by the applicant, it will include representation from other organizations, as needed to complete work activities. Some of the coordinated work support functions identified by the applicant include work requests, procedures, schedules, radiation work permits, and lockout or tagout requirements.

15.3.3.4 Relationship of Maintenance to Other Management Measures

The applicant states that it will perform maintenance activities in accordance with the QA program, as described in the MPQAP. The applicant also states that approved and controlled documents needed to support maintenance activities will be obtained through the CM program. Furthermore, the applicant states that the training and qualification program will ensure that maintenance personnel are trained to perform their tasks.

The applicant commits to performing audits and assessments of the maintenance program to ensure that the program implementation is effective. The applicant states that it will establish procedures to support the maintenance activities and that records management will provide the framework for review, maintenance, approval, handling, identification, retention, and retrieval of QA records related to maintenance activities. As stated by the applicant, incident investigations will identify the root cause(s) of any failures of the maintenance program.

15.3.4 Training and Qualification

The applicant's QA plan provides training and qualification requirements applicable during the operations phase of the facility, including preoperational functional testing and startup testing. The applicant states that the training program requirements apply to plant personnel who perform activities related to IROFS to ensure competent and safe job performance. The applicant commits to establishing requirements for the training of personnel performing QA Level 1 and Level 2 work activities; personnel performing nondestructive examination, inspections, and tests; and QA auditors.

The applicant states that the principal objective of the training program system is to ensure job proficiency of all facility personnel through effective training and qualification. The applicant commits to providing employees with (1) training to establish the knowledge foundation, (2) on-the-job training (OJT) to develop work performance skills, and (3) continuing training, as required, to maintain proficiency in these knowledge and skill components, and to provide further employee development.

The applicant identifies the requirements for personnel qualification and states that qualification will be indicated by the successful completion of prescribed training, a demonstration of the ability to perform assigned tasks, and the maintenance of requirements established by regulation. The applicant states that training will be designed, developed, and implemented according to a systematic approach that includes a variety of methods to accomplish the analysis, design, development, implementation, and evaluation of training.

15.3.4.1 Organization and Management of Training

The applicant states that line management is responsible and accountable for the development and effective conduct of training. The position description for line managers includes their training responsibilities; they are given the authority to manage, supervise, and implement training for their personnel and are supported by the training organization. The job function, responsibility, authority, and accountability of personnel involved in managing, supervising, and implementing training is clearly defined. The applicant identifies the accountability of line managers on the organizational chart included in the license application. The training manager is responsible for the facility training programs. The applicant will use performance-based training to analyze, design, develop, conduct, and evaluate training.

The applicant will develop and implement administrative procedures to establish requirements for the training of personnel performing activities related to IROFS. The procedures will also provide reasonable assurance that all phases of training are conducted reliably and consistently. The applicant will grant exceptions from training requirements when justified, properly documented, and approved by appropriate management. The applicant will incorporate the results of human factor engineering analysis into the training process and will incorporate the human factors task analysis of the IROFS identified in the ISA into plant procedures.

The applicant will use lesson plans or other approved process-controlling documents, as required, for training to ensure consistent presentation of the subject matter and will include updates to affected lesson plans in the change control process of the CM system when making design changes or plant modifications.

The applicant will maintain accurate and retrievable training records to support management information needs associated with personnel training, job performance, and qualifications. It will maintain individual records on each employee's qualifications, experience, and training. Specifically, training files will include records of general employee training, technical training, and employee development training conducted at the facility. The training manager is responsible for training records, which are retained in accordance with records management procedures. The applicant will use a learning management system to maintain training and qualification records. As stated in the application, all data entries will be peer reviewed within the training organization to ensure accuracy of the data, and data will be backed up nightly by the MOX information technology organization, with backup copies of the tapes stored remotely.

15.3.4.2 Analysis and Identification of Functional Areas Requiring Training or Qualification

The applicant will perform a needs and job analysis and will identify tasks to ensure that it provides appropriate training to personnel engaged in managing, supervising, performing, and verifying activities related to IROFS. The applicant states that it will identify job hazards as precautions and limitations in the procedure related to that task and will include them in the task's needs and job analysis.

The applicant will consult relevant subject matter experts, as necessary, to identify tasks for which training is appropriate. The applicant states that the training organization will identify, document, and address areas requiring training for competent and safe job performance and will consult with relevant subject matter experts, as necessary, to develop a list of tasks for which personnel training for specific jobs is appropriate. The applicant commits to comparing and reviewing the tasks selected for training with training materials as part of a training effectiveness evaluation. As stated in the application, the applicant will update the list of tasks selected for training as necessitated by changes in procedures, processes, plant systems, equipment, or job scope and will create a matrix of the task list and the supporting procedures and training materials.

15.3.4.3 Position Training Requirements

The applicant states that it will develop minimum training requirements for positions where activities are relied on for safety. The initial identification of job-specific training requirements will be based on experience from the MFFF reference facilities of MELOX and La Hague, and other U.S. fuel cycle facilities. The applicant will determine the level at which an employee will initially enter the training program by an evaluation of the employee's past experience, level of ability, and qualifications. The applicant will describe, in position descriptions, the entry-level criteria for positions where activities are relied on for safety and will grant exceptions from training requirements when justified and documented in accordance with the approved MFFF procedure. The applicant will conduct radiation safety training commensurate with each employee's duties.

The applicant also states that facility personnel may be trained through participation in general employee training or technical training. The applicant states that it will design the training program to prepare initial and replacement personnel for the safe, reliable, and efficient operation of the facility. The applicant commits to providing appropriate training for personnel of various abilities and experience backgrounds. As stated in the application, training requirements will be applicable, but not necessarily restricted, to personnel within the plant organization who have a direct relationship to the operation, maintenance, testing, or other technical aspects of the facility IROFS. The applicant will update training courses before use to reflect plant modifications and changes to procedures, when applicable.

15.3.4.3.1 General Employee Training

The applicant describes the general employee training that is required for access to the Savannah River Site and the MOX facility. The applicant states that general employee training will include QA, radiation protection, safety, emergency, and administrative procedures that are established by facility management and applicable regulations. The applicant states that persons that are under the supervision of facility management, including subcontractors, must participate in general employee training; however, certain temporary service and maintenance personnel will receive training to the extent necessary to ensure the safe execution of their duties.

15.3.4.3.2 Technical Training

The applicant states that it will design, develop, and implement technical training to assist employees in understanding applicable fundamentals, procedures, and practices related to IROFS. In addition, the applicant will use the technical training to develop the manipulative skills necessary to perform work related to IROFS. The applicant states that technical training will consist of initial training, OJT, and continuing training.

<u>Initial Training</u>

The applicant described the initial training as that used to provide employees with an understanding of the fundamentals, basic principles, and procedures involved in work related to

IROFS. The applicant states that initial training will consist of, but will not be limited to, live lectures, taped and filmed lectures, required reading, self-guided study, demonstrations, laboratories, workshops, and OJT.

The applicant states that certain new employees or employees transferred from other sections of the facility may be partially qualified by reason of previous training or experience, and thus it will determine the extent of training for these employees by applicable regulations, performance in review sessions, comprehensive examinations, or other techniques that can identify the employee's level of ability.

The applicant will develop initial job training and qualification programs for operations, maintenance, and technical services classifications and will group training for each program into logical blocks or modules. It will present training in a manner that ensures that specific behavioral objectives are accomplished, and it will evaluate trainee progress by written examinations and oral or practical tests.

On-the-Job Training

The applicant will conduct OJT to provide the required job-related skills and knowledge for positions. It will conduct OJT in an environment as close to the work environment as feasible and will supplement and complement classroom training. The OJT and qualifications program will comprise applicable tasks and related procedures for each technical area. The applicant states that it will derive technical areas for OJT based on the activities identified in the ISA Summary, job and task analyses, and associated procedures.

Continuing Training

The applicant defines continuing training as any training not provided as initial qualification or basic training that maintains and improves job-related knowledge and skills. The applicant will establish continuing training courses on a frequency needed to ensure that facility personnel remain proficient and will use periodic exercises, computer or classroom instruction, and any other type of training that is appropriate. The applicant states that, once it has established the objectives for continuing training, the methods for conducting it may vary; however, the method selected will provide clear evidence of objective accomplishment and consistency in delivery.

15.3.4.4 Basis and Objectives for Training

The applicant states that the objective of the training program shall be to ensure the safe and efficient operation of the facility and compliance with applicable established regulations and requirements. The applicant also states that its training requirements shall be applicable, but not necessarily restricted, to those personnel within the plant organization who have a direct relationship to the operation, maintenance, testing, or other technical aspects of the facility IROFS.

The applicant's learning objectives will identify the training content based on needs, and job analyses, and position-specific requirements. The applicant will use the task list from the needs and job analysis to develop action statements describing desired post-training performance. The training program's learning objectives will include: the knowledge, skills, and abilities to be demonstrated by the trainee; the conditions under which required actions will take place; and the standards of trainee performance expected by the applicant.

15.3.4.5 Organization of Instruction

The applicant will develop lesson plans from training learning objectives, which are based on job performance requirements. The lesson plans and other training guides are developed under guidance from the training organization. These plans are reviewed by the training organization and, generally, by the organization responsible for the subject matter before approval and or use. The applicant will use lesson plans as required for classroom training and OJT and will include standards for evaluating acceptable trainee performance in the plans.

15.3.4.6 Evaluation of Trainee Learning

The applicant will use observation, demonstration, or oral or written tests to evaluate a trainee's mastery of learning objectives. The evaluations will measure the trainee's skills and knowledge of job performance requirements.

15.3.4.7 Conduct of On-the-Job Training

The applicant will use OJT in combination with classroom training for selected activities. The applicant states that it will use lesson plans for classroom and OJT as required. The applicant also commits to using well-organized and current performance-based training materials and to including standards for evaluating acceptable trainee performance in training materials. As stated in the application, OJT will be conducted by personnel who are competent in the program standards of the job being performed and the methods of conducting the training. The applicant states that the completion of OJT will be demonstrated through actual task performance, where feasible and appropriate, or through performance of a simulation of the task, with the trainee explaining task actions based on the conditions that would be encountered during actual performance of the task and using references, tools, and equipment appropriate for the actual task, to the extent practical.

15.3.4.8 Systematic Evaluation of Training Effectiveness

The applicant will evaluate the training program periodically to measure its effectiveness in producing competent employees. The evaluation will consider feedback provided from trainees after completion of classroom training sessions and will evaluate program strengths and weaknesses, determine whether the program content matches current job needs, and identify any corrective actions needed to improve the program's effectiveness.

The applicant may address the following elements of training as they apply to the evaluation objectives of the training program or topical area being reviewed:

- management and administration of training and qualification programs
- development and qualification of the training staff
- position training requirements
- determination of training program content, including its facility change control interface with the CM system
- design and development of training programs, including lesson plans

- conduct of training
- trainee examinations and evaluations
- training program assessments and evaluations

The applicant will document the evaluation results and will highlight the program's noteworthy practices and weaknesses. The applicant will review identified deficiencies, recommend improvements, and make any changes to the affected procedures, practices, or training materials. In the event of plant modifications and procedure changes, the applicant will update affected training materials before their use.

Designated facility or contracted training personnel will periodically monitor training and qualification activities. The applicant states that the QA organization will audit the facility training and qualification program. Trainees and vendors can also provide input related to training program effectiveness. Methods used to obtain training program feedback include surveys, questionnaires, performance appraisals, and staff evaluations, as well as instruments to evaluate the overall effectiveness of the training program. The applicant states that it does not evaluate frequently conducted training classes every time they are held but evaluates them routinely at a frequency sufficient to determine program effectiveness.

15.3.4.9 Personnel Qualification

The applicant will determine qualification requirements for technical personnel in accordance with Chapter 15 of the MOX license application and will identify training and qualification requirements associated with quality-affecting activities in the MPQAP. These requirements will include QA training for project personnel and qualification of nondestructive examination personnel, inspection and test personnel, personnel performing special processes, and auditors. The applicant provides the requirements for key management positions in Chapter 4, "Organization and Administration," of the MOX license application.

15.3.4.10 Provisions for Continuing Assurance

The applicant states that it will evaluate personnel who perform activities relied on for safety at least biennially to verify that they continue to understand, recognize the importance of, and have the qualifications to perform such activities. The applicant will evaluate personnel using written or oral tests or on-the-job performance evaluations and will document the results of the evaluation. The applicant will provide retraining or other appropriate action when results of the evaluations dictate a need. The applicant will retrain personnel in the event of plant modifications, procedure changes, or QA program changes that result in new or revised information.

15.3.5 Plant Procedures

The applicant commits to conducting all activities involving SNM in accordance with approved procedures. The applicant describes the procedures it will use for control of overall facility operations, including the conduct of all operations involving controls identified in the ISA as IROFS and all management control systems supporting IROFS. The applicant states that MFFF management policies will require strict adherence to procedures when performing work. The applicant states that it will require personnel to notify their supervisor in the event that a procedure cannot be executed as written. The applicant also state that each MOX Services employee will have the authority to stop work that is being conducted within his or her scope of responsibility whenever it involves the health and safety of workers or the public, or protection of the environment, or when continued work will produce results that are not in compliance with the MOX Services QA program.

The applicant commits to implementing the requirements of the MPQAP for the development and control of plant procedures. The applicant states that activities associated with the development and control of plant procedures will be performed by personnel who have undergone training in accordance with the requirements of MPQAP Section 2, "Quality Assurance Program." The applicant states that all MFFF maintenance, testing, and operating procedures will meet the requirements of MPQAP Section 5, "Instructions, Procedures, and Drawing," and that plant procedures will be distributed and controlled in accordance with the requirements of MPQAP Section 6, "Document Control." The applicant also states that it will maintain documents that contain the results of procedure implementation (e.g., sign-offs, checklists, data sheets) in the records management system in accordance with the requirements of MPQAP Section 17, "Quality Assurance Records."

15.3.5.1 Types of Procedures

The applicant states that it will categorize MFFF procedures as either administrative procedures (which apply to functions or specific interfaces with other organizational functions) or operating procedures (which provide specific direction for functional task-based work). The applicant states that operating procedures can apply to all MOX Services organizations or only to a specific organization within MOX Services.

15.3.5.1.1 Administrative Procedures

The applicant states that administrative procedures will specify controls that apply to specific MFFF functions or to specific interfaces with other MFFF organizational functions. The applicant commits to implementing administrative procedures to address the administration and conduct of the following process activities: (1) training and qualification, (2) audits and assessments, (3) incident investigation, (4) records management, (5) CM, (6) human systems interface, (7) reporting, (8) QA, (9) equipment control (lockout or tagout), (10) shift turnover, (11) work control, (12) management control, (13) procedure management, (14) NCS, (15) fire protection, (16) radiation protection, (17) radioactive waste management, (18) maintenance, (19) environmental protection, (20) chemical process safety, (21) operations, (22) calibration control, (23) PM, (24) design control, and (25) test control.

15.3.5.1.2 Operating Procedures

The applicant states that operating procedures at MFFF will provide specific direction for functional task-based work within an organizational function and will include production, maintenance, and emergency procedures that address startup, operation, shutdown, control of process operations, and recovery after a process upset. The applicant identifies the specific areas addressed by these procedures as follows: ventilation; criticality alarms; shift routines, shift turnover, and operating practices; decontamination operations; plant utilities (air, other gases, cooling water, firewater, steam); temporary changes in operating procedures; and abnormal operation or alarm responses, including loss of cooling water, loss of instrument air, loss of electrical power, loss of the criticality alarm system, loss of containment, fires, and chemical process releases. The applicant commits to using the results of the ISA to identify the need for, and to support the development of, specific ACs for IROFS contained in operating procedures.

The applicant states that its operating procedures will include operating limits and controls and specific IROFS ACs necessary to ensure nuclear criticality safety, chemical safety, fire protection, emergency planning, and environmental protection. The applicant commits to identifying safety checkpoints (e.g., hold points for radiological or criticality safety checks, QA verifications, independent operator verification) at appropriate steps in operating procedures, if needed, to ensure the proper accomplishment of work.

The applicant states that it will organize all of the documents that comprise operating procedures with a consistent structure. The applicant commits to applying a consistent structure to general rules for production, maintenance, operational safety, and security; abnormal operating procedures; emergency planning procedures; emergency operating procedures; and the environmental protection program. The applicant will also apply a consistent structure to unit operating instructions and maintenance instructions, which provide instructions for operating and maintaining process units, systems, and equipment.

The applicant describes three categories of operating procedures that it will maintain at the MFFF: production, maintenance, and emergency procedures. Production procedures will control the startup, operation, and shutdown functions at the facility, as well as provide instructions for dealing with abnormal conditions, responding to alarms, controlling process and laboratory operations, and recovering after a process upset. Maintenance procedures will control preventive and corrective maintenance, calibration, surveillance, functional testing, and work control activities. Emergency procedures will describe the applicant's response to a criticality event, a hazardous chemical release, or an emergency external to the MFFF that may affect the MFFF.

15.3.5.1.2.1 <u>Production Procedures</u>

The applicant states that production procedures will control MFFF process operations and will apply to utility, workstation, and control room operations. The applicant commits to including the following elements, as applicable, in all production procedures: (1) purpose of the activity, (2) regulations, policies and guidelines governing the procedure, (3) type of procedure, (4) steps for each operating process phase, (5) initial startup and periodic startup and shutdown, (6) normal operations, (7) offnormal operations, (8) temporary operations, (9) emergency shutdown, (10) emergency operations, (11) normal shutdown, (11) startup following an emergency or extended downtime, (12) hazards and safety considerations (13) operating limits, (14) precautions necessary to prevent exposure to hazardous chemicals or SNM, (15) measures to be taken if contact or exposure occurs, (16) safety controls and the functions associated with the process, and (17) specified time period or other limitations on the validity of the procedure.

15.3.5.1.2.2 <u>Maintenance Procedures</u>

The applicant states that MFFF maintenance procedures will include requirements for premaintenance activities, as necessary, and that these activities may include reviews of the work to be performed, work controls, and reviews of procedures. The applicant commits to requiring clearance from, or notification of, the operations organization as appropriate, when maintenance work and associated postmaintenance functional testing are complete. The applicant commits to monitoring and assessing maintenance activities in accordance with the MPQAP.

The applicant states that it will maintain facility SSCs in accordance with written procedures, documented instructions, checklists, or drawings appropriate to the circumstances. The applicant further states that maintenance activities will address repair, calibration, surveillance, and functional testing and will specifically include repairs and preventive repairs of IROFS, testing of criticality alarm units, calibration of IROFS, maintenance of high-efficiency particulate air filters, functional testing of IROFS, relief valve replacement and testing, surveillance and monitoring, pressure vessel testing, piping integrity testing, and containment device testing. The applicant identifies the organizational responsibilities for the preparation of maintenance procedures; specifically, the applicant states that the MFFF maintenance department, which is led by the Maintenance Manager, will be responsible for the preparation and implementation of maintenance procedures.

The applicant will use approved, written procedures for periodic tests performed to determine various facility parameters and verify the continuing capability of IROFS to meet performance requirements. The applicant states that periodic test procedures will be sufficiently detailed so as to enable qualified personnel to perform the required functions without direct supervision. The applicant commits to implementing compensatory measures when testing is performed on IROFS that are not redundant, to ensure that they are able to perform their safety functions until they are returned to service.

15.3.5.1.2.3 Emergency Procedures

The applicant commits to implementing emergency procedures to address the preplanned actions of operators and other plant personnel in response to an incident, criticality event, hazardous chemical release, or external emergency that may affect the MFFF. The applicant also commits to reviewing applicable procedures after unusual incidents (e.g., accidents, unexpected transients, significant operator errors, equipment malfunctions, system modifications) and to making revisions, as needed.

15.3.5.2 Preparation of Procedures

The applicant states that its facility procedures will be consistent in format, well organized, clear, concise, and comprehensive. The applicant also states that its procedures may include (approved) checklists or data sheets as documented records of completion. The applicant describes its approval process for plant procedures; other members of the MFFF staff and vendors, as appropriate, will review procedure drafts for inclusion and correctness of technical information, including formulas, set points, and acceptance criteria. The applicant will require a peer review of all procedures that are written for the operation of equipment related to IROFS. The Functional Area Manager will be responsible for (1) determining whether procedures require any additional, cross-disciplinary review, and (2) approving procedures. The applicant commits to clearly identifying safety limits associated with IROFS in the applicable procedures.

15.3.5.2.1 Identification and Preparation

The applicant commits to using the results of the ISA and other processes to identify specific operating and administrative procedures that are developed for MFFF. The applicant also states that plant procedures will be prepared by qualified individuals who are assigned by the organization's management to be responsible and accountable for the operation associated with the procedure.

The applicant commits to including consideration of ISA results or changes in ISA results in the process of identifying procedures needed for facility operation. The applicant further commits to incorporating a methodology for identifying, developing, approving, implementing, and controlling operating procedures. The applicant states that the methodology it is committed to implementing will ensure that, as a minimum, (1) the procedure will specify operating and safety limits related to IROFS, (2) procedures will include required actions for normal and abnormal conditions of operation. (3) safety checkpoints will be identified at appropriate steps in the procedure, if necessary, (4) procedures will be validated through field tests, (5) Functional Area Managers who are responsible and accountable for the operation will approved procedures, (6) a mechanism will be specified for revising and reissuing procedures in a controlled manner, (7) the QA elements and CM program at the facility will provide reasonable assurance that current procedures are available and in use at work locations, and (8) the facility training program will train the required persons in the use of the latest procedures available.

15.3.5.2.2 Review and Approval

The applicant states that managers who are responsible and accountable for an operation will review and approve the associated operating and administrative procedures. The applicant further states that the functional management may specify a review to be performed by another functional group. The applicant commits to verifying and validating production and maintenance procedures before initial use or after major revisions.

15.3.5.2.3 Revisions

The applicant commits to preparing and approving procedure revisions, including temporary changes, in the same manner as the original. The applicant also commits to defining the procedure change process in an MFFF procedure.

15.3.5.3 Use of Procedures

The applicant states that it will require compliance with operating and maintenance procedures and will train operators and technicians to report inadequate procedures or the inability to follow procedures. The applicant states that procedures will either be available at work stations or readily accessible where needed to perform work.

15.3.5.4 Control of Procedures

The applicant describes its process for document control of plant procedures and states that, after approval, plant procedures will be processed for entry into the EDMS and issued for use. The applicant commits to implementing the MFFF training program, which is addressed in Section 15.4 of the MPQAP, to ensure that necessary personnel are trained in the use of approved procedures before implementation.

The applicant commits to applying the same change control measures to operating and administrative procedures that are applied to other items in the document management system. The applicant states that document management procedures will ensure that changes to the facility, including procedures, are entered into the EDMS. The applicant also states that document management procedures will address control and distribution of changes, including changes implemented for emergency conditions, temporary procedure changes, and temporary modifications. The applicant states that the MPQAP will provide requirements for QA procedures, which will detail the controls for design input, design output, processes, verification, interfaces, changes, approval, and records.

The applicant commits to reviewing radiation protection, respiratory protection, operating, maintenance, and administrative procedures every five years to ensure technical adequacy and to verify their continued applicability and accuracy. The applicant also commits to reviewing respiratory protection procedures, as appropriate, whenever the MFFF undergoes a modification, change in process, or replacement of equipment. The applicant commits to reviewing emergency procedures annually for the first two years of MFFF operation and at least every two years thereafter. The applicant states that periodic reviews will be performed by qualified individuals who are assigned to be responsible and accountable for the associated operation by functional management. The applicant states that any reissue or approval of a procedure will meet the requirements for periodic review, and if a procedural inadequacy is identified as a root cause from an incident investigation, it will review and modify the applicable procedures as necessary.

15.3.6 Audits and Assessments

As described in Section 16 of the application, the applicant will maintain the audits and assessment program in accordance with Section 18, “Audits” of the MPQAP. The applicant states that it will review any changes to the MPQAP to ensure that the audit and assessment program will be current and will reflect the program description. As described by the applicant, audits will focus on verifying compliance with regulatory and procedural requirements, licensing commitments, and selected operating limits, and assessments will focus on evaluating the effectiveness of activities and ensuring that IROFS and items that affect the function of IROFS are available and reliable to perform their intended safety functions. In addition, the applicant states that it will perform audits and assessments to ensure that facility activities are conducted in accordance with the written procedures and that the processes reviewed are effective. As a minimum, the applicant commits to performing audits and assessments for activities related to radiation protection; criticality safety control; hazardous chemical safety; industrial safety, including fire protection; results of the ISA; environmental protection; and other areas identified through trends.

As stated in the application, the applicant will perform audits in accordance with a written plan that identifies the audits to be performed and their schedules. The applicant confirms that qualified staff personnel who are not directly responsible for production activities will perform audits and assessments on an annual basis. The applicant states that audit team members will not have direct responsibility for the function and area being audited, will have technical expertise or experience in the area being audited, and will be trained in audit techniques.

The applicant commits to performing technical and programmatic audits and assessments internally and externally to provide a comprehensive independent verification and evaluation of procedures and activities for IROFS. As described by the applicant, the QA Department will be responsible for audits related to Quality Level 1 work activities and items required to satisfy regulatory requirements for which Quality Level 1 requirements are applied. The applicant states that it will provide audits results to the Plant Manager and the managers responsible for the activities audited.

The applicant states that any deficiencies identified during audits or assessments that require corrective action will be forwarded to the responsible manager in accordance with the CAP procedure. The manager will then be responsible for promptly responding to any deficiencies noted in the audits. The applicant states that it will enter deficiencies into the CAP, track them to completion, and re-examine them during future audits to ensure that associated corrective actions have been completed. As described by the applicant, the audit and assessment program will provide for on-the-spot corrective actions with appropriate documentation, in accordance with the CAP procedure, and will include the evaluation of corrective actions to determine their effectiveness.

In the application, the applicant describes two assessments categories: (1) management assessments conducted by the line organizations responsible for the work activity and (2) independent assessments conducted by individuals not involved in the area being assessed.

The applicant states that it will maintain records of the instructions and procedures, persons conducting the audits or assessments, identified violations of license conditions, and any corrective actions taken.

15.3.6.1 Areas to be Audited or Assessed

The applicant identifies a list of areas that it will audit or assess at the MFFF, including radiation safety; nuclear criticality safety; chemical safety; industrial safety, including fire protection; environmental protection; emergency management; QA; CM; maintenance; training and qualifications; procedures; CAP and incident investigations; records management; and other ISA safety areas. The applicant commits to performing assessments of nuclear criticality safety in accordance with ANSI/ANS-8.19 to ensure that operations conform to criticality requirements.

15.3.6.2 Scheduling of Audits and Assessments

The applicant states that it will establish a schedule identifying audits and assessments to be performed and the responsible organization assigned to conduct the activity. As described by the applicant, the frequency of audits and assessments will be reviewed periodically and revised as necessary to ensure coverage commensurate with current and planned activities. The applicant states that major activities will be audited or assessed on an annual basis.

The applicant also states that it will conduct and document nuclear criticality safety audits such that all aspects of the Nuclear Criticality Safety Program will be audited every two years and assessed annually.

15.3.6.3 Procedures for Audits and Assessments

The applicant commits to conducting internal and external audits and assessments in accordance with approved procedures. Among the audit and assessment activities that will be controlled by procedures are scheduling, planning, certifying personnel, developing audit plans, and reporting, tracking, and closure of findings. The applicant states that it will emphasize, through the applicable procedures, the importance of reporting and correcting findings to prevent recurrence.

As described in the application, the applicant will conduct audits and assessments by using checklists (where applicable); interviewing personnel; performing plant area walkdowns, including accessible out-of-the-way and limited-access areas; reviewing plans and procedures;

observing work in progress; and reviewing completed QA documentation. The applicant commits to tracking the results of audits and assessments in the CAP. The applicant states that it will evaluate audit and assessment results for trends and needed improvements, which will be reported to the appropriate levels of management when identified. As described in the application, deficiencies will require corrective action in accordance with the applicable CAP procedure, and the QA organization will be responsible for performing followup reviews on significant deficiencies reported as a result of the trend analysis and for verifying completion of corrective actions.

The applicant states that the audit or assessment team leader will develop a report documenting the findings, observations, and recommendations for program improvement and provide it to management. As described by the applicant, the report will include documented verification of performance against established performance criteria for IROFS and will be developed, reviewed, approved, and issued in accordance with applicable procedures. The applicant states that audit reports will contain an effectiveness evaluation and statement for each of the applicable QA program elements that were reviewed during the audit. The applicant commits to closing the audit or assessment with the proper documentation, in accordance with the applicable audit and assessment procedure.

As described in the application, the QA organization will conduct followup audits or assessments to verify that corrective actions were taken in a timely manner and to assess their effectiveness.

15.3.6.4 Qualifications and Responsibilities for Audits and Assessments

The applicant states that the QA Manager will initiate audits and will determine the scope of each audit in coordination with the lead auditor. The QA Manager will also be responsible for the initiation of any special audits or the expansion of the scope of audits, when necessary. As described by the applicant, the lead auditor will direct the audit team in conducting the audit as well as in developing the applicable checklists, instructions, or plans for performing the audit. The applicant states that audit teams will consist of one or more auditors, and, should the team deem it necessary, it may expand the scope of the audit during the audit activity. The applicant commits to ensuring that audits will be performed in accordance with applicable checklists.

The applicant states that auditors and lead auditors will hold the appropriate certifications, as required by the MPQAP. As stated in the application, to be certified under the MOX Services QA program, MFFF auditors will be required to complete training in areas such as the MFFF QA program, audit fundamentals, objectives and techniques of performing audits, and OJT.

As described in the license application, to form the basis of each auditor's certification, the QA Manager will evaluate the auditors' and lead auditors' education, experience, professional qualifications, leadership, sound judgment, maturity, analytical ability, tenacity, past performance, and success in completion of QA training courses. The applicant states that lead auditors must meet additional requirements for qualification, such as a minimum of five QA audits or audit equivalents within a period of time not to exceed three years before the date of certification, at least one of which must be a nuclear-related QA audit or audit equivalent within the year before certification.

The applicant states that it will require personnel performing assessments to complete QA orientation training, as well as training on the assessment process. The applicant states that personnel performing assessments will not report to the production organization and will have no direct responsibility for the function or area being assessed, enabling them to maintain independence and objectivity.

15.3.7 Incident Investigations

Section 15.7 of the application describes the two MFFF programs for investigating discrepancies during operations: the corrective action process and incident investigations.

15.3.7.1 Corrective Action Process

The applicant states that it will use the corrective action process, which is described in Section 16 of the MPQAP, "Corrective Action," to identify, investigate, report, track, correct, and prevent recurrence of conditions that are adverse to quality or that may affect radiation protection, safety, quality, regulatory compliance, reliability, human performance, or project performance. The applicant states that MOX Services employees have the authority and responsibility to initiate the corrective action process if they discover deficiencies. The applicant further states that it will analyze reports of conditions adverse to quality to identify trends in quality performance, and these will be reported to senior management in accordance with corrective action process procedures.

15.3.7.2 Incident Investigation

The applicant commits to using the incident investigation program for investigating abnormal events other than those that involve a condition adverse to quality. As described in the application, the process that will be used for incident investigations may be similar to that of the CAP; the applicant states that it will consider events in terms of their regulatory reporting requirements and the level of investigation required. The applicant commits to providing guidance in written procedures for classifying occurrences (including examples of the threshold for offnormal events), incident identification, investigation, root-cause analysis, environmental protection analysis, recording, reporting, and followup.

The applicant states that the depth of incident investigations will depend upon the severity of the classified incident in terms of the levels of SNM released or the degree of potential for exposure of workers, the public, or the environment. The applicant commits to addressing radiological, criticality, hazardous chemical, and other ISA-related safety requirements in incident investigations and states that anyone in the MFFF organization may identify the need for an incident investigation, which will be performed by one or more individuals assigned by the manager of production. As described in the application, MOX Services will maintain a record of corrective actions, including lessons learned and worker training, to be implemented as a result of investigations of offnormal occurrences and will track the corrective actions to completion.

The applicant states that it will establish an incident investigation program to investigate abnormal events that may occur during operation of the facility to determine their specific or generic root cause(s) and generic implications, to recommend corrective actions, and to make reports to the NRC as required by 10 CFR § 70.50, "Reporting Requirements," and 10 CFR § 70.74, "Additional Reporting Requirements." As described by the applicant, the investigation teams will include at least one process expert and one team member trained in root-cause analysis. The applicant commits to monitoring and documenting corrective actions taken through to completion.

The applicant states that it will maintain auditable records and documentation related to abnormal events, investigations, and root-cause analyses so that it may apply the lessons learned to future operations of the facility. The applicant will compare details of the event sequence with accident sequences already considered in the ISA and, as appropriate, will modify the ISA and ISA Summary to include an evaluation of the risk associated with accidents of the type actually experienced.

The investigation process, as described by the applicant, will include a prompt risk-based evaluation that, depending on the complexity and severity of the event, may be conducted by one individual. The applicant states that incident investigator(s) will be (1) qualified individuals appointed from internal or external staff, (2) independent from the line function(s) involved with the incident under investigation, and (3) assured of no retaliation for participating in investigations. The applicant commits to initiating investigations within forty-eight hours of the abnormal event, or sooner, depending on the safety significance of the event. As described in the application, the applicant will review the record of IROFS failures required to be maintained by 10 CFR § 70.62(a)(3), "Safety program and integrated safety analysis," as part of the investigation. and, following completion of the investigation, the applicant will record revisions necessitated by post-failure investigation conclusions.

The applicant states that it will develop CAP procedures for conducting incident investigations that will contain elements such as the following:

- a documented plan for investigating an abnormal event;
- a description of the functions, qualifications, and responsibilities of the manager who will lead the investigative team and those of the other team members, the scope of the team's authority and responsibilities, and the assurance of management cooperation;
- assurance of the team's authority to obtain the information considered necessary and its independence from responsibility for or to the functional area involved in the incident under investigation;
- requirements for retention of documentation related to abnormal events for two years or for the life of the activity, whichever is greater;
- guidance for personnel conducting the investigation on how to apply a reasonable, systematic, structured approach to determine the specific or generic root cause(s) and generic implications of the problem;
- requirements to make original investigation reports available to the NRC on request; and
- a system for monitoring the completion of appropriate corrective actions and for ensuring that those actions are completed in a timely manner.

15.3.8 Records Management

The applicant describes the records management requirements in Section 15.8 of the application and states that Section 17 of the MPQAP contains additional details related to the records management program. The applicant identifies QA records as documents that include the results of tests and inspections required by applicable codes and standards; construction, procurement, and receiving records; personnel certification records; design calculations; purchase orders; specifications; procedures; corrective action records; source surveillance and audit reports; and any other QA documentation required by specifications or procedures. As described in the application, the applicant will use a controlled and systematic approach to records management to provide identifiable and retrievable documentation during design, construction, and operation of the MFFF. The applicant commits to controlling QA records in accordance with approved procedures and will not consider these records valid until they are authenticated by authorized personnel. The applicant further commits to developing and implementing records management procedures that establish the requirements and responsibilities for record selection, verification, protection, transmittal, distribution, retention, maintenance, and disposition. In addition, the applicant states that it will establish procedures to promptly detect and correct deficiencies in the records management system or in the system's implementation.

The applicant states that the MPQAP requires procedures for the review, approval, identification, handling, retention, retrieval, and maintenance of QA records. The applicant commits to maintaining records at locations where they can be reviewed and audited to ensure that the required quality of the records is maintained. The applicant further states that applicable design specifications, procurement documents, and other documents will specify applicable QA record requirements.

The applicant states that it will manage classified records in accordance with approved project procedures that will identify the required physical protection and access control measures. As stated in the application, the applicant will establish a satellite records retention facility in accordance with the records management procedure.

The applicant commits to establishing procedures to control and manage computer codes and electronic data used for IROFS over the life cycle of the MFFF. The applicant states that the MFFF Records Center will maintain control over access and use of records, either originals or reproductions, that are entered into the EDMS and will ensure that documents in the EDMS are legible and can be identified with the subject to which they pertain. The applicant states that documents will only be considered valid if stamped, initialed, signed, or otherwise authenticated by authorized personnel.

The applicant will establish requirements to preclude deterioration of records in the EDMS, specifically, requirements pertaining to the records storage arrangement, to prevent damage from moisture, temperature, and pressure. For hardcopy records, the applicant will require records to be: (1) firmly attached in binders, placed in folders, or placed in envelopes for storage in steel file cabinets, or (2) stored on shelving in containers appropriate for the record medium. The applicant further states that the storage arrangement will provide adequate protection of special processed records (e.g., radiographs, photographs, negatives, microform, and magnetic media) to prevent damage from moisture, temperature, excessive light, electromagnetic fields, or stacking, consistent with the type of record being stored.

The applicant identifies measures to ensure the accurate retrieval of information without undue delay. The applicant states that it will store and preserve the records in the Records Center in accordance with an approved QA procedure that contains the following:

- a description of the storage facility;

- a description of the filing system to be used;
- a method for verifying that the records received are in agreement with the transmittal document;
- a method for verifying that the records are those designated and the records are legible and complete;
- a description of rules governing control of the records, including access, retrieval, and removal;
- a method for maintaining control of, and accountability for, records removed from the storage facility;
- a method for filing supplemental information and disposition of superseded records;
- a method for precluding entry of unauthorized personnel into the storage area, to guard against larceny and vandalism; and
- a method for providing for replacement, restoration, or substitution of lost or damaged records.

The applicant lists examples of the records that it will retain, including operating logs, procedures, non-conforming item reports, drawings and specifications, procurement documents, audit reports, and dosimetry records. As stated in the application, retention times will be specified in records management procedures and will ensure that records are retained in accordance with regulatory requirements. The applicant commits to storing one-of-a-kind records in two hour fire-rated cabinets, to ensure records are adequately protected from damage.

15.4 Evaluation Findings

15.4.1 Quality Assurance

The staff reviewed the QA program for a license for the MFFF to possess and use SNM according to Chapter 15.1 of NUREG-1718. Based on its review of the MOX Services QA Plan, the NRC staff concluded that the applicant has adequately described its QA program, and the applicant's QA program meets the regulatory requirements of 10 CFR Part 70, as applied to SSCs, and will provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents. The staff's review and approval of the MFFF QA program is documented in the safety evaluation report dated October 1, 2001, as updated by the letter dated October 19, 2009.

Configuration Management

The staff reviewed the CM system for MFFF according to Section 15.2 of NUREG-1718. Based on its review of the material submitted in the license application, the NRC staff concluded that the applicant suitably and acceptably described its commitment to a proposed CM system, including the method for managing changes in procedures, facilities, activities, and equipment for IROFS identified in the safety assessment for the design bases.

The applicant described management-level policies and procedures, including an analysis and independent safety review of proposed activities involving IROFS that will ensure that the relationship among design requirements, construction, and facility documentation is maintained as part of a new design or change to an existing design. The MFFF ACs, as described in the application, will ensure that the organizational structure, procedures, and responsibilities necessary to implement CM are in place; that the design requirements and bases are documented and supported by analyses and the documentation is maintained current; that documents, including drawings, are appropriately stored and accessible; that drawings and related documents adequately describe IROFS; that procedures adequately describe how the applicant will achieve and maintain strict consistency among the design requirements, facility construction, and facility documentation; and that methods are in place for suitable analysis, review, approval, and implementation of identified changes to IROFS.

The staff concludes that the applicant’s CM function meets the requirements of 10 CFR Part 70 and provides reasonable assurance that the environment and the health and safety of the public are protected.

15.4.2 Maintenance

The staff reviewed the maintenance program for MFFF according to Section 15.3 of NUREG-1718. Based on the review of the license application, the staff concluded that the applicant committed to performing maintenance of IROFS, with the exception of personnel activities (safety controls). The staff reviewed and evaluated the maintenance commitments, which contain measures to ensure availability and reliability of IROFS through surveillance and monitoring, corrective maintenance, PM, and functional testing activities. The functional testing activities comprise a detailed test control program that covers preoperational and operational activities, including initial startup testing and periodic testing. The applicant’s maintenance function is proactive, using both surveillance and monitoring and maintenance records to analyze equipment performance and identify the root causes of repetitive failures.

In addition, the surveillance and monitoring activities described in this section of the application provide assurance of the validity of the ISA by examination, calibration, and testing of equipment that monitors process safety parameters and acts to prevent or mitigate accident consequences.

The maintenance function (1) is based on approved procedures, (2) employs work control methods that properly consider personnel safety, awareness of facility operating groups, QA, and the rules of CM, (3) links IROFS requiring maintenance to the ISA, (4) justifies the PM intervals in terms of equipment reliability goals, and (5) creates documentation that includes detailed records of all surveillances, inspections, equipment failures, repairs, and replacements.

The staff concludes that the applicant's maintenance program meets the requirements of 10 CFR Part 70 and provides reasonable assurance that the environment and the health and safety of the public are protected.

15.4.3 Training and Qualifications

The staff reviewed the application for the MFFF according to Section 15.4 of NUREG-1718. The applicant described the structure of the MFFF training and qualification program and committed to providing plant personnel with a combination of general and technical training that includes initial training, OJT, and continuing education, as required, to establish and maintain the proficiency of personnel in their work duties. The applicant commits to performing a needs and job analysis to identify tasks that require training, to ensure that appropriate training is provided to personnel managing, supervising, performing, and verifying activities related to IROFS. The applicant further commits to systematically evaluating the effectiveness of the training program at periodic intervals. Based on its review of the application, the NRC staff concludes that the applicant adequately described its training and qualification of plant personnel and that the applicant's training and qualification of plant personnel will, based on commitments, meet the requirements of 10 CFR Part 70 and provide reasonable assurance of the protection of public health and safety and of the environment.

15.4.4 Procedures

The staff reviewed procedural controls described in the license application for the MFFF according to Section 15.5 of NUREG-1718. The applicant described the administrative and operating procedures for control of overall facility operations, including the conduct of all operations involving controls identified in the ISA as IROFS and all management control systems supporting IROFS. The applicant committed to conducting all activities involving SNM in accordance with approved procedures and to reviewing all radiation protection, respiratory protection, operating, maintenance, and administrative procedures every five years to ensure technical adequacy and to verify the continued applicability and accuracy of the procedures. The applicant has suitably described the processes for development, review, approval, control, and implementation of procedures. As described in the application, MOX Services has established or made commitments to establish sufficient procedural guidance to ensure the proper control and protection of IROFS, as well as systems important to the health of workers and the public and the protection of the environment during testing, startup, and operation of the facility. Based on its review of the application, the NRC staff concludes that the applicant has adequately described its controls for the establishment, maintenance, use, and revision of MFFF procedures, and those controls meet the requirements of 10 CFR Part 70 and provide reasonable assurance of the protection of public health and safety and of the environment.

15.4.5 Audits and Assessments

The staff reviewed the MFFF audit and assessment program description, as described in the license application, according to Section 15.6 of NUREG-1718. The staff reviewed the applicant's description of its policy directives, plans, and procedural requirements with respect to (1) the general structure of the audits and assessments program, (2) the activities to be audited or assessed, (3) the scheduling of audits and assessments, (4) the procedures for audits and assessments, and (5) the qualifications and responsibilities for audits and assessments.

Based on its review of the application, the NRC staff concludes that the applicant has adequately described its system of audits and assessments, and this system meets the requirements of 10 CFR Part 70 and provides reasonable assurance of the protection of public health and safety and of the environment.

15.4.6 Incident Investigations

The staff reviewed the license application for MFFF as it pertains to incident investigations according to Section 15.7 of NUREG-1718. As described, the MFFF incident investigation program specifies the process for investigating abnormal events, the qualification requirements for investigation personnel, the size and composition of investigation teams, corrective action commitments, and records requirements for investigation-related documents. The applicant commits to performing incident investigations in accordance with approved procedures.

Based on its review, the NRC staff concluded that the applicant has established an organization for (1) investigating incidents that occur during operation of the facility, (2) determining the root cause(s) and any generic implications of each incident, and (3) taking corrective actions for ensuring the safety of the MFFF and its operations. Furthermore, the applicant has committed to reviewing the results of the investigation against the ISA, to monitoring and documenting corrective actions through to completion, to maintaining investigation-related documentation, and to applying lessons learned to future operations of the facility. Based on its review of the application, the NRC staff concludes that the applicant has adequately described its program for incident investigations, and the applicant's controls for investigating incidents meet the requirements of 10 CFR Part 70 and provide reasonable assurance of the protection of public health and safety and of the environment.

15.4.7 Records Management

The staff reviewed the MFFF records management controls, as described in the license application, according to Section 15.8 of NUREG-1718. The staff reviewed the applicant's records management requirements for the control and handling of the records and concluded that there is reasonable assurance that the system will (1) be effective in collecting, verifying, protecting, and storing information about the health and safety aspects of the facility and its operations and will be able to retrieve the information in readable form for the designated lifetimes of the records, (2) provide record storage facilities capable of protecting and preserving records that are stored there during the mandated periods, including protecting the stored records against loss, theft, tampering, or damage during and after emergencies, and (3) ensure that any deficiencies in the records management system or its implementation will be detected and corrected in a timely manner. The staff concludes that the applicant's facility records management system meets the requirements of 10 CFR Part 70 and is acceptable.

REFERENCES

(MOX, 2010a) Shaw AREVA MOX Services, "MFFF—License Application," Aiken, SC, March 2010.

(MOX, 2010b) Shaw AREVA MOX Services, "MFFF—Integrated Safety Analysis Summary," Aiken, SC, March 2010.

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(NRC, 2001) U.S. Nuclear Regulatory Commission, Letter from Andrew Persinko to Peter Hastings, "Duke Cogema Stone & Webster Quality Assurance Program for Construction of the MOX Fuel Fabrication Facility," Washington, D.C., October 1, 2001.

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10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities”

10 CFR Part 21, “Reporting of Defects and Noncompliance”

10 CFR 70, “Domestic Licensing of Special Nuclear Material”

16.0 AUTHORIZATIONS AND EXEMPTIONS

16.1 Purpose of Review

This chapter of the safety evaluation report (SER) contains the U.S. Nuclear Regulatory Commission (NRC) staff's review of Chapter 16 of the "Mixed Oxide Fuel Facility (MFFF) License Application" (MOX, 2010). The staff performed the review using general guidance from NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility" (NRC, 2000), and the regulatory requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection against Radiation"; 10 CFR Part 40, "Domestic Licensing of Source Material"; and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material." The objective of this review is to verify whether the applicant (Shaw AREVA MOX Services, also referred to as MOX Services) has included exemptions and authorizations that are acceptable as determined by the staff's review.

16.2 Areas of Review

16.2.1 Exemptions

16.2.1.1 Decommissioning

In Section 16.1.1 of the license application (LA), the applicant stated that the U.S. Department of Energy (DOE) will assume responsibility for decommissioning the MFFF and that MOX Services has submitted a request for exemption (DCS, 2006a) from decommissioning requirements. The staff's review of the acceptability of this exemption follows.

As stated in Section 1.2.4.1 of the Construction Authorization Request (DCS, 2004), DOE intends to assume responsibility for decommissioning the MFFF.

The contract between DOE and MOX Services includes a requirement that, following completion of its mission for disposition of excess plutonium, the facility will be deactivated and returned to DOE. As discussed in Issue 8 of SECY 99-177, "Current Status of Legislative Issues Related to NRC Licensing a Mixed Oxide Fuel Fabrication Facility," dated July 8, 1999 (NRC, 1999), it was suggested that the LA state that the MFFF be returned to DOE at the conclusion of the contract between DOE and MOX Services for the operation of the MFFF following deactivation of the MFFF to the satisfaction of DOE.

Pursuant to 10 CFR § 70.17(a) and 10 CFR § 40.14(a), MOX Services requested an exemption from the requirements of 10 CFR § 70.38(d)–(k) and 10 CFR § 40.42, "Expiration and Termination of Licenses and Decommissioning of Sites and Separate Buildings or Outdoor Areas," relating to the responsibility for decommissioning. Based on the agreement for DOE to assume responsibility for decommissioning, the staff finds the requested exemption to be acceptable. The requested exemption is authorized by law and will not endanger life, property, or the common defense and security and is in the public interest. Since DOE will assume responsibility for decommissioning, the method of financial assurance for decommissioning is in accordance with 10 CFR 70.25(f)(5) and 10 CFR § 40.36(e)(5).

This exemption will be included in the license to possess and use radioactive material that may be granted to the applicant after completion of other regulatory requirements in 10 CFR Part 70.

16.2.1.2 Financial Protection

In Section 16.1.2 of the LA, the applicant addressed the issue of Price-Anderson liability coverage. DOE has agreed to indemnify MOX Services in accordance with Section 170(d) of the Atomic Energy Act of 1954, as amended, and DOE Acquisition Regulation (DEAR) 952.250-70, "Nuclear Hazards Indemnity Agreement" (48 CFR § 952.250-70). The applicant submitted a request (DCS, 2006b) for exemption from 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," including the requirement in 10 CFR § 140.13a, "Amount of Financial Protection Required for Plutonium Processing and Fuel Fabrication Plants." Chapter 2 of this SER describes the staff's review of the acceptability of this exemption.

16.2.1.3 Labeling

The applicant submitted a request (DCS, 2006c) for an exemption from the labeling requirements of containers with licensed material set forth in 10 CFR § 20.1904(a). Section 9.5 of this SER presents the staff's review of the acceptability of this exemption.

16.2.2 Authorizations

16.2.2.1 Prior Commitments

In Section 16.2.1 of the LA, the applicant provided the authorization that all commitments made to the NRC prior to the most recent NRC-approved revision of the LA shall no longer be binding upon MOX Services unless imposed by license condition. The applicant has included this authorization under the assumption that all commitments relating to the MFFF will be included in or referenced by the license for the facility. The staff believes that upon issuance of a license, all issues that require commitments will be resolved or will be conditions of the license. The staff therefore finds that this authorization is acceptable.

16.2.2.2 Frequencies

When measurement, surveillance, and/or other frequencies are specified in the LA or other license commitments, the applicant proposes a list of definitions of the time period for specific frequency designations. The staff reviewed the frequencies associated with each time period and finds that the frequencies are well defined and reasonable. The staff therefore finds the use of the defined frequencies provided in the LA and their use to be acceptable.

16.2.2.3 Changes to the License Application

In Section 16.2.3 of the LA, the applicant provided the conditions to make changes to the LA and the criteria for determining when prior approval from the NRC would be needed. MOX Services maintains the LA so that it is accurate and up to date by means of the MFFF configuration management processes, which include written procedures. MOX Services evaluates changes to the facility and its processes for impact on the LA and updates the LA as necessary to ensure its continued accuracy. Responsibility for maintaining and updating the LA belongs to the manager of the support services function, as described in Chapter 4 of the LA.

A change to the facility or its processes is evaluated before the change is implemented. The applicants' evaluation of the change determines, before the change is implemented, whether an application for an amendment to the LA must be submitted in accordance with 10 CFR 70.34, "Amendment of licenses." The LA describes the sites, structures, processes, systems, equipment, components, computer programs, and activities of personnel. MOX Services may make changes to these items, as described in the LA, without prior NRC approval, if the change meets the following criteria:

- It does not decrease the level of effectiveness of the design basis as described in the LA.
- It does not result in a departure from a method of evaluation described in the LA and used in establishing the design bases.
- It does not result in a degradation in safety.
- It does not affect compliance with applicable regulatory requirements.
- It does not conflict with an existing license condition.

If a change to the LA is made, the applicant will promptly update the affected onsite documentation per written procedures. MOX Services will maintain records of changes to its facility. These records include a written evaluation that provides the bases for the determination that the changes to the LA do not require prior NRC approval. The applicant retains these records until termination of the license. Changes are communicated to the NRC as follows:

- For changes that require NRC preapproval, MOX Services submits an amendment request to the NRC in accordance with 10 CFR 70.34 and 10 CFR 70.65, "Additional Content of Applications."
- For changes that do not require NRC preapproval of the LA, MOX Services submits to the NRC annually, within 30 days after the end of the calendar year during which the changes occurred, a brief summary of the changes.

The staff has reviewed the commitments and requirements for making changes to the LA. Consistent with the change process of 10 CFR 70.72, "Facility changes and change process," for the facility safety program, the requirements provided by the applicant have three key elements:

(1) the criteria used to evaluate changes to determine when preapproval by the staff is required

(2) the timeliness of updates to onsite documentation and reporting of changes to the staff

(3) the commitment to providing documentation for the evaluation for determining prior NRC approval and to maintaining records of changes

The staff finds that the criteria provided by the applicant for determining whether prior NRC approval is needed are consistent with the type of changes that would be made to the LA. The staff finds that the timeliness required for prompt updating of the onsite documentation and the timeframe for reporting changes not requiring NRC prior approval are reasonable and consistent with the process for making changes to the safety program as described in 10 CFR 70.72. The staff also finds that the commitment to performing and documenting the evaluation of NRC prior approval and maintaining records is acceptable. The staff therefore finds that the authorization for making changes to the LA is acceptable.

REFERENCES

10 CFR Part 70 *U.S. Code of Federal Regulations*, “Domestic Licensing of Special Nuclear Material,” Part 70, Chapter I, Title 10, “Energy.”

Atomic Energy Act, as amended 1954.

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(DCS, 2006b) Duke Cogema Stone & Webster Mixed Oxide Fuel Fabrication Facility to USNRC, “Request for Exemption from Indemnity Agreement and Financial Protection Requirements,” Aiken, SC, September 27, 2006.

(DCS, 2006c) Duke Cogema Stone & Webster Mixed Oxide Fuel Fabrication Facility to USNRC, “Request for Exemption from Radiation Labeling Requirements,” Aiken, SC, September 27, 2006.

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10 CFR Part 20, Standards for Protection Against Radiation

10 CFR Part 40, Domestic Licensing of Source Material

10 CFR Part 74, Material Control and Accounting of Special Nuclear Material

10 CFR Part 140, Financial Protection Requirements and Indemnity Agreements

48 CFR Part 952, Solicitation Provisions and Contract Clauses