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NUREG-1951

Safety Evaluation Report for the Eagle Rock Enrichment Facility in Bonneville County, Idaho

Docket No. 70-7015

AREVA Enrichment Services, LLC

Office of Nuclear Material Safety and Safeguards

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ABSTRACT

This report documents the U.S. Nuclear Regulatory Commission (NRC) staff's review and safety and safeguards evaluation of AREVA Enrichment Services LLC's (AES's) application for a license to construct a gas centrifuge uranium enrichment facility and possess and use byproduct material, source material, and special nuclear material (SNM). The proposed facility is known as the Eagle Rock Enrichment Facility (EREF). AES proposes that the EREF be located in Bonneville County, Idaho, about 32 kilometers (20 miles) west northwest of the city of Idaho Falls. The EREF will possess natural, depleted, and enriched uranium, and will be authorized to enrich uranium up to a maximum of 5 weight percent uranium-235.

The objective of the NRC's review is to evaluate the facility's potential adverse impacts 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, administrative programs, and financial qualifications provided to ensure safe design and operation of the facility.

The NRC staff concludes, in this safety evaluation report, that the applicant's descriptions, specifications, 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.

Potential environmental impacts associated with the proposed facility and its reasonable alternatives will be addressed in a separate NRC document, the Final Environmental Impact Statement, which is expected to be issued in February 2011.

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EXECUTIVE SUMMARY

On December 30, 2008, AREVA Enrichment Services LLC (AES or the applicant) submitted, to the U.S. Nuclear Regulatory Commission (NRC), an application requesting a license, under Title 10 of the *Code of Federal Regulations* (10 CFR) Parts 30, 40, and 70, to possess and use byproduct material, source material, and special nuclear material (SNM) in a gas centrifuge uranium enrichment facility. The applicant proposes that the facility, known as the Eagle Rock Enrichment Facility (EREF), be located in Bonneville County, Idaho, about 32 kilometers (20 miles) west-northwest of the city of Idaho Falls. On April 23, 2009, AES submitted a revised application to increase the facility's nominal capacity from 3 million separative work units (SWU)/year to 6 million SWU/year. (A SWU is a unit of enrichment that measures the effort required to separate isotopes of uranium). The facility will possess natural, depleted, and enriched uranium, and will be authorized to enrich uranium up to a maximum of 5 percent uranium-235. The applicant also requested a facility clearance for classified information under 10 CFR Part 95.

The NRC staff conducted its safety review in accordance with NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility." The staff's safeguards review involved reviews of the applicant's Fundamental Nuclear Material Control Plan; the Physical Security Plan, which includes transportation security; and a Standard Practice Procedures Plan for the Protection of Classified Matter. The staff also reviewed the applicant's Quality Assurance Program Description and Emergency Plan. Where the applicant's design or procedures should be supplemented, the NRC staff has identified license conditions to provide assurance of safe operation.

In conducting its safety review, the staff assessed, among other things, whether the AES's proposed equipment, facilities, and procedures will adequately protect public health and safety. The staff evaluated AES's existing facility designs and procedures, which are in various stages of completion, together with the applicant's stated commitments to complete certain design and procedures according to criteria specified by the applicant. If the Commission issues a license, it may contain conditions and limitations imposed to assure compliance with applicable regulations. The determination whether there is reasonable assurance that public health and safety will be adequately protected will be based in part on a comparison of the regulatory requirements against existing information, applicant commitments, and conditions imposed by the staff.

Once a license is granted (assuming the NRC's decision is to issue a license), construction of the facility may begin. In accordance with 10 CFR 70.72(d)(2), the applicant (then licensee) will submit to the NRC annual updates to the Integrated Safety Analysis Summary, along with a brief summary of the changes made during the year. The NRC will review these submissions, as well as any license amendment requests that may be submitted.

Although the applicant (then licensee) can start construction following issuance of the license, it may not begin operation of the EREF until after it successfully completes a second step. Prior to operation, the NRC must verify through inspection that the facility has been constructed in accordance with the requirements of the license (see 10 CFR 70.32(k)). Only after this step is successfully completed will the enrichment facility be allowed to begin operations.

The staff used several guidance documents to evaluate the applicant's license application and to complete the Safety Evaluation Report for the EREF. As previously mentioned, the primary guidance document used by the staff is NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility." As needed, the staff used other NUREG and Regulatory Guidance documents in its review. For the staff's review of the applicant's chemical safety program, the staff used NUREG-1601, "Chemical Process Safety at Fuel Cycle Facilities" and NUREG-1513, "Integrated Safety Analysis Guidance Document" in addition to NUREG-1520. For the staff's review of the Material Control and Accounting section of the Safety Analysis Report (SAR), the staff used NUREG/CR-5734 "Recommendations to the NRC on Acceptable Standard Format and Content for the Fundamental Nuclear Material Control Plan Required for Low-Enriched Uranium Facilities," in addition to NUREG-1520. For its review of the security-related portions of the SAR, the staff used Regulatory Guide 5.59, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance" and NRC Regulatory Issue Summary 2005-22, "Requirements for the Physical Protection During Transportation of Special Nuclear Material of Moderate and Low Strategic Significance: 10 CFR Part 73 vs. Regulatory Guide 5.59 (1983)."

NUREG-1520 was the primary document used to assist in defining the scope, level of detail, and acceptance criteria for reviews. NUREG-1520 provides generic guidance for reviewing and evaluating the health, safety, and environmental protection aspects of applications for licenses to possess and use SNM in fuel cycle facilities. The principal purpose of NUREG-1520 is to ensure the quality and uniformity of reviews conducted by the staff. Because NUREG-1520 describes the scope, level of detail, and acceptance criteria for reviews, it also serves as regulatory guidance for applicants who need to determine what information to present in a license application. Because NUREG-1520 is regulatory guidance, it does not preclude applicants from identifying portions of NUREG-1520 that are not directly applicable or from suggesting alternative approaches to those specified in NUREG-1520 to demonstrate compliance with applicable regulations. Should an applicant suggest alternative approaches, the staff retains the responsibility to make an independent determination concerning the adequacy of the applicant's suggested approaches.

NUREG-1520 was developed as a generic document for licensing fuel cycle facilities under 10 CFR Part 70, including fuel fabrication facilities and uranium enrichment facilities. Extensive communication occurred with current fuel cycle licensees to ensure that all necessary safety and environmental issues were addressed. While it is true that there are differences among these types of plants and among the relative risks of certain hazards at different fuel cycle facilities, hazards that will exist at the proposed EREF are similar to the types of hazards at other fuel cycle facilities for which NUREG-1520 was prepared. These hazards include handling of uranium hexafluoride (UF_6) cylinders; processing of UF_6 as a gas, and sometimes as a liquid; use of autoclaves or similar devices for feeding and sampling uranium; nuclear criticality; and equipment decontamination operations.

Based on relative risk, the staff adapts NUREG-1520 to review applications for different types of 10 CFR Part 70 facilities. The relative risk of the applicant's facility depends on the specific hazards associated with a particular technology (e.g., enrichment facility or fuel fabrication facility). The staff's review of each type of facility focuses on those specific types of hazards. The goal of the review is to determine whether applicable regulatory requirements are met to ensure that an adequate level of safety is provided to protect the health and safety of the workers, public, and the environment. Specific regulatory requirements for each type of facility are found in the applicable sections of NRC's regulations. The staff recognizes that the types and magnitudes of potential hazards vary greatly between the various types of facilities and even within each type of hazard.

As part of the license application, AES submitted an Environmental Report, which is under a parallel review by the NRC staff. The NRC staff will use the Environmental Report to prepare the Final Environmental Impact Statement, which is expected to be issued in February 2011.

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

General Information

AES provided an adequate overview of the facility layout and a summary description of the proposed facility and processes so that the staff has an overall understanding of the relationships of the facility features, as well as the function of each feature.

AES also adequately described and documented the corporate identity, structure, financial information, and the types, forms, and quantities and proposed purpose and authorized uses of licensed materials to be permitted at the facility. The staff plans to impose a license condition concerning funding for each incremental phase of the EREF.

AES also provided information on liability insurance. Because full liability insurance coverage will not be provided until prior to receipt of licensed material, the staff plans to impose a license condition for AES to provide proof of full liability insurance prior to obtaining licensed material.

AES also provided information on its plans to protect classified matter, including security controls and procedures, to ensure that classified matter is used, processed, stored, reproduced, transmitted, transported, and destroyed appropriately. Because a specific facility for use and storage of classified matter has not been identified, the staff plans to impose a license condition prohibiting AES to use, process, store, reproduce, transmit, handle, or allow access to classified matter, except provided by applicable personnel and facility clearances. The staff also plans to impose a license condition requiring notification to NRC once AES designates areas for the routine use and handling of classified information.

Under 10 CFR 40.14 and 10 CFR 70.17, AES requested an exemption from 10 CFR 40.36(d) and CFR 70.25(e) to provide forward-looking incremental funding for decommissioning. The staff will grant the exemption and impose a license condition to address AES's commitments for updating the decommissioning funding plan over time.

In addition, AES requested special authorization to make changes to the license application that do not decrease the effectiveness of its safety commitments in the license application without prior NRC approval. The staff will grant the authorization and impose a license condition for address the criteria, documentation, and reporting of changes made to the license application without prior NRC approval.

AES also adequately described and summarized information concerning site geography, nearby population, meteorology, hydrology, and geology, and potential effects of natural phenomena at the facility.

Organization and Administration

AES adequately described the applicant's organization, key management positions, qualifications, and management controls—including the responsibilities and associated resources for the design, construction, and operation of the facility and its plans for managing the project. The plans and commitments described in the SAR provide reasonable assurance that an acceptable organization, administrative policies, and sufficient, competent resources have been or will be established in such a manner as will allow for safe operation of the facility.

Integrated Safety Analysis (ISA) and ISA Summary

AES provided sufficient information about the facility, facility processes, hazards, and types of accident sequences. Area boundaries, including the controlled area boundary and the locations of restricted areas, are adequately described. AES commits to maintaining process safety information and to conducting an ISA. AES's ISA methodology uses appropriate methods for identifying potential accidents, determining consequences, and deriving likelihood. The likelihood evaluation, including the definition of "not credible," and the determination of chemical consequences are acceptable. AES has addressed the baseline design criteria and defense-in-depth practices required of new facilities.

The staff will impose a license condition to ensure that the boundaries of the items relied on for safety (IROFS) are developed and available to staff prior to the operational readiness review. The staff will also impose a license condition concerning the conduct of human factors engineering reviews of the human-system interfaces supporting IROFS requiring operator actions. And, the staff will impose a license condition addressing the design of IROFS that may use software, firmware, microcode, programmable controllers, and/or any digital device.

AES's ISA methodology and ISA Summary meet the requirements of 10 CFR Part 70.

Radiation Protection

AES provided sufficient information to evaluate its Radiation Protection Program. The SAR adequately describes: (a) the program for ensuring that worker and public doses are as low as (is) reasonably achievable (ALARA); (b) the qualification requirements for radiation protection personnel; (c) AES's commitment to approved, written radiation protection procedures and radiation work permits; (d) the training for all personnel who have access to restricted areas; (e) the program to control airborne concentrations of radioactive material with engineering controls and respiratory protection; (e) the radiation survey and monitoring program; and (f) recordkeeping. AES's Radiation Protection Program meets the requirements of 10 CFR Parts 19, 20, 30, 40, and 70.

Nuclear Criticality Safety

AES provided sufficient information to evaluate its Nuclear Criticality Safety (NCS) Program. The SAR adequately describes: (a) AES's commitment to have a group of qualified staff to develop, implement, and maintain the NCS Program; (b) AES's NCS methodologies and technical practices; and (c) the criticality accident alarm system. AES's NCS methodologies and technical practices will provide an adequate margin of subcriticality, and the safety programs and management measures will ensure that the margin of subcriticality is maintained. AES's NCS Program meets the requirements of 10 CFR Part 70.

Chemical Process Safety

AES adequately described and assessed accident consequences that can result from the handling, storage, or processing of licensed materials that can potentially have significant chemical consequences and effects. AES performed hazard analyses that identified and evaluated those chemical process hazards and potential accidents and established safety controls providing reasonable assurance of safe facility operation. AES's plan for managing chemical process safety and chemical process safety controls meets the requirements for 10 CFR Part 70.

Fire Safety

AES provided sufficient information to evaluate the potential fire hazards, consequences, and required controls for the proposed processes. The applicant identified a reasonable set of controls and defense-in-depth protection to meet the performance requirements of 10 CFR 70.61 and the baseline design criteria and defense-in-depth practices required of new facilities.

Emergency Management

AES provided an adequate Emergency Plan for the facility that meets the regulatory requirements. The facility will be properly configured to limit releases of radioactive materials in case of an accident; the capability will be established for measuring and assessing the significance of accidental releases of radioactive materials; appropriate emergency equipment and procedures will be provided; a system will be established to notify Federal, State and County government agencies; and recovery procedures will be established to return the facility to a safe condition after an accident. AES commits to implement the Emergency Plan through approved, written procedures.

Environmental Protection

AES has developed a program to implement adequate environmental protection measures during operation, including: (a) environmental and effluent monitoring and (b) effluent controls to maintain public doses ALARA as part of the Radiation Protection Program. The applicant's program is adequate to protect the environment and the health and safety of the public, and complies with the regulatory requirements in 10 CFR Parts 20, 30, 40, 51, and 70.

Decommissioning

Under 10 CFR 70.38(i), AES requested an alternate schedule for decommissioning since it expects decommissioning will take about eight years. 10 CFR 70.38(h) requires that decommissioning be completed no later than 24 months following the initiation of decommissioning, unless the Commission approves an alternate schedule. The staff reviewed the request in light of the considerations listed in 10 CFR 70.38(i)(1)-(i)(5) and determined that the alternative schedule is warranted.
AES provided a conceptual decontamination and decommissioning plan including the initial site characterization data and plans to collect and analyze additional samples to determine a background value for the site. The staff will impose a license condition to require AES to implement its plans to collect and analyze additional samples.

The applicant submitted an adequate decommissioning funding plan cost estimate that is consistent with NUREG-1557, Volume 3, and satisfies the regulatory requirements of 10 CFR Parts 30, 40, and 70. The applicant also provides a draft letter of credit, standby trust agreement, and supporting documentation. The staff will impose a license condition concerning AES's method of providing decommissioning financial assurance on an incremental, forward-looking basis.

Management Measures

AES has adequately described the management measures that will be applied to the proposed facility. The information describes: (a) the organization responsible for developing, implementing, and assessing the management measures; (b) the commitment to quality assurance elements and administrative measures for staffing, performance, assessing findings, and implementing corrective actions; (c) the process for the development, approval, and implementation of procedures; (d) the commitment to establishing and documenting surveillances, tests, and inspections to provide reasonable assurance of satisfactory performance of the IROFS; (e) the audit program; and (f) training requirements. AES's quality assurance staff has the independence and authority to carry out their function without undue influence and quality assurance elements cover the IROFS. The proposed management measures meet the regulatory requirements of 10 CFR Part 70.

Physical Protection

AES provided an adequate Physical Security Plan including information on the policies, methods, and procedures to be implemented to protect SNM of low strategic significance used and possessed at the facility. In the Physical Security Plan, AES also provided information on the policies, methods, and procedures to be implemented to protect SNM of low strategic significance in transit to and from the facility. The Physical Security Plan is acceptable and meets the requirements in Part 73.

Materials Control and Accountability

AES provided an adequate Fundamental Nuclear Material Control (FNMC) Plan for the proposed facility that will meet the applicable 10 CFR 74 requirements. The staff will impose a license condition requiring AES to maintain and follow the FNMC Plan.

ACRONYMS AND ABBREVIATIONS

ac	acres
ACI	American Concrete Institute
AEC	active engineered control
AEGL	Acute Exposure Guideline Level
AES	AREVA Enrichment Services LLC
AISC	American Institute of Steel Construction
ALARA	as low as is reasonably achievable
Al_2O_3	aluminum oxide
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARF	airborne release fraction
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing and Materials
BDC	baseline design criteria
BHS	Bureau of Homeland Security
BLM	Bureau of Land Management
Bq	Becquerel
BSPB	Blending, Sampling and Preparation Building
BTP	branch technical position
°C	degree(s) Celsius
CAA	controlled access area
CAAS	criticality accident alarm system
CAB	Centrifuge Assembly Building
CAP	Corrective Action Program
CEDE	Committed Effective Dose Equivalent
CFR	Code of Federal Regulations
Ci	curie
cm	centimeter
CM	configuration management
CMP	Classified Matter Plan
CRSB	Cylinder Receipt and Shipping Building
DBE	Design Basis Earthquake
DFP	Decommissioning Funding Plan
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
DR	damage ratio
DU	depleted uranium
EBR	Argonne National Lab- West
EHS&L	Environmental, Health, Safety and Licensing
EIS	Environmental Impact Statement
EOC	Emergency Operations Center
EP	Emergency Plan
EPA	U.S. Environmental Protection Agency

EPIP	emergency plan implementing procedure
ER	Environmental Report
EREF	Eagle Rock Enrichment Facility
ERO	emergency response organization
ETC	Enrichment Technology Company Limited
°F	degree(s) Fahrenheit
FA	financial assurance
FHA	Fire Hazard Analysis
FNMC	Fundamental Nuclear Material Control
FOCI	foreign ownership, control, or influence
FTC	full tails cylinder
ft	feet
g	gram
gal	gallon
GEVS	gaseous effluent ventilation system
gpm	gallons per minute
ha	hectare
HAZOP	hazard and operability
HEPA	high efficiency particulate air
HF	hydrogen fluoride
HFE	human factors evaluation
HPS	Health Physics Society
HS&E	Health, Safety, and Environment
H/U	hydrogen/uranium ratio
HVAC	Heating, Ventilating, and Air Conditioning
H_2O	Water
IBC	International Building Code
ICBO	International Conference of Building Officials
in	inch
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
IROFS	items relied on for safety
ISA	Integrated Safety Analysis
ISO	International Organization for Standardization
IT	information technology
kg	kilogram
km	kilometer
L	liter
lb	pound
LCS	local control system
LEL	lower explosive limit
LES	Louisiana Energy Services
LLW	low-level waste
LOC	letter of credit

LPF	leak path factor
lpm	liter per minute
LSS	low strategic significance
LTTS	Low Temperature Take-off Station
m	meter
m^3	cubic meter
MAR	material at risk
MBq	megabecquerel
MC&A	material control and accounting
MFC	Materials and Fuels Complex
mg	milligram
mi	mile
ml	milliliter
mm	millimeter
MOU	Memorandum of Understanding
Mph	miles per hour
m/s	meter per second
mrem	milli Roentgen equivalent man
Ms	surface wave magnitude
mSv	millisievert
MT	*metric tonne*
Mw	momentum magnitude
N/A	not applicable
NaF	sodium fluoride
NAVFAC	Naval Facilities Engineering Command
NCS	Nuclear Criticality Safety
NEI	Nuclear Energy Institute
NELAC	National Environmental Laboratory Accreditation Conference
NELAP	National Environmental Laboratory Accreditation Program
NESHAPS	National Emission Standards for Hazardous Air Pollutants
NFPA	National Fire Protection Association
NIOSH	National Institute for Occupational Safety and Health
NIST	National Institutes of Standards and Technology
NMSS	Office of Nuclear Material Safety and Safeguards
NOAA	National Oceanic and Atmospheric Administration
NRC	U.S. Nuclear Regulatory Commission
OJT	on-the-job training
OSB	Operation Support Building
OSHA	U.S. Occupational Safety and Health Administration
PAR	protective action recommendation
PFPE	perfluorinated polyether
PM	preventive maintenance
PNNL	Pacific Northwest National Laboratory
PSC	Process Service Corridor
psia	pounds-force per square inch absolute
psig	pounds-force per square inch gauge

PSHA	probabalistic seismic hazard assessment
PSP	Physical Security Plan
QA	quality assurance
QAPD	Quality Assurance Program Description
QC	quality control
RAI	request for additional information
RASCAL	Radiological Assessment System for Consequence Analysis
rem	Roentgen equivalent man
REMP	Radiological Environmental Monitoring Program
RF	respirable fraction
RG	Regulatory Guide
RP	radiation protection
RWP	Radiation Work Permit
SAR	Safety Analysis Report
SBM	Separations Building Module
SER	Safety Evaluation Report
SNM	special nuclear material
SOP	Standard Operating Procedure
SPCC	spill prevention, control and countermeasures
SPPP	Standard Practice Procedures Plan
SRC	Safety Review Committee
SRP	Standard Review Plan
SSC	structures, systems, and components
ST	source term
Sv	sievert
SWU	separative work unit
Tc	technetium
TEDE	total effective dose equivalent
TLD	thermoluminescent dosimeter
TSB	Technical Support Building
TSP	Transportation Security Plan
TWA	time-weighted average
U	uranium
µCi	microcurie
µg	microgram
UF_4	uranium tetrafluoride
UF_6	uranium hexafluoride
UHRS	uniform hazard response spectra
UO_2F_2	uranyl fluoride
UPS	uninterruptible power supply
USEC	U.S. Enrichment Corporation
USGS	U.S. Geological Survey
USL	upper subcritical limit
wt	weight

yd	yard
yd^3	cubic yard
yr	year

CHAPTER 1.0 GENERAL INFORMATION

1.1 Facility and Process Description

The purpose of the U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's facility and process description is to evaluate whether the application includes an overview of the facility layout and a summary description of the proposed processes. A more detailed description of the facility and processes is contained in the Integrated Safety Analysis (ISA) Summary (AES, 2010b).

1.1.1 Regulatory Requirements

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) 30.33, 10 CFR 40.32, and 10 CFR 70.22 require each application for a license to include information on the proposed activity and the equipment and facilities that the applicant will use to protect public health and safety and minimize danger to life and property. In addition, the regulations in 10 CFR 70.65 require each application to include a general description of the facility, with emphasis on those areas that could affect safety, including identification of the controlled area boundaries.

1.1.2 Regulatory Guidance and Acceptance Criteria

The guidance applicable to the NRC's review of the facility and process description section of the Safety Analysis Report (SAR) (AES, 2010a) is contained in Chapter 1 of "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 (NRC, 2002). This chapter is applicable in its entirety. The acceptance criteria applicable to this review are contained in Section 1.1.4.3 of NUREG-1520 (NRC, 2002).

1.1.3 Staff Review and Analysis

In Section 1.1 of the SAR (AES, 2010a), the applicant, AREVA Enrichment Services (AES), provided a summary description of the proposed gas centrifuge uranium enrichment facility and processes. The summary description includes discussion of the major chemical and mechanical processes to be used in the facility.

The applicant is proposing to use a gas centrifuge enrichment process to enrich natural uranium by separating a feed stream containing a natural isotopic concentration of about 0.7 weight percent uranium-235 (U-235) into a product stream enriched in U-235 at 5 weight percent and a tails stream depleted in U-235. The proposed plant, known as the Eagle Rock Enrichment Facility (EREF), will have a plant design capacity of 6.6 million SWUs per year (AES, 2010a). (A SWU is a measure of the effort required to perform isotopic separation.) The plant design capacity accounts for production margin for centrifuge failures and occasional production losses during the operational lifetime of the facility, thus the nominal production capacity will be 6 million SWUs per year (AES, 2010a).

The feed material for the enrichment process will be uranium hexafluoride (UF_6) with a natural composition of isotopes U-234, U-235, and U-238 (AES, 2010a). The enrichment process will be a mechanical separation of isotopes using a fast rotating cylinder (centrifuge) based on a difference in centrifugal forces. Gaseous natural UF_6 will enter a high-speed centrifuge. Centrifugal forces press the heavier isotope of uranium, U-238, to the outer wall of the centrifuge. The lighter isotope, U-235, will remain closer to the center, away from the wall of the centrifuge. Internal scoops will be used to collect the heavier and lighter fractions and circulate them to other centrifuges piped in a cascade arrangement. No chemical changes or nuclear reactions will take place. The feed, product, and tails streams will all be in the form of UF_6.

The applicant has proposed that the EREF be located in Bonneville County, Idaho approximately 32 kilometers (km) (20 miles [mi]) west northwest of the city of Idaho Falls. The facility will be located on piece of property of about 1,700 hectares (ha) (4,200 acres [ac]) (AES, 2010a). The footprint of the facility will be about 172 ha (426 ac) in the north central portion of the plot (AES, 2010a).

The property is currently privately-held by a single landowner (AES, 2010a). This privately-held land will be purchased by AES. Within the proposed site, there is also a 16 ha (40 ac) parcel which is administered by the U.S. Bureau of Land Management (AES, 2010a).

The major structures of the facility are described in Section 1.1.2 of the SAR (AES, 2010a). The EREF facility will consist of multiple buildings, each of which will perform a specific function. A list of selected buildings and their specific functions follows:

Separations Building Modules (SBMs): The EREF will include four identical SBMs. Each module will consist of two cascade halls. Each hall will have twelve cascades with each cascade having hundreds of centrifuges.

Technical Support Building (TSB): This building will contain the radiological support areas for the EREF, including solid waste collection, workshops, liquid effluent collection and treatment, laundry sorting, laboratories, radiation monitoring, truck bay/shipping and receiving, ancillary areas, and maintenance for contaminated facility equipment.

Operation Support Building (OSB): This building will contain the non-radiological support for areas for the EREF, including workshops, medical, lockers, cafeteria, lobby, ancillary areas, control room, training, security alarm system, and environmental laboratory.

Centrifuge Assembly Building (CAB): This building will be used to assemble centrifuges before they are moved into the SBMs.

Administration Building: This building will include general office areas.

Security and Secure Administration Building: This building will contain secure office areas and the Entry-Exit Control Point for the EREF.

Guard House: This building will function as the security checkpoint for all incoming and outgoing traffic.

Cylinder Receipt and Shipping Building: This building will be used to receive, inspect, weigh, and temporarily store cylinders of feed UF_6; to temporarily store, inspect, weigh, and ship cylinders of enriched UF_6; to receive, inspect, weigh, and temporarily store empty and depleted uranium tails cylinders; and to transfer filled depleted uranium cylinders to the Full Tails Cylinder Storage Pads.

Blending, Sampling, and Preparation Building: This building will be used to fill cylinders and to liquefy, homogenize, and sample cylinders prior to shipment to customers.

Cylinder Storage Pads: Outside storage areas will be used for the storage of full cylinders containing UF_6 and of empty cylinders. Feed cylinders, containing natural UF_6, will be temporarily stored on the Full Feed Cylinder Storage Pads, prior to use in the facility. Tails cylinders, containing depleted UF_6, will be temporarily stored on the Full Tails Cylinder Storage Pads. Product cylinders, containing enriched UF_6, will be stored on the Full Product Cylinder Storage Pad, prior to shipment offsite to a fuel fabrication facility. Empty cylinders will be temporarily stored on the Empty Cylinder Storage Pads. The Full Feed, Full Tails, Full Product, and Empty Cylinder Storage Pads will be sized to store approximately 712 feed cylinders, 25,718 tails cylinders, 1,032 product cylinders, and 1,840 empty cylinders, respectively.

Electrical Services Building: This building will house four standby diesel generators, day tanks, switchgear, control panels, and building heating, ventilation, and air conditioning (HVAC) equipment.

Mechanical Services Buildings: There will be two mechanical services buildings; and they will house air compressors, the demineralized water system, centrifuge cooling water system pumps, heat exchangers, and expansion tanks.

Electrical Services Building for the CAB: This building will house four transformers and switchgear.

Gasoline and Diesel Fuel Station: The station will include a fuel pump island for vehicle fueling and a building for on-site vehicle repair and maintenance.

Visitor Center: This building will be located outside the security fence area.

Natural uranium feed will be shipped to the EREF by truck in cylinders having a nominal capacity of 12.5-metric tonnes (MT) (14-ton) of UF_6. Under ambient conditions, the UF_6 is a solid. The feed cylinders will be loaded into the solid feed stations, vented to remove the light gases—primarily air and hydrogen fluoride (HF)—and then heated to sublime the solid UF_6 to a gas. The light gases and UF_6 gas generated will be routed to the feed purification subsystem to remove any light gases from the UF_6 feed prior to introduction into the cascades. After purification, the gaseous UF_6 from the solid feed stations will be directed to a cascade for enrichment.

After enrichment in the cascade, both depleted and enriched UF_6 will be withdrawn and desublimed at subatmospheric pressure in the tails take-off system and the product take-off system, respectively. Tails and product take-off systems will be designed to preclude UF_6 from becoming a liquid. The product take-off system will also contain a system to purge light gases.

Sampling to verify the product assay level will be performed in the product liquid sampling system. In an autoclave, UF_6 will be heated to a liquid; the cylinder will be tilted so that UF_6 can flow into sample manifold and sample bottles; and the cylinder will be returned to its original horizontal position. This is the only system in the plant where UF_6 will be in a liquid form. To produce enriched uranium meeting customer-assay specifications, a product blending system will be used to mix enriched uranium at two different enrichment levels to meet the customer specifications. This system could also be used to transfer product between cylinders. The SAR (AES, 2010a) and the ISA Summary (AES, 2010b) provide additional descriptions and process details, including drawings of the plant buildings and the location of plant systems within the buildings.[1] Geographical features of the site are also provided on these drawings.

The EREF will possess natural, enriched, and depleted uranium. The feed, product, and tail streams will all be in the form of UF_6. At full capacity, the EREF will handle, on an annual basis, 1,424 nominal 12.5-MT (14-ton) natural uranium feed cylinders; 1,032 nominal 2-MT (2.5-ton) enriched uranium product cylinders; and 1,222 nominal 12.5-MT (14-ton) depleted uranium tails cylinders (AES, 2010a).

Solid waste will be generated at the EREF, including, Class A low-level radioactive waste, low-level mixed waste, hazardous waste, and industrial (non-hazardous) waste (AES, 2010a). The SAR provides estimates of the annual quantities generated (AES, 2010a). Table 1.1-2 of the SAR provides estimates of the annual quantities of radiological and mixed wastes produced, including uranium content. Table 1.1-4 of the SAR provides estimates of the annual quantities of non-radiological wastes produced. Construction wastes will also be generated during construction of the facility. Table 1.1-5 of the SAR provides estimates of the annual hazardous wastes produced during construction. All low-level radioactive waste and mixed waste will be disposed of at a licensed low-level waste disposal facility (AES, 2010a). All hazardous chemical wastes will be transported to permitted treatment and disposal facilities (AES, 2010a).

Depleted UF_6 tails will be generated at the EREF. Table 10.3-1 of the SAR provides estimates of the tails production during the operating life of the EREF. AES intends to transfer depleted UF_6 to the U.S. Department of Energy (DOE) for conversion and disposal. Depleted UF_6 tails will be stored onsite in cylinders on the cylinder storage pad until the cylinders are transferred to the DOE (AES, 2010a). Once removed, depleted UF_6 tails will be converted to a stable form and disposed of in accordance with the United States Enrichment Corporation Privatization Act and other statutory authorizations and requirements at DOE's conversion facility (AES, 2010a).

Gaseous airborne effluents will be released from the EREF. Table 1.1-1 of the SAR provides estimates of the annual gaseous effluents. The applicant estimates that less than 20 grams (g) (0.0441 pounds [lb]) of uranium and less than 2 kilogram (kg) (4.4 lb) of HF will be released annually in 4.13×10^9 cubic meters of air discharge. Sources of air emissions include the gaseous effluent ventilation system and the HVAC systems. As discussed in Chapter 9 of this SER, these emissions will be significantly below the limits in 10 CFR Part 20 and the As Low as Reasonably Achievable (ALARA) program air effluent goals.

The applicant does not anticipate any liquid discharges of licensed radioactive materials from the EREF. The applicant will collect liquid effluent from plant processes and treat the collected

[1]Drawings of the facility layout, plant buildings, and plant systems within the buildings as well as the ISA Summary have been withheld from public release as security-related information under 10 CFR 2.390.

effluent to remove the uranic material (AES, 2010a). Table 1.1-3 of the SAR provides estimates of the annual liquid effluent collected from plant processes (AES, 2010a). The applicant estimated that a total of about 57,100 liters (L) /year (yr) (15,625 gallons [gal]/yr) of liquid will be generated and treated, containing a total of 114 kg (251 lb) of uranic material. Sources of the liquid effluent to be collected include laboratory effluent, floor washings, condensates, degreaser water, and spent citric acid. The Liquid Effluent Collection and Treatment System for the EREF will employ precipitation and filtration to remove uranic material. The final stage of evaporation will release the treated distillate directly into the atmosphere. The applicant estimated the amount of uranic material in the evaporated distillate to be less than 0.04 grams (g)/yr of total uranium. The uranium removed during treatment will be disposed of offsite at a licensed disposal facility. As discussed in Chapter 9 of this SER, these emissions will be significantly below the limits in 10 CFR Part 20 and the ALARA program air effluent goals.

As described above, the applicant provided information at a level of detail that is appropriate for general familiarization and understanding of the proposed facility and processes. The application summarizes the facility information contained in the ISA Summary and includes descriptions of the overall facility layout on scaled drawings, including the site's geographical features and facility structural features. The applicant's summary also describes the relationship of specific facility features to the major processes that will be ongoing at the facility. The major chemical and mechanical processes involving licensable material are described in summary form, based in part on information presented in the ISA Summary. This description includes: (a) reference to the building locations of major process components; (b) brief descriptions of the process steps; (c) the chemical forms of licensable material in process; and (d) the types, amounts and discharge points of waste materials discharged to the environment from the processes. The applicant presented a summary of the feed materials, by-products, waste, and finished products of the facility. Information concerning expected levels of trace impurities or contaminants appears in Table 1.2-1 of the SAR and is discussed in Section 1.2.3.4 of this SER. The information the applicant provided meets the guidance in Section 1.1.4.3(1), (2), (3), and (4) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

1.1.4 Evaluation Findings

The staff has reviewed the proposed general facility and process descriptions for the EREF according to Section 1.1 of NUREG-1520 (NRC, 2002). The applicant has adequately described: (1) the facility and processes so that the staff has an overall understanding of the relationships of the facility features; and (2) the function of each feature. The staff concludes that the applicant has met the requirements and acceptance criteria applicable to this section.

1.2 Institutional Information

The purpose of NRC's review of the applicant's institutional information is to evaluate whether the application includes adequate information identifying the applicant, the applicant's characteristics, and the proposed activity.

1.2.1 Regulatory Requirements

The regulations in 10 CFR 30.32 and 10 CFR 40.31 require each application for a license to include: (a) information on the identity of the applicant; (b) name, chemical and physical form, and maximum amount that will be possessed; and (c) purpose for which the licensed material will be used. The regulations in 10 CFR 70.22 require each application for a license to include: (a) information on the corporation applying for a license; (b) the location of the principal office; (c) the names and citizenship of the principal officers; (d) information concerning ownership and control; (e) the proposed site activities; (f) financial qualifications; and (g) the name, amount, and specifications of the licensed material to be used.

The regulations for financial qualifications are found in 10 CFR 70.22(a)(8) and 10 CFR 70.23(a)(5). 10 CFR 70.22(a)(8) requires an applicant to submit information with respect to its financial qualifications, and 10 CFR 70.23(a)(5) requires that an applicant be financially qualified to engage in the proposed activities in accordance with the regulations. In addition, the staff took into consideration the Commission's ruling in Louisiana Energy Services, L.P. (Claiborne Enrichment Center), CLI-97-15, 46 NRC 294 (1997), which pertains to an application by Louisiana Energy Services (LES) to construct and operate a uranium enrichment facility pursuant to 10 CFR Part 70. Among other things, the ruling held that "the NRC is not required as a matter of law to apply the strict financial qualification provisions of Part 50 to all Part 70 license applications." *Id.*, 46 NRC at 298. In summary, the Commission concluded that "the general language of Part 70 leaves the Commission free to review the reasonableness of an applicant's financial plan in light of all relevant circumstances," which might or might not lead to application of any or all of the criteria stated in Part 50. *Id.* at 302.

The staff also considered other regulatory requirements, including the following. The regulations found in 10 CFR 40.38 and 10 CFR 70.40 place restrictions on the eligibility of certain applicants to hold licenses. The regulations in 10 CFR 70.22(m) identify the requirements to protect against unauthorized viewing of classified enrichment equipment. 10 CFR Part 95 contain provisions to protect against unauthorized disclosure of classified matter and requirements for obtaining a facility security clearance. The regulations in 10 CFR 140.13(b) require applicants for uranium enrichment facilities to provide and maintain liability insurance.

1.2.2 Regulatory Guidance and Acceptance Criteria

The acceptance criteria applicable to NRC's review of the institutional information section of the application are contained in 10 CFR 30.32, 10 CFR 40.31, 10 CFR 40.38, 10 CFR 70.22, 10 CFR 70.23(a)(5), 10 CFR 70.40, 10 CFR Part 95, 10 CFR 140.13b, and Section 1.2.4.3 of NUREG-1520 (NRC, 2002). Chapter 1 of NUREG-1520 is applicable to the EREF in its entirety. Section 1.2.3.6, "Special Exemptions or Special Authorizations," of this chapter, addresses exemptions and special authorizations.

In addition, the EREF classified matter plan, "Security Plan for the Protection of Classified Matter" was reviewed for compliance with the requirements of 10 CFR Part 95 by using "Standard Practice Procedures Plan Standard Format and Content for the Protection of Classified matter for NRC Licensee, Certificate Holder, or Other Activities as the Commission May Determine" (NRC, 2006).

1.2.3 Staff Review and Analysis

1.2.3.1 Corporate Identity

In Section 1.2.1 of the SAR (AES, 2010a), AES, provided information on the corporate identify and location and ownership organization.

AES is a Delaware limited liability corporation with a principal location of business in Bethesda, Maryland. It has been formed solely to provide uranium enrichment services for commercial nuclear power plants. AES is a wholly owned subsidiary of AREVA NC Inc.

AREVA NC Inc. is a wholly owned subsidiary of AREVA NC SA, which is part of AREVA SA (AREVA), a corporation formed under the laws of France. The principal owners of AREVA SA are identified in Section 1.2.1.2 of the SAR (AES, 2010a) and include the Commissariat a l'Energie Atomique (French Atomic Energy Commission) and the French State.

AES is governed by the AES Management Committee. Section 1.2.3.2 of the SAR (AES, 2010a) identifies the Committee members. The President and Chief Executive Officer of AES is a naturalized citizen of the United States, as well as a citizen of Canada. Any safety decision related to the operation of the facility will be made by the President of AES.

No other companies will be present or operating on the uranium enrichment plant property other than where the applicant has contracted such services.

The AES principal office is located at 4800 Hampden Lane, Bethesda, Maryland 20814.

The EREF will be located in Bonneville County, Idaho, along State Highway 20. The following is the legal description of the EREF site location, as described in the SAR (2010a):

> "All of Sections 13, 14 and 15; the Northeast quarter (NE1/4) of Section 21; the North half (N1/2), and Southeast Quarter of the Southeast Quarter (SE1/4 SE1/4) of Section 22; the North Half (N1/2), the Southeast Quarter (SE1/4), the East Half of the Southwest Quarter (E1/2 SW1/4), and the Southwest Quarter of the Southwest Quarter (SW1/4 SW1/4) of Section 23; West Half (W1/2), and the West Half of the Southeast quarter (W1/2 SE1/4), and the Northeast quarter of the Southeast quarter (NE1/4 SE1/4) and the Northwest quarter of the Northeast quarter (NW1/4 NE1/4) of Section 24; the West 1/2 (W1/2) of Section 25, Less the Highway and that portion of the SW1/4 deeded to the State of Idaho in a Warranty Deed recorded July 25, 1950, in Book 72 of Deeds, at page 565 and the Northeast quarter (NE1/4); the East Half of the Northwest Quarter (E1/2 NW1/4), the Northeast Quarter of the Southwest Quarter (NE1/4 SW1/4), the Northwest Quarter of the Southeast Quarter (NW1/4 SE1/4) and that portion of the South Half of the Southeast Quarter (S1/2 SE1/4) lying north of the centerline of State Highway 20 as surveyed and shown on the official plat of the Twin Buttes F-1422(2) Highway Survey on file in the office of the Department of Highway of the State of Idaho, all in Section 26; All in Township 3 North, Range 34 East of the Boise Meridian, Bonneville County, Idaho, contains four thousand two hundred and ten (4,210) acres, more or less."

As stated above, the applicant furnished its full name and address, a full description of the proposed facility site location, the state where the corporation is incorporated, and the location of the principal offices. The applicant also described the corporation's control and ownership, including any control or ownership exercised by a foreign entity. The applicant provided information on primary ownership and relationships to other components of the organization of the same ownership. The applicant described the presence and operations of any other organization on the site. The information the applicant provided meets the guidance in Section 1.2.4.3(1) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

1.2.3.2 Foreign Ownership, Control, or Influence

With respect to foreign ownership, control or influence (FOCI) for AES's EREF, the NRC staff has determined that any FOCI mitigation measures placed on AES would provide no additional benefit to the National Security of the United States. The staff believes that the recommendation made by the DOE to NRC by letter dated March 31, 2005 (letter has been withheld from public release as "Official Use Only") regarding waiving the FOCI requirement for FOCI mitigation associated with the granting of a facility security clearance to URENCO for the National Enrichment Facility, applies to AES's EREF. This is based on the fact that both URENCO and AES use, or will use, the same classified technology supplied by the Enrichment Technology Company Limited (ETC). The NRC decision is based on an Interagency Agreement between NRC and DOE dated May 6, 2002 (DOE, 2002). The following summarizes the basis for the decision to waive the requirement for FOCI mitigation for AES.

The information and technology that will be classified as restricted data in the United States are already owned and controlled by the European Governments and the foreign-controlled companies associated with URENCO and AREVA. This information and technology are only being classified under U.S. law by the fact that it's being introduced into the United States. The NRC has considered whether the operation of the EREF could generate new restricted data, not otherwise available to the European Governments or their nationals. All of the parties associated with URENCO and AREVA agree that little, if any, new restricted data should be created as a result of the AES facility. In addition, the Pentapartite Agreement (an agreement under development between the United States and four European Governments) will establish protocols to be followed if any new restricted data is created at the EREF and would prevent new restricted data from being disseminated to European nationals.

Thus, the staff finds that any additional FOCI mitigation measures placed on AES would provide no additional benefit to the National Security of the United States. Under 10 CFR 95, AES is required to complete the NRC facility clearance process which entails, among other things, an NRC-approved classified matter plan (CMP), referred to as a Standard Practice Procedures Plan; on-site inspection; and granting of individual NRC personnel security clearances. Section 1.2.4.3 of this SER evaluates AES's CMP for the protection of classified matter at the proposed EREF facility.

1.2.3.3 Financial Qualifications

1.2.3.3.1 Evaluation of Cost Estimate to Construct and Operate EREF

In Section 1.2.2 of the SAR (AES, 2010a) and in supplemental information provided on September 28, 2009 (AES, 2009a), AES estimated the total construction cost of EREF to be approximately $4.1 billion (2007 dollars) excluding escalation, contingency, interest, tails disposition, decommissioning, and any replacement equipment required during the life of the facility. AES asserted that investment in EREF at 6.6 million SWU will be divided into at least two (3.3 million SWU) phases. According to AES, the first phase may be broken down into smaller increments based on calendar time or construction phase (AES, 2009a). The second phase of construction may vary as well. AES stated that, depending on market conditions, the second 3.3 million SWU may be constructed in one or more phases timed with demand (AES, 2009a). AES has provided proprietary information regarding the estimated costs of the two phases. This proprietary information is discussed in Appendix C to this SER.
As part of the cost estimate analysis, the staff reviewed Section 1.1.2, "Facilities Description," of the SAR to ensure the support structures and systems required for operation were addressed and compared this information to NUREG-1827, "Safety Evaluation Report for the National Enrichment Facility in Lea County, New Mexico" (NRC, 2005) to ensure consistency. After this review, the staff finds that the EREF construction cost estimate encompasses the EREF structures and systems and is consistent with the basis of the cost estimate for a similar facility and therefore, is reasonable.

Furthermore, in Section 1.2.2 of the SAR (AES, 2010a), AES states that it will make available updated cost estimates for each phase prior to initiating construction. In Section 1.2.2 of the SAR, AES states that:

> Construction of each incremental phase of the EREF shall not commence before funding for that increment is available or committed. Of this funding, AES must have in place before constructing such increments, commitments for one or more of the following: equity contributions from AES or its parents, a commitment from the parent company to provide the necessary funds for the project, and lending and/or lease arrangements that solely or cumulatively are sufficient to ensure funding for the particular increment's construction costs. AES shall make available for NRC inspection, documentation of both the budgeted costs for each incremental phase and the source of funds available or committed to pay those costs.

The staff will impose this approach as a license condition.

1.2.3.3.2 Financial Information

In Section 1.2.2 of the SAR (AES, 2010a), AES proposed to satisfy the obligation to demonstrate the financial qualifications to carry out the proposed activities, as set forth in 10 CFR 70.23(a)(5) in a manner consistent with the approach previously accepted by the NRC staff in Section 1.2.3.3.2 of NUREG-1851, "Safety Evaluation Report for the American Centrifuge Plant in Piketon, Ohio" (NRC, 2006).

AES plans to fund Phase 1 of the project with a mix of approximately 62 percent debt and 38 percent equity from AES parent contributions and self-generated cash during pre-production. The remaining 62 percent of the estimated $3.2 billion Phase 1 construction costs will be financed through financial institutions (AEs, 2009a). The estimated construction costs for Phase 2 of the project are expected to be financed with a mix of approximately 70 percent debt from financial institutions and 30 percent equity from AES parent contributions and commitments and cash flow from AES operation (AES, 2009a). Furthermore, as discussed in Section 1.2.3.3.1 of this SER, AES has asserted that it shall make available for NRC inspection, documentation of the source of funds available or committed to pay costs prior to commencement of construction of each phase (AES, 2009a).

Additionally, AES stated that, as of October 2009, 55 percent of EREF Phase 1 output is committed from 2014 to 2028 with eight major U.S. utilities with 44 percent already contracted; and the remaining 11 percent committed through letters of intent and contract negotiations in the final stages. Furthermore, AES expects to have an additional 35 percent (for a total of 90 percent of Phase 1's output) contracted before the expected start of construction in 2011, with terms sufficient to guarantee a reasonable return on investment for the entire term of the off-take agreements (AES, 2009a).

AES has no reported income statements since its inception. However, AES' parent company, AREVA, has the assets necessary to provide the aforementioned equity contribution. As described in Section 1.2.3.3.1 of this SER, AES has proposed an approach for funding which the NRC staff will impose as a license condition. The proposed approach for funding includes a statement that equity contributions will be provided by either AES or its parent. In 2008, AREVA had total assets in excess of $48 billion, total revenue of more than $18 billion, a net income of $702 million, and cash and cash equivalents of $1.48 billion (AES, 2009a). Therefore, AES identified sources of debt and equity for construction, and additional financial resources if necessary.

The NRC staff finds that, based on the financial information submitted in the SAR (NRC, 2010) and RAI response (AES, 2009a), AES meets the financial qualifications for the proposed activities in accordance with 10 CFR 70.23(a)(5).

1.2.3.4 Liability Insurance

Under 10 CFR 140.13(b), a uranium enrichment facility is required to carry liability insurance to cover public claims arising from any occurrence, within the United States that causes, within or outside the United States, bodily injury, sickness, disease, death, loss of, or damage to, property, or loss of use of property arising from the radioactive, toxic, explosive, or other hazardous properties of chemicals containing licensed material.

According to the application, by letter dated December 22, 2008, the American Nuclear Insurers documented its expectation to provide nuclear liability insurance for EREF at a maximum policy amount of $300 million by the time AES takes possession of source material or special nuclear material (SNM). Because AES states in the SAR (AES, 2010a) that it will provide proof of, and maintain, nuclear liability insurance in the maximum available amount, the staff finds that the applicant satisfies the regulatory requirements under 10 CFR 140.13(b).

Because the liability insurance coverage will not be provided until AES takes possession of source material or SNM, NRC staff is imposing the following license condition:

> *The licensee shall provide proof of full liability insurance of $300 million, as required under 10 CFR 140.13(b), at least 30 days prior to the planned date for obtaining licensed material.*

1.2.4 Type, Quantity, and Form of Licensed Material

Table 1.2-1 of the SAR (AES, 2010a) lists the types, quantity, and form of licensed material proposed for acquisition, delivery, receipt, possession, production, use, transfer, and/or storage. Table 1.2-1 of this SER lists AES's proposed possession limits for SNM, source material, and by-product material.

The applicant has included technetium-99 (Tc-99), transuranic isotopes, and other contaminants in Table 1.2-1 of the SAR (AES, 2010a). These radionuclides may exist at the EREF as a consequence of the historical feed of uranium at other enrichment facilities, for example, process contaminants and waste or material held in UF_6 cylinders from previous operations. As indicated in Footnote 1 to Table 1.2-1 of the SAR (AES, 2010a), the applicant will require UF_6 suppliers to provide commercial natural UF_6 in accordance with the requirements of American Society of Testing and Materials (ASTM) C787-03, "Standard Specification for Uranium Hexafluoride for Enrichment" (ASTM, 2003). This standard contains the purity requirements of the uranium enrichment feed. In addition, cylinder suppliers will be required to preclude use of cylinders that, in the past, have contained reprocessed UF_6, unless they have been decontaminated. Periodic audits of suppliers will be performed to provide assurance that these requirements are satisfied. Because the applicant has indicated that natural uranium supplied to EREF will meet ASTM Standard C787-03, the staff finds that this is acceptable for ensuring that the Tc-99 and transuranic possession limits contained in Table 1.2-1 of the SAR are not exceeded. Thus, the quantities of Tc-99 and transuranic isotopes from residual contamination are expected to have no significant radiological impact.

In addition, Section 1.2.3 of the SAR (AES, 2010a) states that other source and byproduct materials will be used for instrument calibration purposes. The applicant intends to identify these materials during the design phase for the EREF and will submit future license amendment requests to incorporate the proposed quantities and types for the sealed and unsealed instrument calibration sources to its possession limits (AES, 2010a).

Table 1.2-1 Proposed Possession Limits

Source or Special Nuclear Material	Physical and Chemical Form	Maximum Amount to be Possessed at Any One Time
Uranium (natural and depleted) and daughter products	Physical Form: Solid, liquid and gas Chemical Form: UF_6, UF_4, UO_2F_2, oxides and other compounds	225,000,000 kg (496,000,000 lb)
Uranium enriched in isotope U-235 up to 5 percent by weight and uranium daughter products	Physical Form: Solid, liquid and gas Chemical Form: UF_6, UF_4, UO_2F_2, oxides and other compounds	1,750,000 kg (3,860,000 lb)
Tc-99, transuranic isotopes and other contamination	Any	Amount that exists as contamination as a consequence of the historical feed of recycled uranium at other facilities

Note: Tc-99 - technetium-99
UF_6 - uranium hexafluoride
UF_4- uranium tetrafluoride
UO_2F_2 - uranyl fluoride

As stated above, the applicant identified the elemental name, maximum quantity, and specifications, including the chemical and physical forms, of the licensed material that the applicant proposes to acquire, deliver, receive, possess, produce, use, transfer, or store. The applicant also identified the isotopic content and amount of enrichment by weight percent of the licensed material. The information provided by the applicant meets the guidance in Section 1.2.4.3 (3) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

1.2.4.1 Authorized Uses

The application is for the issuance of a license for the construction and operation of a gas centrifuge uranium enrichment facility in accordance with 10 CFR Parts 30, 40, and 70. The facility will employ centrifuge technology from ETC.
Section 1.2.4 of the SAR (AES, 2010a) lists the authorized uses of the SNM, source, and byproduct material. The AES proposes to use SNM and source material in the enrichment of uranium. The uranium enrichment services would be sold to clients for the production of low-enriched uranium that would ultimately be used in the manufacture of fuel for the production of electricity by commercial nuclear power plants. In Section 1.2.4 of the SAR, AES also proposes a 30-year license term (AES, 2010a).

Since the proposed facility will contain classified information, AES also requested approval of a classified matter facility clearance for the EREF under 10 CFR Part 95 and has submitted a "Security Plan for the Protection of Classified Matter" as part of the license application.

As stated above, the applicant provided a summary, non-technical narrative description for each activity or process in which the applicant proposed to acquire, deliver, receive, possess, produce, use, process, transfer, or store licensed material. The authorized uses of licensed material proposed for the facility are described in Section 1.2.4 of the SAR and are consistent with the Atomic Energy Act of 1954, as amended. The description is also consistent with more detailed process descriptions submitted as part of the ISA Summary (AES, 2010b), as reviewed under Chapter 3 of this SER. The information provided by the applicant meets the guidance in Section 1.2.4.3(4) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

1.2.4.2 Special Exemptions or Special Authorizations

In Section 1.2.5 of the SAR (AES, 2010a), the applicant requested an exemption from the regulations. In addition, in subsequent correspondence, the applicant requested a special authorization (AES, 2010g). The following sections describe the staff's evaluation of these requests.

1.2.4.2.1 Exemption to Provide Forward-Looking Incremental Funding for Decommissioning

In Section 1.2.5 of the SAR (AES, 2010a), the applicant requested an exemption from 10 CFR 40.36(d) and 10 CFR 70.25(e), "Financial Assurance and Recordkeeping for Decommissioning," in order to provide forward-looking incremental funding for decommissioning.

The NRC requirements for financial assurance are described in 10 CFR 40.36(d) and 10 CFR 70.25(e) and include a requirement that a licensee certify that financial assurance has been provided in the amount of the cost estimate for decommissioning. AES's cost estimate for decommissioning is based on a 30-year operating life and includes the estimated decommissioning costs of the facility and site and the expected costs associated with the disposition of depleted uranium tails. AES intends to sequentially install and operate modules of the enrichment equipment, thereby phasing in the enrichment capacity for the EREF and the resultant accumulation of depleted uranium tails. AES has requested the exemption in order to provide financial assurance for decommissioning during the operating life of the EREF at a rate that is proportional to the decommissioning liability for these facilities as they are phased in. Similarly, AES plans to provide financial assurance for the disposition of depleted uranium tails at a rate proportional to the amount of accumulated depleted uranium tails onsite up to the maximum amount of the depleted uranium tails produced by the EREF.

AES has requested the exemption on the basis that AES is committed to updating the decommissioning cost estimates and to providing to the NRC revised funding instruments for facility decommissioning prior to the operation of each facility module (AES, 2010a). AES also commits to updating the decommissioning cost estimates on an annual forward-looking incremental basis and to providing the NRC revised funding instruments that reflect these projections of depleted uranium tails production (AES, 2010a). AES expects to be able to accurately predict the production of depleted uranium tails based on long-term enrichment contracts. If any adjustments to the funding assurance were determined to be needed during the annual period due to production variations, AREVA would make them promptly and provide a revised funding instrument to the NRC. The requested exemption would allow AES to satisfy the applicable decommissioning funding assurance requirements for the EREF without imposing an unnecessary financial burden on itself.

Under the provisions of 10 CFR 40.14, and 70.17, both titled "Specific Exemptions," the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Parts 40 or 70, respectively, when the exemptions are authorized by law, will not endanger life or property or the common defense and security, and are otherwise in the interest of the public.

The NRC staff evaluated the exemption request. The proposed exemption is authorized by law because the Atomic Energy Act of 1954, as amended, contains no provisions prohibiting an applicant from providing decommissioning funding assurance on an incremental basis. Staff also determined that, because the incremental funding approach proposed by the applicant will provide funding for the all the applicant's decommissioning obligations at any point time, the approach will not endanger life or property or the common defense and security. Because the incremental funding approach will reduce the applicant's expenses from having to fund a 30-year decommissioning obligation when, in actuality, the decommissioning obligations prior to the end of the 30-year operating period are less, the staff has determined that the proposed approach will be in the public interest by reducing unnecessary regulatory costs.

The granting of this exemption meets the criteria for categorical exclusion under 10 CFR 51.22. The categorical exclusion described in 10 CFR 51.22(c)(25) addresses the granting of exemptions from regulatory requirements. Granting this exemption request meets the criteria described in 10 CFR 51.22(c)(25), namely, that there are no significant hazard considerations; no change in the types or amounts of effluents; no increase in public or occupational radiation exposure; no construction impact; no increase in the potential for or consequences of a radiological accident; and the exemption involves surety requirements.

Therefore, NRC staff grants the requested exemption. A license condition will be included in the license that will address AES's commitments for updating the decommissioning funding plan over time. This license condition is discussed further in Chapter 10 of this SER.

1.2.4.2.2 Authorization to Make Certain License Application Changes Without Prior NRC Approval

The applicant has requested authorization to make changes to license commitments, without prior NRC approval, that do not decrease the effectiveness of these commitments (AES, 2010g). The requested authorization has two parts covering (1) changes requiring prior approval and (2) changes not requiring prior approval. Changes to the license application that will require prior NRC approval are those that would decrease the effectiveness of any safety commitments (AES, 2010g). For a change that would decrease the effectiveness of a safety commitment, the applicant will submit the change to the NRC as a license amendment and will not implement the change until the NRC has reviewed and approved the amendment (AES, 2010g). Changes to the license application that could be made without prior NRC approval are those for which there is no degradation in the safety commitments in the license application and for which the change, test, or activity does not conflict with any condition specifically stated in the license (AES, 2010g). Records of such changes shall be maintained, including technical justification and management approval, in dedicated records to enable NRC inspection upon request at the facility (AES, 2010g). A report containing a description of each such change, and an appropriate revised section of the license application, will be submitted to the NRC within three months of implementing the change (AES, 2010g).

In evaluating the applicant's request for authorization, the staff considered the guidance in the Draft Regulatory Guide (RG), DG-3037, "Guidance for Fuel Cycle Facility Change Processes" (NRC, 2009). This draft RG states that a license condition can be applied to allow changes to the safety analysis report without prior NRC approval (NRC, 2009). The draft RG also recommends that a license condition allowing changes to the Safety Analysis Report without prior NRC approval should contain the following: (1) criteria; (2) a commitment to document the licensee's evaluation supporting any findings that preapproval is not required; and (3) the reporting frequency for providing changes to the NRC after implementation (NRC, 2009). The staff reviewed the applicant's authorization request and found it to be consistent with the guidance provided in the draft RG.

The staff will impose the following license condition for this authorization:

The licensee is hereby granted the special authorization as identified in Section 1.2.5 "Special Exemptions and Special Authorizations" of the Eagle Rock Enrichment Facility Safety Analysis Report:

a. *The licensee shall not make changes to the license application that decreases the effectiveness of safety commitments in the license application, without prior NRC approval. For these changes, the licensee shall submit to the NRC, for review and approval, an application to amend the license. Such changes shall not be implemented until approval is granted.*

b. *Upon documented completion of a change request for a facility or process, the licensee may make changes in the facility or process as presented in the license application, or conduct tests or activities not presented in the license application, without prior NRC approval, subject to the following conditions:*

    1. *There is no degradation in the safety commitments in the license application and*
    2. *The change, test, or activity does not conflict with any condition specifically stated in the license application.*

Records of such changes shall be maintained, including technical justification and management approval, in dedicated records to enable NRC inspection upon request at the facility. A report containing a description of each such change, and appropriate revised sections to the license application, shall be submitted to the NRC within three months of implementing the change.

1.2.4.3 Protection of Classified Matter

The purpose of this review is to verify that the application provided sufficient information to conclude that there is an adequate CMP for the protection of classified matter at the proposed EREF facility and that a facility clearance can be issued.

1.2.4.3.1 Regulatory Requirements

10 CFR Part 70.22(m) requires an application to contain a full description of an applicant's security program to protect against the unauthorized viewing of classified enrichment equipment and unauthorized disclosure of classified matter in accordance with the requirements of 10 CFR Part 95.

1.2.4.3.2 Regulatory Guidance and Acceptance Criteria

The EREF CMP was reviewed for compliance with the requirements of 10 CFR Part 95 by using "Standard Practice Procedures Plan Standard Format and Content for the Protection of Classified matter for NRC Licensee, Certificate Holder, or Other Activities as the Commission May Determine" (NRC, 2006).

1.2.4.3.3 Staff Review and Analysis

AES submitted its CMP entitled "Eagle Rock Enrichment Facility Standard Practice Procedure Plan" on December 30, 2008, and submitted Revision 1 on April 23, 2009 and Revision 2 on April 30, 2010. The applicant's CMP (AES, 2010d) outlines the facility's proposed security procedures and controls to ensure that classified matter is used, processed, stored, reproduced, transmitted, transported, and destroyed in accordance with the requirements of 10 CFR Part 95. Access to the site will be through the Entry Exit Control Point on the south end of the site.

1.2.4.3.4 Evaluation Findings

The NRC has reviewed the CMP for the EREF and considers it will satisfy the requirements of 10 CFR Part 95 when fully implemented. The CMP review generated a request for additional information (RAI), to which the licensee responded in September 2009 (AES, 2009a).

The applicant stated that access to classified information will be controlled in accordance with 10 CFR Part 25 and 10 CFR Part 95 and has provided an acceptable CMP that establishes controls to ensure that classified matter is used, processed, stored, reproduced, transmitted, transported, and destroyed only under conditions that will provide adequate protection and prevent access by unauthorized persons. The NRC will perform a readiness review of the site against the EREF CMP when facilities are in place prior to classified material being allowed onsite. By meeting these requirements, the applicant complies with the requirements of 10 CFR 70.22(m) to describe the security program to protect against unauthorized viewing of classified enrichment equipment, and unauthorized disclosure of classified matter.

However, NRC's authorization for the applicant to begin implementation of the CMP is contingent upon an NRC inspection and finding prior to receipt of classified matter that AES's classified matter program at EREF is being implemented in accordance with the CMP. The NRC staff will impose the following license condition to ensure that classified matter is not processed, stored, reproduced, transmitted, handled or accessed, except as permitted under 10 CFR Part 95:

> *The licensee shall not use, process, store, reproduce, transmit, handle, or allow access to classified matter except as provided by applicable personnel and facility clearances required under 10 CFR Part 95.*

In Section 2.0 of the CMP, AES also commits to following Nuclear Energy Institute guidelines, NEI 08-11, "Information Security Program Guidelines for Protection of Classified Material at Uranium Enrichment Facilities" (NEI, 2009). The programs addressed in NEI-08-11 represent new program elements to ensure that classified matter is used, processed, stored, reproduced, transmitted, transported, and destroyed only under conditions that will provide adequate protection and prevent access by unauthorized persons. By meeting these requirements, the applicant complies with the requirements of 10 CFR 70.22(m).

Because AES has not yet designated the areas where the use and handling of classified information will occur, NRC staff will impose the following license condition to ensure that areas used for handling classified information are properly protected:

> *Prior to designating areas where the use and handling of classified information will routinely occur, NRC will be notified to determine if additional security measures are required. If NRC does determine the need for additional security measures, an amendment request must be submitted, and approved, prior to establishment and use of the area(s).*

1.2.5 Evaluation Findings

The staff reviewed the institutional information for the proposed EREF, according to Section 1.2 of the Standard Review Plan (NRC, 2002). The applicant has adequately described and documented the corporate identity, structure, and financial information, and is in compliance with those parts of 10 CFR 30.32, 10 CFR 40.31, 10 CFR 70.22, and 10 CFR 70.65 related to institutional information.

The staff reviewed the information provided by the applicant on liability insurance. This information meets the requirements of 10 CFR 140.13b. Because full liability insurance coverage will not be provided until prior to receipt of licensed material, NRC staff will impose the license condition provided in Section 1.2.3.3.3 of this SER:

> *The licensee shall provide proof of full liability insurance of $300 million as required under 10 CFR 140.13(b), at least 30 days prior to the planned date for obtaining licensed material.*

In addition, in accordance with 10 CFR 30.32, 10 CFR 40.31, and 10 CFR 70.22(a)(2) and (4), the applicant has adequately described the types, forms, and quantities and proposed purpose and authorized uses of licensed materials to be permitted at the facility. The applicant provided information on one exemption request related to decommissioning funding, which meets the requirements of 10 CFR 40.14 and 10 CFR 70.17.

The applicant has also adequately described information related to FOCI, 10 CFR 40.38 and 10 CFR 70.40, and its plans to secure classified matter for a facility clearance under 10 CFR Part 95. The staff reviewed the applicant's Standard Practice Procedure Plan and found it to satisfy the requirements of 10 CFR Part 95. Because a specific facility for use and storage of classified matter has not been identified, staff will impose the following license condition, provided in Section 1.2.4.3.4 of this SER:

> *The licensee shall not use, process, store, reproduce, transmit, handle, or allow access to classified matter except as provided by applicable personnel and facility clearances required under 10 CFR Part 95.*

Because the areas where the use and handling of classified information will occur have not been identified, staff will impose a license condition provided in Section 1.2.4.3.4 of this SER:

> *Prior to designating areas where the use and handling of classified information will routinely occur, NRC will be notified to determine if additional security measures are required. If NRC does determine the need for additional security measures, an amendment request must be submitted, and approved, prior to establishment and use of the area(s).*

The staff also reviewed and approved the applicant's special authorization request to make certain license application changes without prior NRC approval, as described in Section 1.2.4.2.2 of this SER, and will impose the license condition stated in Section 1.2.4.2.2.

The staff concludes that the applicant has met the requirements and acceptance criteria in Section 1.2.4.3 of NUREG-1520 (NRC, 2002).

1.3 Site Description

The purpose of NRC's review of the applicant's site description is to evaluate whether the application adequately describes the geographic, demographic, meteorological, hydrologic, geologic, and seismologic characteristics of the site and the surrounding area. The site description is a summary of the information that the applicant used in preparing the Environmental Report (ER), the Emergency Plan (EP), and the ISA Summary.

1.3.1 Regulatory Requirements

The regulations in 10 CFR 30.33, 10 CFR 40.32, 10 CFR 70.22, and 10 CFR 70.65(b)(1) require each application to include a general description of the site, with emphasis on those factors that could affect safety (i.e., nearby facilities, meteorology, and seismology). In addition, 10 CFR 70.61(f) requires each licensee to establish a controlled area, as defined in 10 CFR 20.1003,[2] and to retain authority to exclude or remove personnel and property from the area.

1.3.2 Regulatory Guidance and Acceptance Criteria

The acceptance criteria applicable to NRC's review of the site description section of the SAR (AES, 2010a) are contained in 10 CFR 30.33, 10 CFR 40.32, 10 CFR 70.22, 10 CFR 70.65(b)(1), and Section 1.3.4.3 of NUREG-1520 (NRC, 2002). Chapter 1 of NUREG-1520 is applicable to the EREF in its entirety.

1.3.3 Staff Review and Analysis

1.3.3.1 Site Geography

Section 1.3 of the SAR (AES, 2010a) describes the EREF's location and nearby roadways and bodies of water, and significant geographical features.

The proposed site is located in southeastern Idaho, in Bonneville County, near the border with Bingham County, and about 113 km (70 mi) west of the Idaho/Wyoming State line. The city of Idaho Falls is located about 32 km (20 mi) east southeast from the site. The towns of Rigby and Rexburg are located approximately 23 km (14 mi) and 42 km (26 mi) north of Idaho Falls, respectively. Atomic City is about 32 km (20 mi) west of the site. The towns of Blackfoot and Pocatello are located approximately 40 km (25 mi) and 76 km (47) miles south of the site, respectively. The Fort Hall Indian Reservation comprises about 220,150 ha (544,000 ac) and also lies to the south of the site. The nearest boundary of the reservation to the site is about 44 km (27 mi). The town of Fort Hall is about 60 km (37 mi) from the site.

The site is described as native rangeland, non-irrigated seeded pasture, and irrigated cropland. The lands north, east and south of the site are a mixture of private-, State-, and Federal-owned lands (AES, 2010a).

The proposed site is located in the east central part of the East Snake River Plain. The proposed site is relatively flat with a gentle sloping surface with small ridges and areas of rock outcrop. Most of the site is semi-arid steppe. Elevations at the site range from 1,556 m (5,106 feet [ft]) to 1,600 m (5,250 ft).

The proposed EREF site contains no surface water bodies (AES, 2010b). The closest surface water bodies are the Snake River and the Market Lake Wildlife Management Area. These two surface water bodies are located about 32 km (20 mi) to the east and northeast of the site, respectively (AES, 2010b).

The EREF will be located on the north side of U.S. Highway 20, a two-lane highway. A dirt road currently provides site access from U.S. Highway 20, while other dirt roads provide access

[2] A controlled area is defined in 10 CFR 20.1003 as an area, outside of a restricted area, but inside the site boundary, access to which can be limited by the licensee for any reason.

throughout the proposed site. To the east, U.S. Highway 20 intersects with Interstate 15 on the west side of Idaho Falls, Idaho. To the west, U.S. Highway 20 intersects with U.S. Highway 26 northwest of Atomic City and ultimately intersects with Interstate 84 outside the town of Mountain Home, Idaho, southeast of Boise.

There are no gas pipelines (industrial gases, natural gas, etc.) located on or nearby the proposed facility site (AES, 2010b).

AES's SAR describes the main land uses nearby the EREF. Grazing and cropping are the main land uses within 8 km (5mi) of the site (AES, 2010a). State land immediately west of the proposed site and U.S. Bureau of Land Management land immediately east of the site are grazed. Section 1.3.2.5 of the SAR provides additional information about nearby land uses (AES, 2010a).

The SAR also includes information on nearby industrial facilities (AES, 2010a). The nearest is the DOE's Idaho National Laboratory (INL). Its eastern boundary is 1.6 km (1 mi) west of the proposed site. The INL property near the site is undeveloped rangeland. The closest facility on the INL property is the Materials and Fuels Complex, located about 16 km (10 mi) west of the EREF. In addition, there are landfills in Jefferson, Bonneville, and Bingham counties and two waste transfer stations in Bonneville County. The nearest commercial carrier airport is Fanning Field (Idaho Falls Regional Airport) in Idaho Falls about 32 km (20 mi) from the site. Pocatello Regional Airport is located in Pocatello, about 113 km (70 mi) south of the site.

Information on public facilities, such as schools and hospitals, is also provided in the SAR (AES, 2010a). Most of the public facilities are located in Idaho Falls, including hospitals, nursing homes, schools and churches.

There are four fire departments within about a 48-km (30-mi) radius of the site; the Idaho Falls Fire Department, the Ucon Volunteer Fire Department, the Shelley Firth Rural Fire Department, and the Central Fire District which operates in Jefferson County. Fire support service for Idaho Falls is provided by the Idaho Falls Fire Department, located approximately 32 km (20 mi) from the EREF.

Information on rail lines is provided in Section 1.3.1.2 of the SAR (AES, 2010a). The nearest rail lines include:

- The Union Pacific Railroad Aberdeen Branch—about 40 km (25 mi) south of the EREF
- The Union Pacific Railroad Scoville Branch—leading onto the INL
- The Eastern Idaho Rail Road operates short line tracks connecting towns north and east of Idaho Falls to the Union Pacific Line

The site property boundary and the controlled area boundary for the EREF are shown on Figure 1.1-3, "Site Plan with Property and Control Area Boundary" of the SAR (AES, 2010a).[3]

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[3] Drawings of the site plan and facility layout have been withheld from public release as security-related information under 10 CFR 2.390.

1.3.3.1.1 Site Geography Conclusions

AES provided a summary describing the site geography, including its location relative to prominent natural and manmade features (such as rivers, airports, population centers, schools, and commercial and manufacturing facilities). The summary also described the site boundary and controlled area boundary. The NRC staff reviewed information provided in the SAR (AES, 2010a) and ISA Summary (AES, 2010a) onsite geography and finds the data used to be accurate. The applicant's descriptions are consistent with the more detailed information in the ISA Summary (AES, 2010b), the ER (AES, 2010c), and the EP (AES, 2010f). The information the applicant provided is consistent with the guidance in Sections 1.3.4.3(1) and 1.3.4.3(5) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

1.3.3.2 Demographics

The proposed site is located in Bonneville County, Idaho. Portions of Bonneville, Jefferson, and Bingham counties are within 8 km (5 mi) of the site. The combined population of Bonneville, Bingham, and Jefferson counties in the EREF vicinity is 143,412, based on the 2000 U.S. Census (AES, 2010a). This population represents an average annual increase of 1.4% from the 1990 population, less than that for the state of Idaho during the same period. It is expected that the population growth in these three counties in the next 30 years will be at a lower rate than the rate for the state of Idaho (AES, 2010a).

Major population centers near the proposed site include:

- Idaho Falls, Idaho, about 32 km (20 mi) east southeast of the site
- Shelley, Idaho, about 45 km (28 mi) southeast of the site
- Blackfoot, Idaho, about 77 km (48 mi) southeast of the site
- Pocatello, Idaho, about 113 km (70 mi) south of the site
- Rexburg, Idaho, about 82 km (51 mi) northeast of the site
- St. Anthony, about 101 km (63 mi) northeast of the site

Aside from these communities, the population density is generally low. There are no residences, schools, stores or other population centers within a 1.6 km (1 mi) radius of the site. The nearest residence is 7.7 km (4.8 mi) east of the proposed site boundary (AES, 2010b).

The three hospitals in Bonneville County are located in Idaho Falls. The Eastern Idaho Regional Medical Center is the largest of these. It is a short-term, acute care hospital with 242 beds. The other two hospitals are the Idaho Falls Recovery Center, a 7-bed, acute care facility; and the Mountain View Hospital, a 20-bed, acute care facility. The closest nursing homes or retirement facilities, schools, and churches are located in Idaho Falls.

Public use areas include a hiking trail south of the proposed site in Hell's Half Acre Wilderness Study Area and a small lava tube cave located approximately 8 km (5 mi) east and south.

The applicant provided a summary of demographic information based on the most recent census data that showed the population distribution as a function of distance from the proposed facility. The applicant's descriptions are consistent with the more detailed information in the ISA

Summary (AES, 2010b), the ER (AES, 2010c), and the EP (AES, 2010f). The information the applicant provided is consistent with the guidance in Sections 1.3.4.3(2) and 1.3.4.3(5) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

1.3.3.3 Meteorology

Section 1.3.3 of the SAR (AES, 2010a) and Section 3.2.3 of the ISA Summary (AES, 2010b) provide a meteorological description of the site and its surrounding area.

1.3.3.3.1 Tornado Hazard

Forty tornadoes were recorded from January 1, 1950, through April 30, 2008 (58 years of data), in the four-county (Bonneville, Bingham, Butte, and Jefferson) region (AES, 2009a,b). Among them, 1 is an F2 (Fujita Scale) tornado, 19 are F1 tornadoes, and the remaining are F0 tornadoes. Based on these data, the applicant calculated the annual probability for a tornado to strike in any 2,589-km^2 (1,000-mi^2) area in the vicinity of the EREF site to be 0.09 years (AES, 2010b). The applicant concluded that the probability of a tornado to strike the EREF is small. The NRC staff verified that the applicant's assessment of the probability of a tornado strike at the site as small (approximately 1.0×10^{-5} to 1.0×10^{-4}/year) and, therefore, a tornado strike is considered unlikely based on the applicant's likelihood definition. Thus, the applicant assessed potential effects of tornado hazard to the EREF. NUREG/CR-4461 (NRC, 2006b) provides recommended tornado design wind speed maps for the continental US. The applicant used Figure 8-1 from NUREG/CR-4461 (NRC, 2006b) and determined that the recommended tornado design wind speed, associated with a 1.0×10^{-5}/year tornado at the EREF, to be 0 km/hour (0 mi/hour). The NRC staff verified that this value, presented in NUREG/CR-4461, is obtained by smoothing out the spatial variation in tornado wind speeds for a given probability level. Because tornadoes cover a finite discrete area for each event, a wind speed of zero indicates that the probability of any part of a tornado wind field impacting the site is less than $1.0 \times 10^{-5}\ yr^{-1}$. Staff note that the non-smoothed value of maximum tornado wind speed for a probability level of $1.0 \times 10^{-5}\ yr^{-1}$ is 82 mph, however, this is less than the design extreme wind of 105 mph and does not need to be considered as a separate design bases. Consequently, no special consideration of tornado hazard is necessary because of the estimated zero tornado design wind speed. The NRC staff concludes that the applicant used appropriate meteorological data to assess the probability of a tornado strike and has appropriately concluded that no special consideration of tornado hazard is necessary in the design basis for the EREF, thereby meeting the regulatory requirements in 10 CFR 70.65(b)(1) to provide a general description of the site with emphasis on those factors that could affect safety.

1.3.3.3.2 High Winds

The applicant characterized the high-wind hazard at the proposed EREF site using the high-wind hazard probability relationship reported in "Natural Phenomena Hazards Modeling Project: Extreme Wind/Tornado Hazard Models for Department of Energy Sites," (Coats, 1985) for the INL, whose eastern boundary is approximately 1.6 km (1 mi) west of the proposed site (AES, 2010b). The NRC staff found that the applicant's assessment that the high-wind hazard probability relationship for the INL is equally applicable to the proposed site, given the close proximity of the proposed site to INL, is reasonable. The NRC staff also found that the high-wind hazard recommended in "Natural Phenomena Hazards Modeling Project: Extreme Wind/Tornado Hazard Models for Department of Energy Sites," (Coats, 1985) is acceptable because this high-wind hazard was developed using an acceptable methodology and was reviewed and commented on by experts from organizations, including the National Oceanic and Atmospheric Administration, the NRC, and the U.S. Geologic Survey (USGS). According to "Natural Phenomena Hazards Modeling Project: Extreme Wind/Tornado Hazard Models for Department of Energy Sites" (Coats, 1985), the 10-m (33-ft) aboveground fastest mile wind speed with an annual probability of 1.0×10^{-5} is 169 km/hr (105 mi/hr).

Because the proposed EREF is not located near a coastal area (approximately 925 km [575 mi] from the coast), hurricanes affecting the coastal area will have no effect on the performance of the facility. Consequently, consideration of hurricane wind hazards on the design of the proposed EREF is not needed; and the staff conclude that the regulatory requirements in 10 CFR 70.65(b)(1) have been met with regard to this topic.

1.3.3.3.3 Extreme Precipitation

The applicant discusses extreme precipitation at the proposed EREF site in ISA Summary Section 3.2.3.4.4 (AES, 2010b). The applicant estimated the extreme precipitations for 1-, 24-, and 48-hour duration rainfalls corresponding to annual probability of 1.0×10^{-5} using the rainfall data extracted from the "Precipitation Frequency Atlas of the Western United States" (NOAA, 1973); the "Two-To-Ten-Day Precipitation for Return Periods of 2 to 100 Years in the Contiguous United States (Commerce, 1964); and the National Weather Service's least square regression procedure. The estimated 1-, 24-, and 48-hour, all-season, extreme local precipitations for the annual probability of 1.0×10^{-5} are 60 mm (2.37 in), 112 mm (4.39 in), and 135 mm (5.31 in), respectively (AES, 2010b).

The NRC staff reviewed the analysis supporting the applicant's estimates of extreme precipitations during an onsite review and finds that this analysis is acceptable because it used rainfall data from reliable sources; and the least square regression procedure the applicant used to extrapolate the extreme precipitations is an industry-accepted, appropriate method for rainfall hazard assessment. The NRC staff concludes that the applicant used appropriate meteorological data and an industry-accepted methodology to estimate the extreme precipitations, thereby meeting the regulatory requirements in 10 CFR 70.65(b)(1).

1.3.3.3.4 Flood

ISA Summary Section 3.2.4.3 discusses potential for flooding at the EREF (AES, 2010b). The applicant indicated that the proposed EREF is not located near any large body of water that could cause a flood at the facility site. The nearest large surface waters are the Snake River, which is approximately 32 km (20 mi) east, and Lake Wolcott, which is approximately 120 km (75 mi) southwest of the site. Therefore, the applicant concluded that no credible flooding can occur at the site resulting from either existing river sources or failure of upstream dams. The NRC staff finds the applicant's conclusion, that flooding caused by overflow of surface rivers and failure of upstream dams is not credible, is acceptable. The staff agrees that this type of flooding hazard is highly unlikely.

The applicant indicated that local precipitation-induced roof-ponding would be limited through roof design. Consequently, roof-ponding hazard resulting from local intense precipitation is highly unlikely (AES, 2010c). The NRC staff finds the applicant's conclusion acceptable because, with appropriate roof design, the applicant will be able to limit accumulation of extreme local precipitation on the roof to ensure that the resulting roof-ponding load does not exceed the roof design load.

Regarding the floodwater intrusion potential because of extreme local precipitation, the applicant indicated that the site of the proposed facility is located at a localized topographical high ground (AES, 2010c). In addition, the ground floor levels of the proposed facility safety-related structures will be 0.15 m (6 in) above the adjacent finished grade. Based on these two reasons, the applicant concluded that intrusion of floodwater into safety-related structures is highly unlikely because the ground floor level is higher than the flood depths corresponding to the 1- and 24-hour storm deluges corresponding to an annual probability of 1.0×10^{-5}. The NRC staff reviewed the information and concludes it is acceptable.

Additionally, the applicant's intent to make the finished grade slope away from buildings will further prevent precipitation accumulations against the structures (AES, 2010c) and, therefore, reduce floodwater intrusion potential.

In summary, the NRC staff concludes that flood hazards resulting from surface river overflows, *upstream dam failures, and extreme local precipitation are highly unlikely because no large* bodies of water are located near the proposed site. The NRC staff also concludes that because the ground floor levels of the safety-related structures are above the potential flood levels for the storm deluges corresponding to an annual probability of 1.0×10^{-5}, flood hazard resulting from extreme local precipitation is not a safety concern to the proposed facility. Consequently, the NRC staff concludes that the regulatory requirements in 10 CFR 70.65(b)(1) have been met with regard to this topic.

1.3.3.3.5 Snow

In Section 1.3.3.2 of the SAR, the applicant accounts for the extreme environmental snow hazard through design considerations so that the potential events associated with this hazard are highly unlikely (annual probability smaller than or equal to 1.0×10^{-5}) (AES, 2010a).

The design basis extreme environmental snow load for the proposed EREF was the sum of the 50-year snowpack load and the load corresponding to the 48-hour winter season extreme precipitation with an annual probability of 1.0×10^{-5} for the area (AES, 2009a, b).

The applicant used the data that the National Resources Conservation Service collected from two locations near the proposed site to estimate the 50-year snowpack load (216.0 kg/m^2 [44.2 lb/ft^2]) (AES, 2009a). The applicant adopted the generalized extreme value method to determine frequency distributions and calculate the snowfall depth corresponding to the 50-year return period.

The applicant estimated the 48-hour winter season extreme precipitation with an annual probability of 1.0×10^{-5} for the EREF site to be 94.0 mm (3.70 in), corresponding to a load of 92.8 kg/m^2 (19.0 lb/ft^2). As a result, the design basis extreme environmental snow load is 309.0 kg/m^2 (63.2 lb/ft^2).

The NRC staff reviewed the analysis the applicant performed for determining design basis extreme environmental snow load and finds that the analysis is acceptable because: (i) the snowfall data used are from a reliable source and (ii) the methods used to estimate the 50-year return snowpack load and the 48-hour winter season extreme precipitation with annual probability of 1.0×10^{-5} are the industry-accepted methods. Thus, the NRC staff concludes that the regulatory requirements in 10 CFR 70.65(b)(1) have been met with regard to this topic.

1.3.3.3.6 Lightning

According to the applicant's ISA Summary (AES, 2010b), nine lightning strikes from January 1, 1950, to May 31, 2008, were recorded in the National Climate Data Center Storm Event Database for the four-county region where the EREF site will be located. The applicant estimated the lightning strike frequency for the proposed EREF site to be 0.75 flashes per year using the methodology proposed by considering the attractive area of structures (Marshall, 1973). Based on the lightning strike frequency estimate, lightning strike hazard is likely at the proposed site based on the applicant's likelihood definition. The applicant accounts for the lightning hazard by providing lightning protection to the EREF as described in Section 7.3.7 of the SAR (AES, 2010a). In Section 7.3.7, the applicant states that lightning protection will be provided in accordance with National Fire Protection Association (NFPA) 780 "Standard for the Installation of Lightning Protection Systems." In Section 7.3 of this SER, the staff finds the use of NFPA standards to be in accordance with the guidance of Section 7.4.3 of NUREG-1520 (NRC, 2002) in regard to nationally recognized codes and standards that may be used. The NRC staff finds that the applicant adequately considered lightning strike hazard through lightning protection design. Consequently, the NRC staff concludes that the regulatory requirements in 10 CFR 70.65(b)(1) have been met with regard to this topic. Meteorology Conclusions

As discussed in Sections 1.3.3.3.1 through 1.3.3.3.6 of this SER, the applicant provided appropriate meteorological data, including a summary of design-basis values for accident analysis of maximum snow loads and probable maximum precipitation, as presented in the ISA Summary (AES, 2010b). The applicant also provided appropriate design-basis information for lightning, high winds, tornadoes, hurricanes, extreme precipitation, and temperature extremes. The applicant's descriptions are consistent with the more detailed information in the ISA Summary (AES, 2010b), the ER (AES, 2010c), and the EP (AES, 2010f). The information the applicant provided is consistent with the guidance in Sections 1.3.4.3(3) and 1.3.4.3(5) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

1.3.3.4 Geology

1.3.3.4.1 Seismic Hazards

The applicant provided information on seismic hazards in Section 3.3.7 of the ER (AES, 2010c), Section 1.3.5 of the SAR (AES, 2010a), and Section 3.2.6 of the ISA Summary (AES, 2010b). In addition, the applicant provided additional details about the results of its seismic hazard assessment in response to staff's request for additional information (AES, 2009a).

Specific areas of the seismic hazard analysis, that are applicable to the seismic design of the proposed EREF, were reviewed:

Seismic source characterization

- ground motion attenuation models
- probabilistic seismic hazard assessment (PSHA)
- development of site-specific ground motion spectra

1.3.3.4.1.1 Seismic Source Characterization

Geological and Tectonic Settings

As the applicant discussed in the SAR and supporting documents (AES, 2009a, b, h, and g), the proposed EREF site is located near the center of the Eastern Snake River Plain in Southeastern Idaho. The Snake River Plain is a topographically subdued physiographic province that has been structurally and volcanically active over the last 17 million years. The geology of the Snake River Plain and resulting volcanic landscape was produced by the westward and southwestward drift of the North American tectonic plate over the Yellowstone mantle hotspot. The hotspot is now located beneath Yellowstone National Park.
The Snake River physiographic province is bordered on the northwest, west, south, and southeast by the Basin and Range; on the northeast by the Yellowstone Plateau; and on the north by Idaho Batholith provinces. These four physiographic provinces also correspond to unique seismotectonic settings, each with the potential to generate earthquakes that could impact the facilities at the proposed site. Tectonic activity in the surrounding provinces has been ongoing for several hundred million years, reflecting the active tectonic evolution of western North America. Over the last 60 million years, extensional tectonics of western North America developed the classic Basin and Range topography, manifest as north-south or northwest-southeast elongate and internally drained basins situated between fault-bound mountain ranges of exhumed basement rock. Most of the faulting takes place along normal faults at the interface between the basins and the ranges. Normal faulting along basin-bounding faults in the Basin and Range is the main initiator of large-magnitude earthquakes in the region.

The applicant used the USGS database to develop a list of Quaternary fault sources. The USGS database includes all known faults that have produced a magnitude 6.0 or larger earthquake in the last 1.6 million years (Petersen, 2008). These include the Lost River fault, which produced the 1983 Borah Peak earthquake; the Lemhi fault, which bounds the southwestern face of the Lemhi Range; and the Beaverhead fault, which bounds the southwestern side of the Beaverhead Mountains. The full list of seismic sources the applicant used is documented in ISA Summary Appendix F, Table 4 (AES, 2010b).

Historic Seismicity

Figure 6 in the ISA Summary, Appendix F (AES, 2010b), shows earthquake epicenters of the Snake River Plain and surrounding areas based on earthquake records from the collection of regional and national earthquake catalogs (USGS cited Peterson, 2008). As shown in the figure, few earthquakes occurred within the Snake River Plain in contrast to the relatively large number of earthquakes in the surrounding regions. Many geologists have attributed the lack of seismicity in the Snake River Plain, compared to the surrounding provinces to continued high heat flux of the Snake River Plain crust, such that a large fraction of the regional tensional tectonic stress is accommodated by intrusion of magmatic dikes and crustal ductility (e.g., Parsons, 1998; Parsons, 1991). In contrast, tectonic stresses in the crust of the surrounding provinces accumulate until the rocks are overstressed and ruptured, thus producing an earthquake. The greater ductility of the Snake River Plain crust, combined with dike intrusions, moderates the tectonic stress before it can reach levels sufficient to produce earthquakes.

For the PSHA, the applicant developed a catalog of historic earthquakes from regional and national databases, including the Advanced National Seismic System catalog and the USGS regional catalog. In the applicant's composite catalog, all earthquake magnitudes were converted to moment magnitudes based on the equation of Johnston (Johnston, 1996). The applicant's composite catalog identifies 66 earthquakes within approximately 322 km (200 mi) of the site with a moment magnitude of 5.0 or larger. The applicant identified the 1959 Hebgen Lake earthquake (moment magnitude [Mw[= 7.3, surface wave magnitude [Ms[= 7.5) and the 1983 Borah Peak earthquake (Mw = 6.8, Ms = 7.3) as the most significant historic earthquakes in the region. The 1959 Hebgen Lake earthquake was the largest historical earthquake in the intermountain region. The 1983 Borah Peak earthquake was the largest earthquake ever recorded in Idaho. Also included in the applicant's list is the 1905 Shoshone earthquake, which is important because it is one of the few earthquakes that may have occurred within the Snake River Plain. Although, because it took place before seismic instrumentation, the location of its epicenter is highly uncertain.

Seismic Sources

The applicant developed a list of fault sources as inputs to the PSHA based on the location of known Quaternary faults in the region, as defined in the USGS database. The applicant derived regional seismic source zones from interpretations of the historic seismicity. Seismicity parameters, including activity rates, recurrence, and maximum magnitude for each of the fault sources were adopted from the USGS database. Seismicity parameters for the regional source zones were derived from the earthquake catalog information. To account for uncertainty in the regional seismic source zones, especially the distributions of spatial densities of past earthquakes, the applicant developed four alternative source models as inputs to the PSHA. These alternative source models, which include both regional seismic source zones and the fault sources, were used to develop four initial branches of the PSHA logic tree.

Staff Evaluation of Seismic Source Characterization

The NRC staff reviewed the information in ISA Summary, Appendix F, and finds that the applicant properly identified and characterized the seismic sources that could impact the site. The applicant's seismic source characterization is also consistent with other recent NRC licensing actions, including the Idaho Spent Fuel Facility at the nearby INL reservation

(NRC, 2004). Based on an evaluation of the applicant's database of historical seismicity, the NRC staff also finds that the applicant developed an adequate set of historical seismic data used to develop seismic source parameters. Based on this review, the NRC staff concludes, that the information concerning seismic source characterization presented in the SAR (AES, 2010a), ISA Summary (AES, 2010b), and Appendix F to the ISA Summary is acceptable. The information adequately summarized seismicity and potential fault and tectonic sources to meet the regulatory requirements in 10 CFR 70.65(b)(1).

1.3.3.4.1.2 Ground Motion Attenuation

The applicant used the SEA99 (Spudich, 1999) and the B&A08 (Boore, 2008) ground motion attenuation models in its PSHA. The applicant selected these models because they included specific predictions for normal faults. Additionally, the models were based on a large set of strong motion recordings and were applicable to bedrock conditions with shear wave velocities greater than 620 m/s (2,034 feet per second [ft/s]) in the upper 30 m (98 ft) of the ground surface (VS30). The applicant stated that these high shear wave velocities are consistent with conditions present at the site. The site has basalt as bedrock with a thin veneer of alluvial sediments no more than 4.3 m (14 ft) thick. Geophysical measurements from Payne (Payne, 2002) made at the INL indicate that the basaltic bedrock has shear wave velocities in the range of 1,200–1,500 m/s (4,000–5,000 ft/s). In the PSHA, the applicant developed seismic hazard results for SEA99— assuming VS30 = 620 m/s (2,034 ft/s), B&A08 assuming VS30 = 760 m/s (2,493 ft/s), and B&A08 assuming VS30 = 1,300 m/s (4,265 ft/s). The applicant will conduct additional geophysical investigation to verify the VS30 values (AES, 2009a).

Staff Evaluation of Ground Motion Attenuation Models

The NRC staff reviewed the information in the ISA Summary, Appendix F (AES, 2010b), and finds that the ground motion modeling approach the applicant used reasonably predicts the earthquake-induced ground motions at the proposed EREF site. Both the SEA99 and B&A08 models are well-established attenuation models that were derived from a large dataset of western United States strong motion records. Use of these models for bedrock conditions with VS30 values greater than 620 m/s (2,034 ft/s) is also appropriate to the geology of the site because of similarity of the bedrock conditions to those at the INL. Based on this review and the applicant's commitment to verify the VS30 values, the NRC staff concludes that the information concerning ground motion attenuation models presented in the SAR (AES, 2010a); the ISA Summary (AES,2010b); and Appendix F to the ISA Summary is acceptable. The information provides an adequate approach to model ground motion attenuation to meet the regulatory requirements in 10 CFR 70.65(b)(1).

1.3.3.4.1.3 Probabilistic Seismic Hazard Assessment Calculation (PSHA) Results

The applicant developed a PSHA based on a logic tree formulation using the four seismic source models and two ground motion attenuation models described in the previous sections of this review. This results in eight logic tree nodes. The applicant gave each of these nodes equal weight of 0.125. The B&A08 attenuation model nodes were then subdivided based on the two possible VS30 values, and these subnodes were given weights of 0.0625. For the source parameters, the applicant developed single activity rates for each source and single estimates of maximum magnitude without additional variations in these parameters.

Results of the PSHA were the probabilities of exceeding the peak ground acceleration and a set of spectral accelerations at 5-percent damping for ground motion periods ranging from 0.01 to 10.0 seconds. Results of the hazard analysis indicate that the peak horizontal accelerations at an annual exceedance frequency of 1×10^{-3} would be 0.063 g (61.8 cm/s^2), increasing to 0.30 g (294 cm/s^2) at an annual exceedance frequency of 1×10^{-5}. Spectral acceleration at 5 Hz (with 5 percent damping) increases from 0.16 g (161.15 cm/s^2) at an annual exceedance frequency of 1×10^{-3} to 0.76 g (743.50 cm/s^2) at an annual exceedance frequency of 1×10^{-5}. The applicant also showed that seismic activity close to the site and within the Eastern Snake River Plain or from the source zone at Yellowstone contributes the most to the total hazard. The applicant noted that the major Basin and Range faults are too far away from the site to produce significant ground motion, given the high rate of ground motion attenuation predicted in both the SEA99 and B&A08 attenuation models for normal fault earthquakes.

Comparison to Other PSHA Results

The results of the applicant's PSHA are up to 40 percent lower than hazard results indicated by the 2008 USGS National Seismic Maps indicated (Peterson, 2008). In ISA Summary Appendix F (AES, 2010b) and supplemented by the supporting calculations in its response to NRC staff's RAI (AES, 2009a), the applicant cited three reasons why it concludes that USGS results are too high.

- The USGS hazard maps are based on an assumption of equal contributions to the hazard from both strike-slip and normal faults, whereas the applicant's model only considered normal faults. The attenuation models the USGS National Seismic Hazard Maps used predict higher amplitudes of ground motions for a strike-slip earthquake compared to a similar-sized earthquake on a normal fault.

- The USGS models are based on an assumed site condition with VS30 = 760 m/s (2,493 ft/s), whereas the applicant's PSHA includes 75 percent of the cases with a VS30 = 760 m/s (2,493 ft/s) and the remaining 25 percent of the cases with

  VS30 = 1,300 m/s (4,265 ft/s). Given the same earthquake, sites with higher VS30 values (e.g., harder rock conditions) generally experience smaller ground motion amplitudes than sites with smaller VS30 values.

- The method the USGS used to develop seismicity parameters for the seismic Eastern Snake River Plain seismic source region leads to a conservative estimation of seismic activity in this zone. For example, the USGS approach estimates 11 to 14 earthquakes with moment magnitudes greater than or equal to 4.0 should have occurred in the region in the last 45 years. The applicant's site-specific model predicts only about six earthquakes with moment magnitudes greater than or equal to 4.0 over the same period. In comparison, the historic record shows that the region has only experienced two earthquakes with moment magnitudes greater than or equal to 4.0 since 1963.

The results of the applicant's PSHA also differ from those developed for facilities at the INL (e.g., the Idaho Nuclear Technology and Engineering Center (INTEC) PSHA given in Payne, 2002). Although the INL hazards also tend toward greater ground motion hazards than the applicant's PSHA, the main difference is in the shape of the response spectra. The INTEC uniform hazard spectra are much flatter than those the applicant developed. The applicant considered these differences from the relative proximity of the INL sites to the Basin and Range normal faults.

Staff Evaluation of Probabilistic Seismic Hazard Assessment Calculation

The NRC staff reviewed the information presented in the ISA Summary Appendix F and the response to NRC staff's request for additional information and find that applicant's PSHA adequately predicts the earthquake-induced ground motions at the proposed EREF site. The NRC staff finds that the applicant developed its PSHA based on available seismic and geologic information, supported by reasonable assumptions about the future seismic activity in the region. The rationale explaining differences from other existing PSHA results, especially those from the USGS National Seismic Hazard Map, are well founded and clearly reasoned. The staff agrees that normal faulting is the most likely style of faulting, given the present tectonic setting of the region. The staff also agrees that the USGS models are applicable to softer site conditions, and they overestimate seismicity in the source zones. Based on this review, the NRC staff concludes, that the PSHA presented in the SAR, the ISA Summary, and the ISA Summary Appendix F is acceptable. The information provided an adequate approach to develop seismic inputs for design and performance consideration in the application and thereby meets the regulatory requirements in 10 CFR 70.65(b)(1).

1.3.3.4.1.4 Development of Site-Specific Ground Motion Spectra

Based on the PSHA results, the applicant developed the horizontal motion Uniform Hazard Response Spectra (UHRS) for selected annual exceedance frequencies between 2.1×10^{-3} and 1.0×10^{-5}, which correspond to return periods between approximately 475 and 100,000 years. Based on an in-office review of the applicant's calculations, for seismic design inputs, the applicant developed UHRS for 4×10^{-4}, 1×10^{-4}, and 1×10^{-5} for damping values of 5, 7, and 10 percent. These UHRSs were used to adjust the design response spectra to meet their target performance objectives per the guidance in American Society of Civil Engineers (ASCE) 43–05 (ASCE, 2005).

The applicant also developed vertical response spectra as two-thirds that of the horizontal spectra at all periods based on Bozorgnia and Campbell (Bozorgnia, 2004). Bozorgnia and Campbell showed that a vertical-to-horizontal ratio of two-thirds is appropriate for site conditions that include hard bedrock and a site-to-source distance of 60 km (37.3 mi) or more. The applicant noted that the proposed EREF site meets these conditions because it is situated above bedrock with high VS30 values and seismic sources that are outside the Eastern Snake River Plain province and thus more than 60 km (37.3 mi) distant. The applicant will develop acceleration time histories necessary for some engineering analyses to support design of the facilities (AES, 2009a). The applicant will use stochastic methods to calculate acceleration time histories with response spectra matching the design response spectrum at all frequencies. In addition, the applicant will further verify that the resulting response spectra from these acceleration time histories envelope the design basis ground response spectra developed using the ASCE 43–05 methodology.

Staff Evaluation of Site-Specific Ground Motion Spectra

The NRC staff reviewed the information in the ISA Summary, Appendix F (AES, 2010b), and the response to NRC staff's request for additional information and find that applicant's site-specific ground motion spectra adequately represent potential future earthquake activity at the proposed EREF site. The NRC staff finds that the applicant developed its site-specific ground motion spectra based on standard seismological practice with sufficient detail to support seismic design. Vertical ground motions that are two-thirds the horizontal motions are appropriate and consistent with information in the seismological literature. Based on this review and the applicant's plan to develop acceleration time histories to support design analyses, the NRC staff concludes that the site-specific ground motion spectra presented in the SAR; the ISA Summary; and the ISA Summary, Appendix F is acceptable. The information provided an adequate approach to develop seismic inputs for design and performance consideration in the application and thereby meets the regulatory requirements in 10 CFR 70.65(b)(1).

1.3.3.4.2 Volcanic Hazard

The proposed EREF site is located within the eastern half of the Snake River Plain Physiographic Province. The Snake River Plain is characterized by extensive volcanic activity; and much of the region has been volcanically active since approximately 17 million years ago, when this portion of the North American tectonic plate overrode the Yellowstone hotspot. Based on historical patterns of volcanic activity in the region, three potential impacts of volcanic events were considered:

- Ash fallout from Cascade Range volcanoes
- Near-field silicic volcanism
- Inundation from basaltic lava flows

Each of these three types of volcanic activity could have potentially adverse effects on structures, systems, and components relied on for safety if such volcanic activity occurred during operation of the proposed EREF.

Because of the potential for volcanic activity at the site, the applicant developed a volcanic hazard assessment, which is described in Appendix D of the ISA Summary and is summarized in the ISA Summary Section 3.2.8 (AES, 2010b). The NRC staff reviewed information the applicant provided in the ISA Summary as well as responses to staff requests for additional information (AES, 2009a) regarding volcanic features of the site. The NRC staff also reviewed relevant literature cited in the ISA Summary, previous studies of volcanic hazards potentially affecting the INL (Kuntz, 1992; Lawrence Livermore, 1990; Hackett, 1994), and other literature cited therein to evaluate the potential volcanic hazards at the proposed EREF site.

Ash Fallout from Cascade Range Volcanoes

The applicant estimated that the maximum ash thickness that could be deposited on roofs of the proposed EREF is less than 8 cm (3.1 in) according to the results from the Volcanism Working Group Assessment for the nearby INL site (Lawrence Livermore, 1990). Based on dry and wet ash densities from the 1980 Mount Saint Helens eruption (Sarna-Wojcicki, 1981), the applicant estimated that an 8-cm (3.1-in)-thick ash blanket would result in roof loads of 4–10 g/cm^2

(8.2–20.5 lb/ft^2). The applicant noted that these potential ash loads are much smaller than the design basis roof loads for the extreme environmental snow load, which is 30.9 g/cm^2 (63.2 lb/ft^2). Thus, the roof loads from volcanic ash are bounded by the design basis snow load. The NRC staff reviewed the information provided in the ISA Summary (AES, 2010b) and responses to staff requests for additional information (AES, 2009a) and find it acceptable because the potential ash loads at the EREF site from a Cascade Range volcanic eruption rely on logical analyses that are supported by well-established volcanic data. Moreover, the analysis and results the applicant provided are consistent with a similar analysis the NRC staff relied on in support of the SED for the Idaho Spent Fuel Facility at the nearby INL site (NRC, 2004). In that SER, NRC staff found that a 10-cm (3.94-in) ash load was bounding. The NRC staff therefore concludes that the regulatory requirements in 10 CFR 10.65(b)(1) have been met with regard to this topic.

Near-Field Silicic Volcanism

Within the past 1.2 million years, five silicic dome volcanoes have formed in a region of the Eastern Snake River Plain named the Axial Volcanic Zone (Lawrence Livermore, 1990). As shown in Figure D–1 of Appendix D in the ISA Summary, the closest of these domes is East Butte, which is located more than 10 km (6.1 mi) from the proposed EREF (AES, 2010b). The applicant estimated a recurrence interval for silicic volcanism of 220,000 years (5 events in 1.2 million years) or a rate of less than 4×10^{-6}/year, which is an order of magnitude smaller than recurrence rates for basaltic volcanism.

Small-volume flows and tephra falls are common features of silicic dome eruptions (Heiken, 1987). Active silicic dome volcanoes comparable in size to those in the Eastern Snake River Plain produce only local flows and ash-fall deposits approximately 10 cm (4 in) thick near the volcano (Scott, 1987), with deposit thicknesses decreasing to 1–2 cm (0.5–1 in) thick at distances of about 25 km (15.5 mi) from the volcano (Heiken, 1987). Thus, a 10-cm (4-in)-thick ash-fall deposit is determined to be a credible upper limit for potential hazards from a new silicic dome volcano forming at least 10 km (6.2 mi) from the proposed EREF.

Because of these characteristics of near-field silicic volcanism—small recurrence rate and limited effects at the site—the applicant considered these types of volcanoes to pose no significant hazard to the proposed EREF. The applicant also noted that the Volcanism Working Group (Lawrence Livermore, 1990) at INL reached a similar conclusion. This group examined the potential hazards from silicic volcanism with respect to critical nuclear facilities within the INL.

The NRC staff reviewed the information provided in the ISA Summary (AES, 2010b) and finds it acceptable because the potential hazards at the EREF site from near-field silicic dome volcanoes are derived from well-known and appropriate volcanic information. In addition, the analysis and results the applicant provided are consistent with a similar analysis the NRC staff relied on in support of the Safety Evaluation Report for the Idaho Spent Fuel Facility at the nearby INL site (NRC, 2004). The NRC staff therefore concludes that the regulatory requirements in 10 CFR 70.65(b)(1) have been met with regard to this topic.

Inundation from Basaltic Lavas

The applicant identified that inundation by basaltic lava flows is the most significant of the three potential volcanic hazard types at the site. Inundation could be from a new basaltic volcano at the site or, more likely, from slow-moving, effusive lava moving downgradient from volcanic vents or cones within the known Eastern Snake River Plain volcanic zones. Based on their similar characteristics to basaltic lavas from Hawaii, the applicant describes these flows as hot, low viscosity, and slowly moving *pahoehoe* lavas with advancement rates of less than 1 km (0.6 mi) per day. The main hazards to the proposed EREF would be inundation and burning with associated release of corrosive gases.

The applicant developed a probabilistic volcanic hazard analysis for the proposed site based on extensive volcanic and geologic data from the Eastern Snake River Plain (Appendix D of AES, 2009b). The probabilistic analysis considered two approaches to develop estimates for the likelihood of basaltic lava inundation at the proposed site. The first approach used an event tree to define possible outcomes based on the probability of a random eruption in the axial volcanic zone. Subsequent branches of the event tree considered the likelihood that lava will reach the proposed site based on topographic effects and run-out distance from the volcano to the site. The second approach also assumes that a volcano forms randomly within the volcanic source zone and then conditions the probability that a volcano will form by the ratio of the average area of a lava flow divided by the area of the volcanic source zone. Both methods yield similar estimates of lava inundation at the proposed site on the order of 5×10^{-6}/year. Based on this probability, the applicant defines volcanism at the site as highly unlikely.

The NRC staff reviewed the probabilistic volcanic hazard assessment provided in the ISA Summary (AES, 2010b), including the supporting geologic and volcanic information, and finds it acceptable. The applicant's probabilistic volcanic hazard assessment is clearly documented and supported by well-established volcanic information. The results of the applicant's analysis are consistent with independent estimates of basaltic lava inundation for critical facilities within the INL reservation, including the analysis for the Central Facilities Area (Hackett, 2002) and the study done for the New Production Reactor (Lawrence Livermore, 1990). The probability estimate the applicant provided is also consistent with the results NRC staff relied on in support of the Safety Evaluation Report for the Idaho Spent Fuel Facility at the nearby INL site (NRC, 2004). The NRC staff therefore finds that the regulatory requirements in 10 CFR 70.65(b)(1) have been met with regard to this topic.

1.3.3.4.3 Slope Stability

The applicant indicated that the proposed facility site is relatively flat with gently sloping surfaces and small ridges (AES, 2009a,b). The NRC staff examined the topography maps where the proposed site is located [e.g., Figures 3.2-5 and 3.2-9 of the ISA Summary (AES, 2010b) and concur with the applicant that the proposed site is gently sloping; and slope instability is not a safety concern to the EREF operations. Consequently, the NRC staff concludes that the regulatory requirements in 10 CFR 70.65(b)(1) have been met with regard to this topic.

1.3.3.4.4 Liquefaction

The applicant indicated that the soil thickness at the proposed site varies from 0 to 4.3 m (0 to 14.0 ft) (AES, 2010b). The site soils are of eolian (wind-blown) origin and are classified as CL: low plasticity clays according to the Unified Soil Classification System. The soil natural moisture content ranges from 9.6 to 19.0 percent, and the groundwater table is more than 150 m (500 ft) below ground surface. The applicant concluded that liquefaction potential for the site appears highly unlikely because the groundwater table is deep, and the soils at the site are primarily clays. The NRC staff reviewed the information the applicant provided and find that the applicant's conclusion is technically supportable and therefore acceptable.

In addition, the applicant stated in its ISA Summary Section 3.2.7 (AES, 2010b) that it intends to conduct additional site subsurface investigations to support the final design of the proposed EREF and verify through the investigation results the conclusion on soil liquefaction potential. The applicant stated that, if the investigation results show that soil liquefaction at the proposed site is possible, it will assess the site soil liquefaction potential using the applicable guidance of Regulatory Guide 1.198 (NRC, 2003).

The NRC staff reviewed the information presented concerning soil liquefaction potential and find that the because the applicant t will perform liquefaction assessment using Regulatory Guide 1.198 for final facility design, that staff find this is acceptable. Thus, the NRC staff concludes that the regulatory requirements in 10 CFR 70.65(b)(1) have been met with regard to this topic.

1.3.3.4.5 Settlement

The soil settlement, bearing capacity, and static and dynamic soil and rock properties for the proposed EREF site are not currently available. The applicant states in its ISA Summary Section 3.2.7 (AES, 2010b) that it will conduct additional geotechnical investigations of the proposed site to determine static and dynamic soil and rock properties, evaluate foundation bearing capacity, estimate settlement and differential settlement, and provide geotechnical input for soil–rock structure interaction analysis to support the final facility design.

For settlement evaluation, the applicant states in its ISA Summary Section 3.2.7 (AES, 2010b) that it will use the applicable methods provided in one or more of the following documents: Naval Facilities Engineering Command (NAVFAC) Design Manual (DM)–7.01, Soil Mechanics (Department of Navy, 1986a); Foundation Engineering Handbook (Winterkorn, 1975); Foundation Analysis and Design (Bowles, 1996); and Foundation Engineering (Peck, 1974). To determine allowable bearing pressure, the applicant states in its ISA Summary Section 3.2.7 (AES, 2010b) that it will follow the applicable guidance in one or more of the following documents: NAVFAC DM–7.02, Foundations and Earth Structures (Department of Navy, 1986b); Foundation Engineering Handbook (Winterkorn, 1975); Foundation Analysis and Design (Bowles, 1996); Foundation Engineering (Peck, 1974); and Rock Foundations (ASCE, 1996). The NRC staff finds the applicant's plan to conduct an additional geotechnical investigation to develop design information to support final facility design acceptable. In addition, the NRC staff finds the guidance the applicant plans to use to obtain information concerning differential settlements, soil-bearing capacity, and dynamic soil and rock properties for the final facility design acceptable. Based on the applicant's plans to conduct additional geotechnical study and the described guidance, the NRC staff concludes, that the regulatory requirements in 10 CFR 70.65(b)(1) have been met with regard to these topics.

1.3.4 Evaluation Findings

The staff has reviewed the site description for the proposed EREF according to Section 1.3 of the Standard Review Plan (NRC, 2002). The applicant has adequately described and summarized general information pertaining to: (1) the site geography, including its location relative to prominent natural and manmade features such as rivers, airports, population centers, schools, and commercial and manufacturing facilities; (2) population information on the basis of the most current available census data to show population distribution as a function of distance from the facility; (3) meteorology, hydrology, and geology for the site; and (4) applicable design basis events.

The staff has verified that the site description is consistent with the information used as a basis for the ER (USEC, 2003a), the EP (USEC, 2006a), and the ISA Summary (USEC, 2006b); and that it demonstrates compliance with regulatory requirements in 10 CFR 30.33, 10 CFR 40.32, 10 CFR 70.22, and 10 CFR 70.65(b)(1).

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CHAPTER 2.0 ORGANIZATION AND ADMINISTRATION

The purpose of the U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's organization and administration is to evaluate whether the license application describes proposed management policies that provide reasonable assurance that the applicant will plan, implement, and control site activities in a manner that ensures the safety of workers and the public, and protects the environment. The review also ensures that the applicant has identified and provided adequate qualification descriptions for key management positions.

2.1 Regulatory Requirements

Title 10 of the *Code of Federal Regulations* (10 CFR) 70.22(a)(6) requires that the applicant provide the technical qualifications, including training and experience of the applicant and members of the staff. In addition, 10 CFR 70.23(a)(2), 10 CFR30.33(a)(3), and 10 CFR 40.32(b) require that an applicant be qualified by reason of training and experience to use the licensed material for the purpose requested. And, 10 CFR 70.23(a)(4), 10 CFR 30.33(a)(2), and 10 CFR 40.32(c) require that the applicant's proposed equipment and facilities are adequate to protect health and minimize danger to life or property. Also, the management measures required under 10 CFR 70.62(d) establish a management system and administrative procedures that apply to items relied on for safety (IROFS) to ensure their availability and reliability.

2.2 Regulatory Guidance and Acceptance Criteria

The guidance applicable to the NRC's review of the organization and administration section of the Safety Analysis Report (SAR) (AES, 2010a) is contained in Chapter 2 of the "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 (NRC, 2002). Section 2.4.3 of NUREG-1520 lists the acceptance criteria for both new and existing facilities. Applications for new facilities must address both sets of acceptance criteria.

2.3 Staff Review and Analysis

Chapter 2.0 of the SAR (AES, 2010a) describes the organizations that will be responsible for managing the design, construction, operation, and decommissioning of the proposed Eagle Rock Enrichment Facility (EREF). The key management and supervisory positions and functions are described, including the personnel qualifications for each key position at the facility. The shift crew composition is also listed.

2.3.1 Organization

The regulations in 10 CFR 70.23(a)(2), 10 CFR30.33(a)(3), and 10 CFR 40.32(b) require that an applicant be qualified by reason of training and experience to use the licensed material for the purpose requested. And, 10 CFR 70.23(a)(4), 10 CFR 30.33(a)(2), and 10 CFR 40.32(c) require that the applicant's proposed equipment and facilities are adequate to protect health and minimize danger to life or property. Thus, the applicant must implement an organization and appropriate administrative elements to support these regulatory requirements. The acceptance criteria in Section 2.4.3 of NUREG 1520 (NRC, 2002) provide guidance that the applicant must provide the following information: (1) identify and functionally describe the specific organizational groups that are responsible for managing the design, construction, and operation of the facility, (2) organizational charts, and (3) plans to commission the facility's startup and operation, including the transition from the startup phase to operations, under the direct supervision of the applicant's personnel responsible for safe operations.

In Section 2.1 of the SAR (AES, 2010a), AREVA Enrichment Services, LLC (AES) provided a functional description of the specific organizational groups responsible for managing the design, construction, operation, and decommissioning of the EREF. The following discussion summarizes the description.

As described in Section 2.1 of the SAR (AES, 2010a), the top level of the applicant's organization is the AES President. The AES President is responsible for the design, quality assurance (QA), construction, operation, and decommissioning of the EREF (AES, 2010a). During the engineering, procurement, and construction phase, the Project Director, Plant Operations Manager, and QA Manager report to the AES President (AES, 2010a). During the operating phase, the Plant Manager, the QA Manager, and the Safety Review Committee (SRC) report to the President (AES, 2010a). Figure 1-2 of the Quality Assurance Program Description (QAPD) (AES, 2010b), "Engineering, Procurement, and Construction Organizational Chart," illustrates the authority and lines of communication for the engineering, procurement, and construction phase. Figure 1-1 of the QAPD (AES, 2010b), "Eagle Rock Enrichment Facility Operations Organizational Chart," shows the authority and lines of communication for the operating phase. In Sections 2.1.3 and 2.1.4 of the SAR (AES, 2010a), AES stated that the position descriptions of key management personnel for the design, construction organization, and the operating organization will be accessible to all affected personnel and to the NRC.

AES has contracted with Enrichment Technology Company Limited (ETC) to design the core process technology (AES, 2010a). ETC will design, manufacture, and deliver the centrifuges necessary for facility operation (AES, 2010a). In addition, ETC will supply technical assistance and consultation services during installation and operation (AES, 2010a). AES will contract with an architect/engineering firm to further specify, design, and build the supporting structures and systems of the EREF (AES, 2010).

Section 2.1.4 of the SAR (AES, 2010a) describes AES's approach for the transition from the design and construction phase to the operations phase. Towards the end of the construction of the EREF, the focus of the AES organization will shift from design and construction to initial startup and operation (AES, 2010a). As the facility nears completion, AES will staff the EREF operating organization to ensure a smooth transition from construction to operations activities (AES, 2010a). Design and construction personnel will be integrated into the operations organization to provide technical support during initial startup of the facility and the transition into the operations phase (AES, 2010a). Also, ETC will have personnel integrated into the AES organization to provide technical support during startup of the facility and the transition to the operations phase (AES, 2010a).

As the construction of systems is completed, the systems will undergo required acceptance testing, followed by turnover from the construction organization to the operating organization by means of a detailed transition plan (AES, 2010a). This turnover will include the physical systems and corresponding design information and records (AES, 2010a). After turnover, the operating organization will be responsible for system maintenance and configuration management (AES, 2010a). The design basis for the facility is maintained during the transition from construction to operations through the configuration management system described in Chapter 11, "Management Measures," of the SAR (AES, 2010a). Chapter 11 of this Safety Evaluation Report (SER) describes the NRC staff's evaluation of the configuration management system.

As stated above, the applicant has identified and described the proposed organization that would be responsible for managing the design, construction, and operation of the proposed facility. The applicant has also provided organization charts. In addition, AES will have written position descriptions that will be available to all affected personnel and to the NRC. The proposed organization provides an acceptable management system for ensuring that the design, construction, and operation of the facility will meet the NRC's regulatory requirements. The information provided by the applicant is consistent with the guidance in Section 2.4.3 of NUREG-1520 (NRC, 2002) that the applicant should identify and describe the specific organizational groups that are responsible for managing the design, construction, and operation of the facility and provide organizational charts. In addition, the information provided by the applicant is consistent with the guidance in Section 2.4.3 of NUREG 1520 (NRC, 2002) to describe specific plans to commission the facility's startup and operation, including the transition from the startup phase to operations under the direct supervision of the applicant's personnel responsible for safe operations. The description of the proposed organization is, therefore, acceptable.

2.3.2 Organizational Responsibilities and Qualifications

The regulations in 10 CFR 70.22(a)(6) require that the applicant provide the technical qualifications, including training and experience of the applicant and members of the staff. In addition, the regulations in 10 CFR 70.23(a)(2), 10 CFR30.33(a)(3), and 10 CFR 40.32(b) require that an applicant be qualified by reason of training and experience to use the licensed material for the purpose requested. Following the guidance in NUREG-1520 (NRC, 2002), the staff reviewed the applicant's SAR (AES, 2010a) to ensure that the personnel responsible for managing the design, construction, and operation of the facility have substantive breadth and level of experience and are appropriately available; that clear, unambiguous management controls and communications exist among the organizational units that are responsible for managing the design and construction of the facility; that the qualifications, responsibilities, and authorities for key supervisory and management positions with health, safety, and environment (HS&E) responsibilities are clearly defined in position descriptions that are available to all affected personnel and to the NRC, upon request; and that the individual delegated overall responsibility for the HS&E functions will have the authority to shut down operations if they appear to be unsafe and in that case, must approve restart of shutdown operations.

In Section 2.2 of the SAR (AES, 2010a), the applicant provides information concerning the minimum qualifications, responsibilities, authorities, and lines of communication for key management personnel. Responsible managers have the authority to delegate tasks to other individuals; however, the responsible manager retains the ultimate responsibility and accountability for implementing the applicable requirements (AES, 2010a). According to Section 2.2.4 of the SAR (AES, 2010a), the AES President will evaluate the nuclear experience of each individual. "Responsible nuclear experience" for these positions includes:

(a) responsibility for and contributions toward support of facility(ies) in the nuclear fuel cycle and (b) experience with chemical materials and/or processes (AES, 2010a). The AES President will approve the assignment of individuals to management positions which report directly to the President, and to positions on the SRC (AES, 2010a). Assignments for all other staff positions will be made within the normal administrative practices of the facility (AES, 2010a).

The qualifications of the individuals assigned to the key facility positions described in Section 2.2.1 of the SAR will be maintained in employee personnel files or other appropriate files at the facility (AES, 2010a). Position descriptions of key management personnel will be accessible to AES personnel and to the NRC (AES, 2010a). Development and maintenance of qualification records and training programs are the responsibility of the Human Resources Manager (AES, 2010a).

The AES President has overall responsibility for the design, QA, construction, operation, and decommissioning of the EREF (AES, 2010a). He or she is also responsible for the QA Program and for determining the status, adequacy, and effectiveness of its implementation (AES, 2010a).

The Plant Manager; the QA Manager; the Human Resources Manager; and the Communications, Community Affairs Manager will be appointed by, and will report to, the AES President (AES, 2010a). Their responsibilities are described in Section 2.2.1 of the SAR (AES, 2010a). The minimum qualifications for these positions are provided in Section 2.2.4 of the SAR (AES, 2010a). The following summarizes their responsibilities and minimum qualifications.

- Plant Manager: The Plant Manager will have direct responsibility for operation of the facility in a safe, reliable, and efficient manner (AES, 2010a). The Plant Manager will be responsible for the protection of the facility staff and the general public from radiation and chemical exposure and/or any other consequences of an accident at the facility and also bears the responsibility for compliance with the facility license (AES, 2010a). The Plant Manager or designee(s) will have the authority to approve and issue procedures (AES, 2010a). The Plant Manager will be knowledgeable of the enrichment process, enrichment process controls and ancillary processes, criticality safety control, chemical safety, industrial safety, and radiation protection program concepts as they apply to the overall safety of a nuclear facility (AES, 2010a). The Plant Manager will have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and ten years of responsible nuclear experience (AES, 2010a).

- QA Manager: The QA Manager will have overall responsibility for development, management, implementation, and independent oversight of the EREF QA Program (AES, 2010a). The facility line managers and their staff who are responsible for performing quality-affecting work will be responsible for ensuring implementation of and compliance with the QA Program (AES, 2010a). The QA Manager position will be independent from other management positions at the facility to ensure the QA Manager has direct access to the AES President for matters affecting quality (AES, 2010a). The QA Manager will have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and at least six years of responsible nuclear experience in the implementation of a QA program (AES, 2010a). The QA Manager will have at least four years experience in a QA organization at a nuclear facility (AES, 2010a).

- Human Resources Manager: The Human Resource Manager will have responsibility for community relations, ensuring adequate staffing, and providing administrative support services to the facility including document control (AES, 2010a). The Human Resources Manager will have as a minimum, a bachelor's degree in personnel management, business administration or related field, and three years of appropriate, responsible experience in implementing and supervising human resource responsibilities at an industrial facility (AES, 2010a).

- Communications, Community Affairs Manager: The Communications, Community Affairs Manager will have responsibility for providing information about the facility and AES to the public and media (AES, 2010a). During an abnormal event at the facility, the Communications, Community Affairs Manager will ensure that the public and media receive accurate and up-to-date information (AES, 2010a). The Communications, Community Affairs Manager will have as a minimum, a bachelor's degree in public relations, political science or business administration and three years of appropriate, responsible experience in implementing and supervising a community relations program (AES, 2010a).

The Environmental, Health, Safety and Licensing (EHS&L) Manager; the Project Manager; the Training Manager; the Uranium Management Manager; and the Operations Manager will report to the Plant Manager. AES describes the responsibilities for each of these positions in Section 2.2.1 of the SAR (AES, 2010a). The minimum qualifications for these positions are provided in Section 2.2.4 of the SAR (AES, 2010a). The following summarizes the responsibilities and minimum qualifications for these managers.

- EHS&L Manager: The EHS&L Manager will have the overall responsibility for the development and implementation of programs addressing worker health and safety; environmental protection; and licensing and permitting (AES, 2010a). The EHS&L Manager will be responsible for maintaining compliance with safeguards; appropriate rules, regulations, and codes; and implementation and control of the Fundamental Nuclear Material Control (FNMC) Plan, the Physical Security Plan (PSP), and the Standard Practices Procedure Plan for the Protection of Classified Matter (SPPP) (AES, 2010a). EHS&L activities cover nuclear criticality safety, radiation protection, chemical safety, environmental protection, emergency preparedness, industrial safety, and development and implementation of security programs (AES, 2010a). The EHS&L Manager will work with the other facility managers to ensure consistent interpretations of EHS&L requirements, perform independent reviews, and support facility and operations change control reviews (AES, 2010a). This position will be independent from other operations management positions at the facility to ensure objective EHS&L audit, review, and control activities (AES, 20101). The EHS&L Manager will have the authority to order the shutdown of operations if they appear to be unsafe or non-compliant with applicable regulatory requirements and must consult with the Plant Manager prior to restart of shutdown operations after the deficiency, or unsatisfactory condition, has been resolved (AES, 2010a). The EHS&L Manager will have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and at least five years of responsible nuclear experience in EHS&L or related disciplines (AES, 2010a). The EHS&L Manager will also have at least one year of direct experience in the administration of nuclear criticality safety evaluations and analyses (AES, 2010a).

- Project Manager: The Project Manager will have overall responsibility for managing the engineering, procurement, construction, and startup of facility modifications and expansion (AES, 2010a). This will include managing the work and contracts with the technology supplier (i.e., ETC) (AES, 2010a). The Project Manager will have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience (AES, 2010a).

- Training Manager: The Training Manager will have responsibility for the development, implementation, and administration of the plant training programs, including maintenance of the plant training database (AES, 2010a). The training programs will address qualifications of workers to perform work as well as required safety training (AES, 2010a). The Training Manager will have a minimum of five years of appropriate, responsible experience in implementing and supervising a training program (AES, 2010a).

- Uranium Management Manager: The Uranium Management Manager will have responsibility for uranium hexafluoride (UF_6) cylinder management (including compliance with transportation requirements) and directing the scheduling of enrichment operations (AES, 2010a). During an absence of the Plant Manager, the Uranium Management Manager may assume the responsibilities and authorities of the Plant Manager (AES, 2010a). The Uranium Management Manager will have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience (AES, 2010a).

- Operations Manager: The Operations Manager will have the responsibility of directing the day-to-day operation of the facility and ensuring that that the operations are conducted safely and in compliance with any license conditions (AES, 2010a). The Operations Manager will also be responsible for the plant maintenance function, which includes activities to assure that Items Relied on for Safety (IROFS) are reliable and available when needed (AES, 2010a). During the absence of the Plant Manager, the Operations Manager may assume the responsibilities and authorities of the Plant Manager (AES, 2010a). The Operations Manager will have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience (AES, 2010a).

AES stated that during an absence of the Plant Manager, the Uranium Management Manager or the Operations Manager may assume the responsibilities and authorities of the Plant Manager (AES, 2010a). The EHS&L Manager has the authority to order the shutdown of operations if they appear to be unsafe or non-compliant with applicable regulatory requirements and must consult with the Plant Manager before restart of shutdown operations after the deficiency, or unsatisfactory condition, has been resolved (AES, 2010a).

The Radiation Protection/Chemistry Manager; the Nuclear Criticality Safety Manager; the Licensing and Compliance Manager; the Safeguards Manager; the Safety, Security, and Emergency Preparedness Manager; and the Industrial Safety Manager report to the EHS&L Manager. AES described the responsibilities for each of these positions in Section 2.2.1 of the

SAR (AES, 2010a). The minimum qualifications for these positions are provided in Section 2.2.4 of the SAR (AES, 2010a). The following summarizes the responsibilities and minimum qualifications for these managers.

- Radiation Protection/Chemistry Manager: The Radiation Protection/Chemistry Manager will have responsibility for developing and implementing programs to limit personnel radiological exposures and environmental impacts associated with facility operations, including the As Low as Reasonably Achievable (ALARA) program (AES, 2010a). During emergency conditions, the Radiation Protection/Chemistry Manager's duties may expand as described in Section 2.2.1 of the SAR (AES, 2010a). The Radiation Protection/Chemistry Manager will also be responsible for the implementation of chemistry analysis programs and procedures for the facility, including, effluent sample collection, chemical analysis of effluents, comparison of effluent analysis results to limits, and reporting of chemical analysis of effluents to appropriate regulatory agencies (AES, 2010a). The Radiation Protection/Chemistry Manager will have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience (AES, 2010a).

- Nuclear Criticality Safety Manager: The Nuclear Criticality Safety Manager will be responsible for the development and implementation of the nuclear criticality safety program (AES, 2010a). Key responsibilities will include the performance of nuclear criticality safety analyses and evaluations of applicable operations involving special nuclear material and changes to those operations; establishing limits and controls based on those analyses and evaluations; assuring the proper incorporation of limits and controls into applicable procedures and instructions; and monitoring plant compliance with nuclear criticality safety requirements (AES, 2010a). The Nuclear Criticality Safety Manager will have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience (AES, 2010a). The Nuclear Criticality Safety Manager will also have at least one year of direct experience in the administration of nuclear criticality safety evaluations and analyses (AES, 2010a).

- Safeguards Manager: The Safeguards Manager will have responsibility for ensuring the proper implementation of the FNMC Plan (AES, 2010a). This position will be separate from and independent of other departments to ensure separation between the safeguards group and the other departments. In matters involving safeguards, the Safeguards Manager has direct access to the Plant Manager (AES, 2010a). The Safeguards Manager will have, as a minimum, a bachelor's degree in an engineering or scientific field, and five years of experience in the management of a safeguards program for special nuclear material, including responsibilities for material control and accounting (AES, 2010a). No credit for academic training may be taken toward fulfilling this experience requirement (AES, 2010a).

- Safety, Security and Emergency Preparedness Manager: The Safety, Security, and Emergency Preparedness Manager will be responsible for implementation and maintenance of the integrated safety analysis, industrial hygiene and safety, chemical safety, fire protection, security, and emergency preparedness including the responsibility for ensuring the facility remains prepared to react and respond to any emergency situation that may arise (AES, 2010a). This will include emergency preparedness training of facility personnel, facility support personnel, the training of, and coordination with, offsite emergency response organizations (EROs), and conducting periodic drills to ensure the training of facility personnel and offsite response organization personnel is maintained up-to-date (AES, 2010a). The Safety, Security and Emergency Preparedness Manager will also be responsible for the protection of classified matter at the facility and obtaining security clearances for facility personnel and support personnel. In matters involving physical protection of the facility or classified matter, the Safety, Security and Emergency Preparedness Manager has direct access to the Plant Manager (AES, 2010a). The Safety, Security and Emergency Preparedness Manager will have, as a minimum, a bachelor's degree in an engineering or scientific field, and five years of experience in the responsible management of physical security at a facility requiring security capability similar to that required for the facility (AES, 2010a). No credit for academic training may be taken toward fulfilling this experience requirement (AES, 2010a).

- Industrial Safety Manager: The Industrial Safety Manager will have responsibility for the implementation of industrial safety programs and procedures, including programs and procedures for training individuals in safety and maintaining the performance of the facility fire protection systems (AES, 2010a). The Industrial Safety Manager will have, as a minimum, a bachelor's degree (or equivalent) (AES, 2010a).

The Engineering Manager; the Procurement Manager; the Construction Manager; the Startup Manager; and the Information Technology Manager will report to the Project Manager. AES describes the responsibilities for each of these positions in Section 2.2.1 of the SAR (AES, 2010a). The minimum qualifications for these positions are provided in Section 2.2.4 of the SAR (AES, 2010). The following summarizes the responsibilities and minimum qualifications for these managers.

- Engineering Manager: The Engineering Manager will be responsible for site characterization; facility design and the design control process; configuration management; engineering; and acceptance test coordination, including test control of facility modifications and expansion (AES, 2010a). The Engineering Manager will be also responsible for records management and document control, and approving disposition of nonconforming items when dispositioned as "repair" or "use-as-is" during operations (AES, 2010a). The Engineering Manager will have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear engineering program (AES, 2010a).

- Procurement Manager: The Procurement Manager will be responsible for procurement; providing procurement material control services (including supplier qualification coordination, purchasing, contracting, receiving and control of nonconforming items); and material control (including handling, storage and shipping) (AES, 2010a). The Procurement Manager will be responsible for supply strategy and development of qualified long-lead-time and complex-system suppliers (AES, 2010a). The Procurement Manager will have, as a minimum, a bachelor's degree (or equivalent) in an engineering or business field and have a minimum of five years of responsible experience in purchasing and supply chain management (AES, 2010a).

- Construction Manager: The Construction Manager will be responsible for managing the construction of facility modifications and expansion to the EREF (AES, 2010a). This responsibility will include managing the activities of qualified contractors in the preparation of construction documents and the construction of facility modifications and expansion (AES, 2010a). The Construction Manager will have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear construction program (AES, 2010a).

- Startup Manager: The Startup Manager will be responsible for the overall preoperational and startup test program of facility modifications and expansion (AES, 2010a). This individual will be responsible for the development of preoperational and startup test procedures, providing technical advice to personnel conducting the tests, briefing personnel responsible for operation of the plant during the tests, ensuring that the tests are performed in accordance with the applicable procedures, and generating test reports (AES, 2010a). The Startup Manager will have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and have a minimum of five years of responsible experience in nuclear plant operations and maintenance (AES, 2010a).

- Information Technology (IT) Manager: The IT Manager will be responsible for maintaining all computer software programs related to the nuclear material accounting at EREF (AES, 2010a). This individual will also be responsible for EREF computer database for generation of nuclear material control charts (AES, 2010a). The IT Manager will have, as a minimum, a bachelor's degree (or equivalent) in computer science, and five years of experience in the computer related field (AES, 2010a).

The Cylinder Management Manager; Production Scheduling Manager; and the Warehouse and Materials Manager will report to the Uranium Management Manager. The following summarizes the responsibilities and minimum qualifications for these managers.

- Cylinder Management Manager: The Cylinder Management Manager will have responsibility for ensuring that cylinders of UF_6 are received and routed correctly at the facility, and will be responsible for all transportation licensing (AES, 2010a). The Cylinder Management Manager will have a minimum of three years of appropriate, responsible experience in implementing and supervising a continuous production scheduling program (AES, 2010a).

- Production Scheduling Manager: The Production Scheduling Manager will have the responsibility for developing and maintaining production schedules for enrichment services (AES, 2010a). The Production Scheduling Manager will have a minimum of three years of appropriate, responsible experience in implementing and supervising a continuous production scheduling program (AES, 2010a).

- Warehouse and Materials Manager: The Warehouse and Materials Manager will have the responsibility for ensuring spare parts and other materials needed for operation of the facility are ordered, received, inspected, and stored properly (AES, 2010a). The

Warehouse and Materials Manager will have a minimum of three years of appropriate, responsible experience in implementing and supervising a purchasing and inventory program (AES, 2010a).

The Production Managers and the Maintenance Manager will report to the Operations Manager. The following summarizes the responsibilities and minimum qualifications for these managers.

- Production Managers: The Production Managers will be responsible for enrichment operations, feed and withdrawal operations, utilities, shift operations, packaging, and transportation (AES, 2010a). Production Managers will have a minimum of five years of appropriate, responsible experience in implementing and supervising a nuclear operations program (AES, 2010a).

- Maintenance Manager: The Maintenance Manager will have the responsibility of directing and scheduling maintenance activities to ensure proper operation of the facility, including preparation and implementation of maintenance procedures (AES, 2010a). The Maintenance Manager will also have responsible for safe and reliable performance of preventive and corrective maintenance and support services on buildings/facilities and equipment (including IROFS), and for integrated planning and scheduling (AES, 2010a). In addition, the Maintenance Manager will coordinate and maintain testing programs for the facility (AES, 2010a). This will include testing of systems and components to ensure the systems and components are functioning as specified in design documents (AES, 2010a). The Maintenance Manager will have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and four years of responsible nuclear experience (AES, 2010a).

Other key personnel will include the Measurement Control Program Manager; the Production Supervisors; the Criticality Safety Engineer; the Chemical Safety Engineer; the QA Inspectors; the QA Auditors; the QA Technical Support, the Administration Manager; and the Document Control Manager. AES describes the responsibilities of each of these positions in Section 2.2.1 of the SAR (AES, 2010a). The minimum qualifications for these positions are provided in Section 2.2.4 of the SAR (AES, 2010a). (AES, 2010a) The following summarizes the responsibilities and minimum qualifications for these managers and staff.

- Measurement Control Program Manager: The Measurement Control Program Manager will report to the Safeguards Manager and will have responsibility for the EREF Measurement Control Program (AES, 2010a). The EREF Measurement Control Program will be provided to ensure adequate calibration frequencies, sufficient control of biases, and sufficient measurement precisions for nuclear material control and accounting (AES, 2010a). The Measurement Control Program Manager will have, as a minimum, a bachelor's degree in an engineering or scientific field, and five years of experience in the management control program (AES, 2010a).

- Production Supervisors: The Production Supervisors will report to their respective Production Managers (AES, 2010a). The Production Supervisors will be responsible for control of materials, personnel, equipment, and activities in specific areas (AES, 2010a). These responsibilities will include assuring that formal approved procedures are available and adhered to by operators and other applicable personnel (AES, 2010a).

Production Supervisors will have a minimum of three years of appropriate, responsible experience in implementing and supervising a nuclear operations program (AES, 2010a).

- Criticality Safety Engineer: Criticality Safety Engineers will report to the Nuclear Criticality Safety Manager (via a designated supervisory position, if applicable) (AES, 2010a). Criticality Safety Engineers will have responsibility for the preparation and/or review of nuclear criticality safety evaluations and analyses and for conducting and reporting periodic nuclear criticality safety assessments (AES. 2010a). Nuclear criticality safety evaluations and analyses require independent reviews by a Criticality Safety Engineer. Criticality Safety Engineers will hold a Bachelor of Science or Bachelor of Arts degree in an engineering or scientific field and will have successfully completed a training program, applicable to the scope of operations, in the physics of criticality and in associated safety practices (AES, 2010a).

  Should a change to the facility require a nuclear criticality safety evaluation or analysis, an individual who, as a minimum, possesses the equivalent qualifications of the Criticality Safety Engineer will perform the evaluation or analysis (AES, 2010a). In addition, this individual will have at least two years of experience performing criticality safety analyses and implementing criticality safety programs (AES, 2010a). An independent review of the evaluation or analysis will be performed by a qualified Criticality Safety Engineer (AES, 2010a).

- Chemical Safety Engineer: The Chemical Safety Engineer will report to the Radiation Protection/Chemistry Manager (via a designated supervisory position, if applicable) and will be responsible for the preparation and/or review of chemical safety programs and procedures for the facility (AES, 2010a). The Chemical Safety Engineer will have a minimum of two years experience in the preparation and/or review of chemical safety programs and procedures (AES, 2010a). This individual will hold a bachelor's degree (or equivalent) in an engineering or scientific field and will have successfully completed a training program, applicable to the scope of operations, in chemistry and in associated safety practices (AES, 2010a).

- QA Inspectors: The QA Inspectors will report to the QA Manager (via a designated supervisory position, if applicable) and will have the responsibility for performing inspections related to the implementation of the AES QA Program (AES, 2010a).

- QA Auditors: The QA Auditors will report to the Quality Assurance Manager (via a designated supervisory position, if applicable) and will have the responsibility for performing audits related to the implementation of the AES QA Program (AES, 2010a).

- QA Technical Support: The QA Technical Support personnel report to the Quality Assurance Manager (via a designated supervisory position, if applicable) and will have the responsibility for providing technical support related to the implementation of the AES QA Program (AES, 2010a).

- Administration Manager: The Administration Manager will report to the Human Resources Manager and will have the responsibility for ensuring that support functions such as accounting, word processing, and general office management are provided for the EREF (AES, 2010a). The Administration Manager will have a minimum of three years of appropriate, responsible experience in implementing and supervising administrative responsibilities at an industrial facility (AES, 2010a).

- Document Control Manager: The Document Control Manager will report to the Human Resources Manager (AES, 2010a). The Document Control Manager will have a minimum of three years of appropriate, responsible experience in implementing and supervising a document control program (AES, 2010a).

In addition, AES briefly discussed the shift crew composition at the EREF in Section 2.2.2 of the SAR (AES, 2010a). The minimum operating shift crew will consist of a Production Supervisor (or Deputy Production Supervisor in the absence of the Production Supervisor), one Control Room operator, one radiation protection technician, one operator for each Cascade Hall and associated UF_6 handling systems, and security personnel (AES, 2010a). When only one Cascade Hall is in operation, a minimum of two operators will be required (AES, 2010a). At least one criticality safety engineer will be available, with appropriate ability to be contacted by the Production Supervisor, to respond to any routine request or emergency condition (AES, 2010a). This availability may be offsite if adequate communication ability is provided to allow response as needed (AES, 2010a).

The staff concludes that the applicant has identified the responsibilities, qualifications, and authorities of the key personnel responsible for managing the design, construction, and operations of the proposed facility and for health, safety, and engineering responsibilities. These responsibilities, qualifications, and authorities, as described above, are clearly defined and include sufficient breadth and level of experience to ensure that competent managers will be in place. The information the applicant provided is consistent with the guidance in Section 2.4.3(3) of NUREG-1520 (NRC, 2002) that the personnel responsible for managing the design, construction, and operation of the facility have substantive breadth and level of experience and will be available; that the qualifications, responsibilities, and authorities for key supervisory and management position descriptions will be accessible to affected personnel and to the NRC, upon request, that clear management controls and communications will exist among organizational units, and that the individual with overall health, safety, and environment responsibilities is authorized to shut down operations if they appear to be unsafe. The staff therefore finds the applicant's information acceptable.

2.3.3 Management Control

The management measures required under 10 CFR 70.62(d) establish the management system and administrative procedures that apply to IROFS to ensure their availability and reliability. Following the guidance in NUREG 1520, Section 2.4.3 (NRC, 2002), the staff reviewed the applicant's SAR (AES, 2010a) to ensure that the health, safety, and environment organizations are independent of the operations organizations, allowing objective audits, reviews, or control activities; that activities essential for effective implementation of the health, safety, and environment functions are documented in formally approved, written procedures and prepared in compliance with a formal document control program; that a simple mechanism will be available for use by any person in the plant, for reporting potentially unsafe conditions or activities to the health, safety, and environment organization; that formal management measures will be established to ensure the availability and reliability of IROFS; and that written agreements will exist with offsite emergency resources such as fire, police, ambulance/rescue units, and medical services. Section 2.3 of the SAR (AES, 2010a) summarizes how the activities that are essential for implementation of the management measures and other EHS&L functions are documented in formally approved, written procedures and prepared in compliance with a formal document control program.

The QA Manager position will be independent from other management positions at the facility to ensure that the QA Manager has direct access to the AES President for matters affecting quality (AES, 2010a). The EHS&L Manager will be independent from other management positions at the facility to ensure objective EHS&L audit, review, and control activities (AES, 2010a). In matters involving physical protection of the facility or classified matter, the Safety, Security & Emergency Preparedness Manager will have direct access to the Plant Manager. The Safeguards Manager position is separate from and independent of other departments to ensure separation between the safeguards group and the other departments. Thus, the QA, EHS&L, Safety, Security & Emergency Preparedness, and Safeguards Managers are independent from the Operations Managers. In addition, as described in Sections 2.3.1 and 2.3.2 of this SER, the organizations having QA, engineering, safety and health, environmental, security, safeguards, and operations responsibilities have clear and well-defined lines of communication and authority.

In addition, AES will establish management measures to ensure compliance with the performance requirements of 10 CFR 70.61. Management measures, defined in 10 CFR 70.4, include configuration management, maintenance, training, procedures, and other QA measures.

A configuration management program is provided for IROFS throughout facility design, construction, testing, and operation; this is a means of establishing and maintaining a technical baseline for the facility based on clearly defined requirements. Section 11.3.1 of this SER evaluates the configuration management program.

In Section 2.3 of the SAR, AES described plans to implement a maintenance program for the operations phase of the facility that will include planned and scheduled preventive maintenance, surveillance, and performance trending to ensure that IROFS will be available and reliable to perform their intended safety functions.

AES will establish a formal training program that will include indoctrination training for all employees and will address criticality, radiological, chemical, and industrial safety; ALARA practices; and emergency procedures. In-depth training programs shall be provided to individuals, depending on specific job requirements, addressing radiological safety (for all personnel with access to the Restricted Area) and criticality safety control. Refresher training on radiological and criticality safety will be provided at least annually. Changes to the training program will be implemented to address incidents potentially compromising safety or changes to facilities or processes. Training records will be maintained by the Human Resources Manager. Evaluation of the applicant's training program is provided in Section 11.3.3 of this SER.

AES will conduct all activities involving licensed materials in accordance with approved, written procedures. These plant procedures will generally include operating, administrative, maintenance, and emergency procedures.

In Section 2.3.5.3 of the SAR (AES, 2010a), AES stated that the facility operating organization will provide, as part of the routine supervisory function, timely and continuing monitoring of operating activities to keep the Plant Manager current on general facility conditions and to verify that the day-to-day operating activities are conducted safely and in accordance with applicable administrative controls.

AES described the SRC for the EREF in Section 2.2.3 of the SAR (AES, 2010a). The SRC will report to the AES President and provide technical and administrative review and audit of operations that could impact the safety of workers, members of the public, or the environment. The SAR also describes the scope of activities to be reviewed and audited by the committee and the frequency of these audits. The SRC will be composed of at least 5 members, to include experts in operations, criticality safety, radiological safety, chemical safety, and industrial safety. The members, including the chairman and alternate members, will be formally appointed by the AES President. Members of the SRC will have an academic degree in an engineering or physical science field; and a minimum of 5 years of technical experience, of which a minimum of 3 years shall relate directly to one or more of the safety disciplines (criticality, radiological, chemical, or industrial). The SRC will meet at least once per calendar quarter. Review meetings will be held within 30 days of any incident that is reportable to the NRC. After a reportable incident, the SRC will review the incident's causes, the responses, and both specific and generic corrective actions to ensure that the problem is resolved. A written report of each SRC meeting and audit will be forwarded to the AES President, the Plant Manager, and other appropriate managers within 30 days, and be retained in accordance with the EREF records management system.

The AES will implement a QA Program that requires periodic audits of activities affecting quality, by the AES QA Department, to ensure that these activities are being conducted in accordance with QA Program requirements and established procedures. Section 11.3.5 of this SER evaluates audits and assessments.

In Section 2.3.5.4 of the SAR (AES, 2010a), AES stated that audited organizations will correct identified deficiencies in a timely manner. Audited organizations will respond to each audit report within the time period specified in the audit; and, for each identified deficiency, identify the corrective action taken or to be taken—whether or not the deficiency is considered to be indicative of other problems—and the corrective action taken or not to be taken for any such problems determined. Copies of audit reports and responses will be maintained in accordance with AES's records management system.

A Corrective Action Program (CAP) will be implemented in order to identify, report, evaluate, and investigate abnormal events that have the potential to threaten or lessen the effectiveness of AES's health, safety, or environmental protection programs. Written procedures, to be followed in the case of an abnormal event, will address incident identification, investigation, root cause analysis, environmental protection analysis, recording, reporting, and follow-up. Section 11.3.6 of this SER evaluates incident investigations and the CAP.

In Section 2.3.7 of the SAR (AES, 2010a), AES stated that employees who feel that safety or quality is being compromised have the right and responsibility to initiate a "stop work" process. The process is implemented through project or facility procedures; line management or other facility management; the safety organization (i.e., any of the safety engineers or managers); the NRC's requirements under 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations;" and the CAP.

In Section 2.3.8 of the SAR (AES, 2010a), AES discussed the implementation of a records management program to control the preparation and issuance of documents, including any changes to these documents. Document control procedures will be established to formalize the process of reviewing, approving, releasing, transmitting, and distributing new and revised documents and for the destruction or retention of superseded documents. Additional details on the records management program are discussed and evaluated in Section 11.3.7 of this SER.

The Emergency Plan for the EREF includes a description of the ERO and of coordination with offsite EROs through written agreements. Evaluation of emergency management is provided in Chapter 8 of this SER.

Based on the above discussion, the staff concludes that an EHS&L Manager position will be established independent of other management positions to ensure objective EHS&L audits, reviews, and control activities; that activities involving licensed material will be documented in formally approved, written procedures; that document control procedures will be established for the review, approval, release, and distribution of new and revised documents; that a "stop work" process will be available for use by employees for reporting situations where safety is being compromised; that management measures will be established to ensure the availability and reliability of IROFS; and that written agreements with offsite emergency resources will be implemented. (The management measures are evaluated in Chapter 11 of this SER. Emergency management is evaluated in Chapter 8 of this SER.) The information that the applicant provided is consistent with the guidance in Section 2.4.3 of NUREG 1520 (NRC, 2002) and is, therefore, acceptable.

2.4 Evaluation Findings

The staff has reviewed the organization and administration for the EREF in accordance with the acceptance criteria in Chapter 2 of NUREG-1520 (NRC, 2002). The staff reviewed the applicant's organization, key management position summaries and qualifications, and management controls for providing adequate safety management and management measures for the safe operation of the facility. These organizational and administrative elements describe: (1) clear responsibilities and associated resources for the design, construction, and operation of the facility and (2) the applicant's plans for managing and operating the project. The staff has reviewed these plans as described in the SAR (AES, 2010a) and concludes that they provide reasonable assurance that an acceptable organization, administrative policies, and sufficient competent resources have been or will be established in a manner that will allow for the design, construction, and safe operation of the facility.

2.5 References

(AES, 2010a) AREVA Enrichment Services LLC. "Eagle Rock Enrichment Facility Safety Analysis Report," 2010.

(AES, 2010b) AREVA Enrichment Services LLC. "Eagle Rock Enrichment Facility Quality Assurance Program, Revision 2," 2009.

(NRC, 2002) U.S. Nuclear Regulatory Commission. NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," 2002.

CHAPTER 3.0 INTEGRATED SAFETY ANALYIS (ISA) AND ISA SUMMARY

The purpose of the U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's Integrated Safety Analysis (ISA) and ISA Summary is to evaluate whether the application meets the regulatory requirements specified in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 70, Subpart H, "Additional Requirements for Certain Licensees Authorized to Possess a Critical Mass of Special Nuclear Material." The review determines whether appropriate hazards and baseline design criteria have been addressed through independent analysis or qualitative evaluations of accepted engineering practices. The review also determines whether acceptable items relied on for safety (IROFS), management measures, and likelihoods and consequences have been designated for higher-risk accident sequences and whether, with IROFS, the performance requirements of 10 CFR 70.61 have been met.

In particular, the review as described in this chapter considered information provided by the applicant (that is not security-related nor export controlled information) that is related to:

- Commitments regarding the applicant's safety program, including the ISA, pursuant to the requirements of 10 CFR 70.62; and

- The ISA Summary submitted in accordance with 10 CFR 70.62(c)(3)(ii) and 70.65.

The staff's evaluation of the use of baseline design criteria (BDC) for the design of the facility in accordance with 10 CFR 70.64(a), the applicants defense-in-depth practices in accordance with 10 CFR 70.64(b), and those sections of the ISA Summary which contain security-related or export controlled information is discussed in Appendix A of this SER.

3.1 Regulatory Requirements

The following regulatory requirements are applicable to the ISA and ISA Summary content:

- 10 CFR 70.62 specifies the requirement to establish and maintain a safety program, including performance of an ISA that demonstrates compliance with the performance requirements of 10 CFR 70.61;

- 10 CFR 70.62(c) specifies requirements for conducting an ISA, including a demonstration that credible high-consequence and intermediate-consequence events meet the safety performance requirements of 10 CFR 70.61;

- 10 CFR 70.64 specifies requirements for baseline design criteria and facility and system design and facility layout; and

- 10 CFR 70.65(b) describes the contents of an ISA Summary.

The regulations in 10 CFR 70.62 require an applicant to establish and maintain a safety program that demonstrates compliance with the performance requirements of 10 CFR 70.61.

The safety program is required to contain: (1) process safety information, (2) an ISA, and (3) management measures. The ISA must be conducted and maintained by the applicant and must identify the following, in accordance with 10 CFR 70.62(c):

- Radiological hazards related to possessing or processing licensed material at the facility;

- Chemical hazards of licensed material and hazardous chemicals produced from licensed material;

- Facility hazards that could affect the safety of licensed materials and thus present an increased radiological risk;

- Potential accident sequences caused by process deviations or other events internal to the facility and credible external events, including natural phenomena;

- Consequences and likelihood of occurrence of each potential accident sequence identified and the methods used to determine the consequences and likelihoods; and

- Each IROFS identified pursuant to 10 CFR 70.61(e), the characteristics of its preventative, mitigative, or other safety function, and the assumptions and conditions under which the item is relied upon to support compliance with the performance requirements of 10 CFR 70.61.

The regulations, in 10 CFR 70.61, provide that the ISA must evaluate compliance with performance requirements. The requirements in 10 CFR 70.61(b) specify that the risk of each credible, high-consequence event must be limited such that the likelihood of occurrence is highly unlikely; and the requirements in 10 CFR 70.61(c) specify that the risk of each credible, intermediate-consequence event must be limited such that the likelihood of occurrence is unlikely.

The license application must include a description of the safety program under 10 CFR 70.65(a). In addition, the applicant is required to submit to the NRC an ISA Summary. As outlined in 10 CFR 70.65(b), the ISA Summary is required to contain:

- A general description of the site with emphasis on those factors that could affect safety;

- A general description of the facility with emphasis on those areas that could affect safety, including an identification of the controlled area boundaries;

- A description of each process analyzed in the ISA in sufficient detail to understand the theory of operation; and, for each process, the hazards that were identified in the ISA pursuant to § 70.62(c)(1)(i)-(iii) and a general description of the types of accident sequences;

- Information that demonstrates the licensee's compliance with the performance requirements of 10 CFR 70.61, including a description of the management measures; the requirements for criticality monitoring and alarms in 10 CFR 70.24; and, if applicable, the requirements of 10 CFR 70.64;

- A description of the team, qualifications, and the methods used to perform the ISA;
- A list briefly describing each IROFS identified pursuant to § 70.61(e) in sufficient detail to understand their functions in relation to the performance requirements of 10 CFR 70.61;
- 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 licensed materials which are on-site, or expected to be on-site as described in § 70.61(b)(4) and (c)(4);
- A descriptive list that identifies all IROFS that are the sole item preventing or mitigating an accident sequence that exceeds the performance requirements of 10 CFR 70.61; and
- A description of the definitions of unlikely, highly unlikely, and credible, as used in the evaluations in the ISA.

3.2 Regulatory Guidance And Acceptance Criteria

The guidance applicable to the NRC's review of the applicant's ISA and ISA Summary (AES, 2010b) is contained in Chapter 3 of "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 (NRC, 2002). These sections are applicable in their entirety with three exceptions. The first exception is 3.4.3.2(4)(c) which addresses criticality monitoring because criticality monitoring is addressed in Chapter 5 of this SER. The second exception is Section 3.4.3.2(5)(b)(i-ix) regarding process hazard analysis methods. This section provides conditions which should be met for hazard analysis methods used by the applicant if the methods are not described in NUREG-1513 (NRC, 2001). Because the method used by the applicant, the hazard and operability method (HAZOP), is described in NUREG-1513 (NRC, 2001), these conditions in Section 3.4.3.2(5)b(i-ix) do not have to be addressed. The third exception is various subparts in Section 3.4.3.2(9) regarding qualitative methods of defining and evaluating likelihood because the applicant uses a quantitative method. The acceptance criteria applicable to this review are contained in Sections 3.4.3.1 and 3.4.3.2 of NUREG-1520 (NRC, 2002).

3.3 Staff Review and Analysis

This section contains the staff's programmatic review of the applicant's proposed safety program, the proposed ISA commitments, proposed ISA methodology, proposed BDC, and proposed defense-in-depth. The staff's review of sensitive information, including export controlled information provided in the ISA and ISA Summary (AES, 2010b) is found in Appendix A to this SER. The staff's review of other information in the ISA, as determined from onsite reviews, is provided in Appendix A, Section A.3.2, of this SER.

3.3.1 Safety Program and ISA Commitments

The staff reviewed the applicant's proposed safety program commitments identified in Section 3.1 and Chapter 11 of the Safety Analysis Report (SAR) (AES, 2010a) to determine whether the three elements of process safety information, the ISA, and management measures demonstrate compliance with the requirements of 10 CFR 70.62; and that records will be established and maintained for documenting each discovery that an IROFS or management measure has failed or degraded such that it cannot perform its intended safety function.

3.3.1.1 Process Safety information

According to Section 3.0.1 of the SAR (AES, 2010a), the applicant has compiled and maintains process information addressing:

- The hazards of materials used or produced in the process — including information on chemical and physical properties (e.g., toxicity, acute exposure limits, reactivity, and chemical and thermal stability);

- The description of the technology of the process — including block flow diagrams or simplified process flow diagrams, a brief outline of process chemistry, safe upper and lower limits for controlled parameters (e.g., temperature, pressure, flow, and concentration), and evaluation of the health and safety consequences of process deviations; and

- Equipment used in the process, which includes general information on topics such as the materials of construction, piping and instrumentation diagrams, ventilation, design codes and standards employed, material and energy balances, IROFS, electrical classification, and relief system design.

The process-safety information described above will be maintained up-to-date by the configuration management program described in Section 11.1 of the SAR, "Configuration Management." As discussed in Section 11.3.1 of this SER, the applicant uses its configuration management system to control documentation and review design changes.

AES has developed procedures and criteria for changing the ISA. These include implementation of a facility change mechanism that meets the requirements of 10 CFR 70.72. The development and implementation of procedures is described in Section 11.4 of the SAR, "Procedures Development and Implementation" and is evaluated in Section 11.3.4 of this SER.

Therefore, the staff concludes that the above-mentioned program elements provide reasonable assurance of the following:

1. Consistent with Section 3.4.3.1(1)(a) of NUREG-1520 (NRC, 2002), the SAR (AES, 2010a) contains commitments to compile and maintain an up-to-date database of process safety information and is, therefore, acceptable, and

2. Consistent with Section 3.4.3.1(1)(b) of NUREG-1520 (NRC, 2002), the process safety element of the applicant's safety program includes procedures and criteria for changing the ISA, along with a commitment to design and implement a facility change mechanism, and is, therefore, acceptable.

As described in Section 3.2 of the SAR, AES uses personnel with expertise in engineering, safety analysis, and enrichment process operations and experience (individually or collectively) in nuclear criticality safety, radiological safety, fire safety, chemical safety, operations and maintenance, and ISA methods to maintain the ISA. The ISA Team for the various processes consists of individuals who are knowledgeable in the ISA method(s) and the operation, hazards, and safety design criteria of the particular process. The qualifications of the ISA team are evaluated in Section A.3.1.5 of this SER. Training and qualifications of individuals responsible for maintaining the ISA are described in the following sections of the SAR: Section 11.3, "Training and Qualifications" (and evaluated in Section 11.3.3 of this SER), Section 2.2, "Key

Management Positions," and Section 3.2, "Integrated Safety Analysis Team." Therefore, the staff concludes that consistent with Section 3.4.3.1(1)(c) of NUREG-1520 (NRC, 2002), the process safety element of the applicant's safety program contains a commitment to engage personnel with appropriate experience and expertise in engineering and process operations, and is, therefore, acceptable.

3.3.1.2 ISA Commitments

In Section 3.0.2 of the SAR (AES, 2010a), the applicant identifies ISA program elements that were used to establish the ISA process. Those elements include the performance of an ISA for each process that identifies the radiological hazards, chemical hazards that could increase radiological risk, chemical hazards from materials involved in processing licensed material, facility hazards that could increase radiological risk, potential accident sequences, consequences and likelihood of each accident sequence, and IROFS including the assumptions and conditions under which they support compliance with the performance requirements of 10 CFR 70.61. The staff's evaluation of the applicant's methods and criteria for implementing the ISA methodology is contained in Section 3.3.2 of this SER.

AES has implemented programs to maintain the ISA and supporting documentation so that it is accurate and up-to-date. Changes to the ISA Summary will be submitted to the NRC, in accordance with 10 CFR 70.72(d)(1) and (3). The ISA update process accounts for any changes made to the facility or its processes.

Personnel used to update and maintain the ISA and ISA Summary (AES, 2010b) are trained in the ISA methods(s). The general training and qualification of personnel used to update or maintain the ISA are described in Section 11.3, "Training and Qualification," of this SER. The ISA Summary (a non-public document) contains a list of the ISA team members, their areas of expertise, qualifications, and experience.

Proposed changes to the facility or its operations are evaluated by the ISA method(s) described in the ISA Summary. New or additional IROFS and appropriate management measures are designated as required. The adequacy of existing IROFS and associated management measures are promptly evaluated to determine if they are impacted by changes to the facility and/or its processes.

Unacceptable performance deficiencies associated with IROFS will be addressed and identified through updates to the ISA.

Written procedures will be maintained onsite. Section 11.4 of the SAR discusses the applicant's procedures program.

All IROFS will be maintained so that they are available and reliable when needed.

Items A & B of Section 3.0.2 of the SAR (AES, 2010a) contain commitments to conduct an ISA of appropriate complexity for each process. These commitments are consistent with Section 3.4.3.1(2)(a) of NUREG-1520 (NRC, 2002).

Item B of Section 3.0.2 of the SAR (AES, 2010a) contains a commitment to maintain the ISA and its supporting documentation. This commitment is consistent with Section 3.4.3.1(2)(b) of NUREG-1520 (NRC, 2002).

Item C of Section 3.0.2 of the SAR (AES, 2010a) contains a commitment to train personnel in the facility's ISA methods and/or use suitably qualified personnel to update and maintain the ISA and ISA Summary (AES, 2010b). This commitment is consistent with Section 3.4.3.1(2)(c) of NUREG-1520 (NRC, 2002).

Item D of Section 3.0.2 of the SAR (AES, 2010a) contains a commitment to evaluate proposed changes to the facility or its operations by means of the ISA method(s) and to designate new or additional IROFS and appropriate management measures. This commitment is consistent with Section 3.4.3.1(2)(d) of NUREG-1520 (NRC, 2002).

Item E of Section 3.0.2 of the SAR (AES, 2010a) contains a commitment to address any IROFS' unacceptable performance deficiencies that are identified through updates to the ISA. This commitment is consistent with Section 3.4.3.1(2)(e) of NUREG-1520 (NRC, 2002).

Item F of Section 3.0.2 of the SAR (AES, 2010a) contains a commitment to maintain written procedures onsite. This commitment is consistent with Section 3.4.3.1(2)(f) of NUREG-1520 (NRC, 2002).

Item G of Section 3.0.2 of the SAR (AES, 2010a) contains a commitment to establish all IROFS (if not already established) and to maintain them so that they are available and reliable when needed. This commitment is consistent with Section 3.4.3.1(2)(g) of NUREG-1520 (NRC, 2002).

3.3.1.3 Management Measures

In Section 3.0.3 of the SAR (AES, 2010a), the applicant describes management measures that comprise the principal mechanism by which the reliability and availability of each IROFS is ensured. General requirements applicable to each IROFS for configuration management, maintenance, training and qualifications, procedures, audits and assessments, incident investigation, records management, and other quality assurance elements are discussed. Any management measures deviating from these general requirements, which are consistent with the performance requirements assumed in the ISA documentation, are discussed in the ISA Summary (AES, 2010b). Incident investigations are conducted within the applicant's Corrective Action Program. Incidents associated with IROFS, and any items that may affect the function of IROFS, include: processes that behave in unexpected ways, procedural activities not performed in accordance with an approved procedure; discovered deficiency, degradation, or non-conformance of an IROFS; or any items that may affect the function of IROFS. Feedback from the results of incident investigations are used, as appropriate, to modify management measures to provide continued assurance of the availability and reliability of IROFS, to meet the performance requirements of 10 CFR 70.61. All records associated with IROFS, and any item that may affect the function of IROFS, will be managed and controlled in a systematic manner to provide identifiable and retrievable documentation. The management measures are further detailed in Chapter 11 of the SAR (AES, 2010a), and evaluated in Chapter 11 of this SER.

3.3.1.4 Safety Program and ISA Commitments Conclusion

Based on the evaluations in Sections 3.3.1 through 3.3.3, above, the staff concludes that the applicant meets the requirements of 10 CFR 70.62(a)(1) through (3) to establish and maintain a safety program that includes process safety information, integrated safety analysis, management measures, and appropriate safety program records. The staff also concludes that the applicant has an appropriate program to establish and maintain records of IROFS failures that will be retrievable for NRC inspection.

3.3.2 ISA Methodology

The ISA methodology is described in Section 3.1 of the SAR (AES, 2010a) and Section 3.1.1 of the ISA Summary (AES, 2010b). The applicant described the approach used for performing the ISA as a semi-quantitative risk index method to categorize accident sequences in terms of their likelihood of occurrence and their consequences of concern. Further, the applicant stated that the risk index method identifies which accident sequences have consequences that could exceed the performance requirements of 10 CFR 70.61 and, therefore, require designation of IROFS and supporting management measures.

3.3.2.1 Hazard Identification and Evaluation

The applicant used the HAZOP to identify potential hazards related to uranium hexafluoride (UF_6) process systems and Technical Support Building (TSB) systems. The HAZOP method is a structured technique commonly used in the chemical industry that is well suited to analyze processes during or after a detailed design stage. The method uses an interdisciplinary team to identify hazards and operability problems resulting from deviations from the process's design intent that could lead to undesirable consequences which do not meet the 10 CFR 70.61 performance requirements.

The HAZOP method is widely used in the chemical processing industry because it is suitable for performing a detailed analysis of a wide range of hazards to identify potential accident sequences. NUREG-1513, "Integrated Safety Analysis Guidance Document," Appendix A, "Flowchart for Selecting a Hazards Analysis Technique" (NRC, 2001), identifies the HAZOP technique as an acceptable approach. Therefore, the staff concludes that the process hazard analysis method used by the applicant is acceptable for the identification of potential radiological, chemical and facility hazards, and potential accident sequences caused by process deviations; or other events internal to the facility and credible external events, including natural phenomena that could lead to a loss of UF_6 confinement or a criticality.

As stated in Section 3.1 of the SAR (AES, 2010a), the hazard identification process used by the applicant documented materials that are radioactive, fissile, flammable, explosive, toxic, and reactive; and identified potentially hazardous conditions. Hazards were assessed individually for the potential impact on the process systems (e.g., UF_6 feed system). However, hazards related to fires and external events were assessed on a facility-wide basis (Section 3.1 of the SAR). The Fire Hazard Analysis (discussed in Section 7.2 of the SAR [AES, 2010a]) was consulted in order to place reasonable and conservative bounds on the fire scenarios. External events evaluated included seismic, tornado, tornado missile and high wind, snow and ice, flooding, local precipitation, transportation and nearby facility accidents, aircraft, pipelines, highway, railroad, and internal flooding from above-ground storage tanks. The facility wide assessment resulted in natural phenomena events being assessed against all structures without regard to location or design differences and fires were assessed by plant area (or fire area) and included all possible fire hazards within the area. These assessments by the applicant are evaluated by the staff in the discipline related chapters of the SER (primarily, Chapters 5, 6, and 7 and Appendices A, B, D, E, F, and G).

As described in Section 3.1 of the SAR, the applicant gave special consideration to common mode failures and common cause situations, support system failures, divergent impacts of IROFS, non-IROFS impacts on system performance, multiple impact scenarios, system interactions and interdependence, and major hazards or events which tend to be common-cause situations that could lead to interactions between processes, systems, or buildings.

Chapter 6, Tables 6.1-1 through 6.1-6, of the SAR (AES, 2010a) identifies the hazardous properties of all chemicals used onsite and the inventories and locations of chemicals of concern (including UF_6). Potential interactions involving UF_6 and any reaction products are identified in Section 6.1.2 of the SAR (AES, 2010a). These chemicals include hydrogen fluoride (HF) and uranyl fluoride (UO_2F_2). This section also identifies the physical properties, reactivity, toxicological properties, and flammability of these chemicals. Chemical reactions and interactions involving UF_6 and water, Fomblin oil, chemical trap materials, and other materials used in the process are described in Section 6.2.1 of the SAR (AES, 2010a). Section 7.2 of the SAR (AES, 2010a) describes the Fire Hazards Analysis for those facility areas containing licensed material. Chapter 3 of the ISA Summary (AES, 2010b) identifies the external hazards. The applicant identified either a loss of confinement (of UF_6) or a criticality as the hazard of concern in Section 3.1.4 of the ISA Summary (AES, 2010b). Potential accident sequences that could result in an UF_6 release or criticality of high or intermediate consequence are discussed in detail in Section 3.7 of the ISA Summary (AES, 2010b).

Table 3.7-1, Accident Sequence and Risk Index, of the ISA Summary (AES, 2010b) lists the potential accident sequences that were identified that could have consequences that exceed the performance requirements of 10 CFR 70.61. Such sequences could be caused by external events, facility events external to the process being analyzed, deviations from normal operations, and failures of IROFS. In this list, the applicant demonstrates with the application of IROFS that high-consequence accident sequences are highly unlikely and intermediate-consequence accident sequences are unlikely.

In Section 3.1.3 of the SAR (AES, 2010a), the applicant describes the consequence verses likelihood risk matrix. The risk matrix and computed index values are shown in Table 3.1-6 of the SAR (AES, 2010a) with the likelihood categories across the top and the consequence categories along the left side. The risk matrix shows the combinations of likelihood and risk that are unacceptable. Sequences that fall into these combinations will be mitigated or prevented with IROFS.

The risk index evaluation process as described in Section 3.1.4 of the SAR cross references an accident sequence consequence with its likelihood to first determine the total risk, and then to demonstrate that with the application of the selected IROFS, the performance requirements of 10 CFR 70.61 are met. For sequences that place the system in a vulnerable state, the duration of the vulnerable state is considered, and a duration index assigned (Section 3.1.4 of the SAR [AES, 2010a]). The values of all index numbers in a sequence are added to obtain a total likelihood index of T (Section 3.1.4 of the SAR [AES, 2010a]). Accident sequences are then assigned to one of three likelihood categories, depending on the value of the index in accordance with Table 3.1-8 of the SAR (AREVA, 2010a). The criteria of Tables 3.1-9 through 3.1-11 of the SAR (Section 3.1.4 of the SAR [AES, 2010a]) are used to assign index numbers to accident sequences. The staff reviewed the risk index evaluation for selected accident scenarios in the fire, chemical, and criticality safety areas as part of the "vertical-slice" review and determined that the applicant's evaluation process was appropriately applied and consistent with the methodology described in Appendix A of Chapter 3 of NUREG 1520 (NRC, 2002). The staff concludes that the risk evaluation methodology described by the applicant is consistent with the guidance in NUREG-1520 (NRC, 2002a) and suitable for determining which accident sequences require IROFS and the level of risk reduction provided the IROFS to comply with 10 CFR 70.61.

Consequence Analysis Method

Consequence analysis methods for determining the chemotoxic exposure to HF and UO_2F_2 are discussed in Section 6.3.2 of the SAR (AES, 2010a), The radiological and chemical consequence severity levels are provided in SAR Table 3.1-3 (AES, 2010a). Information on the chemical dose limits specific to the EREF is found in Table 3.1-4 of the SAR (AES, 2010a). The applicant developed credible accident scenarios and the dispersion analysis and chemical/radiological dose assessment associated with each accident sequence in accordance with NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook" (NRC, 1998). The consequences of an inadvertent criticality were conservatively assumed to be high for both the public and the workers by the applicant because of the potential high radiation exposure to an individual.

The staff finds that the applicant's method for consequence determination is consistent with the guidance described in NUREG-1513 (NRC, 2001) and NUREG-1520 (NRC, 2002a), and is, therefore, acceptable. Further evaluation of chemical and radiological consequences is found in Chapter 6 and Appendix B of this SER.

Consequence Categories

Accident sequences identified by the applicant as a result of the process hazards analysis are categorized as either as a high-consequence event or an intermediate-consequence event, or are considered a low consequence event in accordance with the performance criteria of 10 CFR 70.61. Table 3.1-3 of the SAR (AES, 2010a) identifies the consequence severity categories for workers, offsite public and the environment for radiation and chemical doses as defined by the applicant. The values proposed by the applicant for both radiological dose and chemical dose are consistent with the guidance in NUREG-1520, Appendix A, Table A-5 (NRC, 2002a). All of the identified high-consequence and intermediate-consequence events are listed in the ISA Summary (AES, 2010b) and are comprised of sequences caused by external events, facility events external to the process being analyzed, deviations from normal operations, and failures of IROFS.

Safe-by-Design IROFS

The SAR defines a special class of nuclear criticality safety (NCS) controls as safe-by-design IROFS. As defined in SER Section 5.3.8.1:

" A safe-by-design IROFS is a passive engineered control which — by its geometry and configuration alone — will prevent a criticality accident from occurring. This means that all NCS parameters, except geometry and interaction, are assumed to be in the optimum or worst case credible condition. By definition, it is considered to be highly unlikely for a safe-by-design IROFS to fail in a manner that would cause a criticality accident. The most significant requirements to qualify as a safe-by-design IROFS are that:

1. The only credible failure mechanism that could result in a criticality would be to implement an improper design change.
2. Credible process deviations or events do not adversely impact performance of the safety function.
3. Quality Assurance Level 1 is applied to the feature.
4. No human actions are required for the component to perform its safety function."

The failure of a safe-by-design IROFS due to a loss of configuration control is considered to be an initiating event with an index of -5 (SER Section 5.3.9). The staff determined that a safe-by-design IROFS would be considered failed if it were installed, approved for use, and an adequate margin of subcriticality (i.e., $k_{eff} < 0.95$) would not be maintained for all credible process deviations or events. It would require multiple human errors in the configuration management process for such a failure to occur, which is sufficient to accept that this event is highly unlikely.

Safe-by-design IROFS must also have a "significant margin," as defined in the SAR. The staff determined that EREF's application of significant margin does not provide any additional benefit to safety, and it is not relied upon to make any conclusions for this review.

The staff concludes that components that meet the safe-by-design criteria will be capable of: (1) preventing a criticality accident under normal and credible abnormal conditions, (2) maintaining an adequate margin of subcriticality for safety, and (3) ensuring that a criticality accident is highly unlikely to occur (SER Section 5.3.9).

3.3.2.2 Definition of Receptors for Consequence Evaluations

The receptors considered for the consequence severity categories as listed in Table 3.1-3 of the SAR are:

- Workers - individuals inside or outside of facility buildings within the controlled area boundary

- Offsite Public - individuals at the controlled area boundary nearest to the release point

- Environment - a point at the restricted area boundary nearest to the point of release for which a 24-hour average concentration is calculated

The boundaries of the controlled and restricted areas are provided in Figure 1.1-3 of the SAR. For workers and the offsite public, both high and intermediate consequence categories may be applicable. For the environment, only the intermediate-level consequence category is applicable.

3.3.2.3 *Likelihood Evaluation Method*

Section 3.1.3.2 of the SAR (AES, 2010a) discusses the definitions of "Not Unlikely," "Unlikely," and "Highly Unlikely," and SAR Table 3.1-5 (AES, 2010a), cross references the three likelihood categories with a probability of occurrence based on approximate order of magnitude ranges. The proposed values, ranging from less than 10^{-5} events per year for "highly unlikely" to more than 10^{-4} events per year for "not unlikely",are consistent with NUREG-1520, Chapter 3, Appendix A, Table A-6 (NRC, 2002a). The ISA Summary Tables 3.7-1 through 3.7-4 (AES, 2010b), show how each designated IROFS acts to prevent or mitigate the consequences of an accident sequence. When multiple IROFS are designated, redundant systems will be separate and independent from each other, as stated in Section 3.8.1 of the ISA Summary (AES, 2010b).

The likelihood of failure was qualitatively evaluated for each IROFS, often based on the operational history of similar facilities. Each sequence was evaluated as "Not Unlikely," "Unlikely," or "Highly Unlikely."

3.3.2.4 *Chemical Consequences*

This section evaluates the proposed chemical quantitative risk levels to protect workers and the public. Radiological consequence limits are specified in 10 CFR 70.61(b) and (c). However, chemical consequence limits are not specified in the regulation but are required to be proposed by the applicant and described in the ISA Summary in accordance with 10 CFR 70.65(b)(7). The applicant's proposed chemical consequence levels for HF and soluble uranium are proposed in the SAR (AES, 2010a) and the ISA Summary (AES, 2010b), and were evaluated by the staff, as discussed below.

The proposed chemical consequences were classified as high or intermediate based on the performance requirements contained in 10 CFR 70.61. Low consequences are not defined in 10 CFR 70.61. However, the applicant defined low consequence accident scenarios as events that do not exceed the intermediate threshold values, that is, those sequences that do not require controls or IROFS to meet 10 CFR 70.61. This definition is acceptable to the staff. The applicant proposed the Acute Exposure Guideline Levels (AEGLs) values, established by the National Advisory Committee for Acute Guideline Levels for Hazardous Substances, as the quantitative standard used to assess the consequences to an individual from acute chemical exposure to licensed material or chemicals produced from licensed materials which are onsite or expected to be onsite. The staff finds that the use of this standard is consistent with page 3-23 of NUREG-1520 (NRC, 2002) which allows the use of values from national and international accepted standards.

The applicant defined the AEGLs as follows:

AEGL-1 (non-disabling) "The airborne concentration of a substance above which it is predicted that the general population, including susceptible individuals, could experience notable discomfort, irritation or certain asymptomatic effects. However, the effects are not disabling and are transient and reversible upon cessation of exposure."

AEGL-2 (disabling) "The airborne concentration of a substance above which it is predicted that the general population, including susceptible individuals, could experience irreversible or other serious, long-lasting adverse health effects, or an impaired ability to escape."

AEGL-3 (lethality) "The airborne concentration of a substance above which it is predicted that the general population, including susceptible individuals, could experience life-threatening health effects or death."

The comparison of the AEGLs with the calculated airborne concentrations allows the applicant to classify a high, intermediate, or low chemical consequence level. In the case of exposures to soluble forms of uranium (e.g., UF_6 and UO_2F_2), the applicant uses intakes of uranium expressed in milligrams (mg) of material. As part of the chemical consequence methodology as presented in Tables 3.1.3 and 3.1.4 of the SAR (AES, 2010a), the applicant states that 21 mg of soluble uranium (U) intake[4] would be the high consequence limit for an individual at the controlled area boundary instead of 30 mg of soluble uranium intake as required in 10 CFR 70.61(b)(3). The high-consequence limit taken by the applicant is more conservative than the value in the 10 CFR 70.61(b)(3); therefore, it is acceptable. The intermediate-consequence limit for an individual at the controlled area boundary is 4.06 mg of soluble U intake. The applicant indicates that the soluble U intake limits do not apply to the worker because the worker is more conservatively protected by the UF_6 AEGL limits. The NRC staff reviewed the calculation to support this statement and finds it acceptable. The soluble U intake limits are taken from NUREG-1391 (NRC, 1991). The AEGL and the soluble U intake limits are shown in Table 3.3.-1.

Table 3.3 - 1 Consequence Level as Related to AEGL Values and Soluble Uranium Intake

Consequence Level	Worker	Individual at the Controlled Area Boundary
High	[HF] > AEGL-3 [UF_6] > AEGL-3	[HF] > AEGL-2 Soluble U intake ≥ 21 mg
Intermediate	AEGL-2 < [HF] ≤ AEGL-3 AEGL-2 < [UF_6] ≤ AEGL-3	AEGL-1 < [HF] ≤ AEGL-2 4.06 mg ≤ Soluble U intake < 21 mg
Low	[HF] ≤ AEGL-2 [UF_6] ≤ AEGL-2	[HF] ≤ AEGL-1 Soluble U intake < 4.06 mg

Note: [HF] and [UF_6] are the concentrations of HF and UF_6 as given by the AEGL value (i.e., mg/m^3) (not molar concentration (i.e., mol/L)), respectively.

The use of the AEGL standard allows the applicant to vary the exposure duration. The chemical consequence levels proposed by the applicant are shown in Table 3-2, for HF, and Table 3-3, for UF_6.

[4]The applicant actually states a 21-mg body burden (i.e., the amount of uranium that stays in the body), but this limit does not take credit for the loss of uranium through exhalation. Therefore, it is appropriate to equate the 21-mg body burden to 21 mg of soluble U intake.

Table 3.3-2 Chemical Consequence Levels Proposed by the Applicant for Hydrogen Fluoride

Exposure Duration	AEGL-1 / (mg/m^3)	AEGL-2 / (mg/m^3)	AEGL-3 / (mg/m^3)
10 min	0.8	78	139
30 min	0.8	28	51
1 hour	0.8	20	36
4 hours	0.8	9.8	18
8 hours	0.8	7	12

Table 3.3-3 Chemical Consequence Levels Proposed by the Applicant for Uranium Hexafluoride

Exposure Duration	AEGL-1 (mg/m^3)	AEGL-2 (mg/m^3)	AEGL-3 (mg/m^3)
10 min	3.6	28	216
30 min	3.6	19	72
1 hour	3.6	9.6	36
4 hours	-	2.4	9
8 hours	-	1.2	4.5

Consistent with Section 3.4.3.2(7) of NUREG-1520 (NRC, 2002), the applicant's quantitative standards for chemical consequences meet the following criteria and, therefore, are acceptable:

- There are unambiguous, quantitative standards for each of the applicable hazardous chemicals that meet the criteria of 10 CFR 70.65(b)(7) onsite;
- The quantitative standard of 10 CFR 70.61(b)(4)(i) addresses exposures that could endanger the life of a worker;
- The quantitative standards for 10 CFR 70.61(b)(4)(ii) and 10 CFR 70.61(c)(4)(i) will correctly categorize all exposures that could lead to health effects on individuals that could be long lasting; and
- The quantitative standard for 10 CFR 70.61(c)(4)(ii) will correctly categorize all exposures that could cause individual mild transient health effects.

3.3.2.5 Radiological Consequences

Radiological consequences limits are specified in 10 CFR 70.61(b) and (c), and are presented along with chemical consequences in Table 3.1-3 of the SAR (AES, 2010a). As presented below, the applicants radiological limits are the same as those specified in 10 CFR 70.61 and are acceptable.

Table 3.3-4 Radiological Consequence Levels Proposed by the Applicant

Consequence level	Worker	Individual at the controlled area boundary
High	≥ 100 rem (1 Sv)	≥ 25 rem (0.25 Sv)
Intermediate	≥ 25 rem (.25 Sv)	≥ 5 rem (.05 Sv)
Low	< 25 rem (.25 Sv)	< 5 rem (.05 Sv)

3.3.2.6 Environmental Consequences

The performance requirements of 10 CFR 70.61(c)(3) specify that the environmental consequences of each credible, intermediate-consequence event must be limited so that the 24-hour averaged release concentration of U outside the restricted area is less than 5,000 times the value specified in 10 CFR 20, Appendix B, Table 2. Appendix E of the ISA Summary (AES, 2010b) presents the methodology that was used to evaluate the consequences of hypothesized accidental releases of UF_6, which is based on NUREG/CR-6410 (Nuclear Fuel Cycle Facility Accident Analysis Handbook) and NUREG/CR-5659 (Control Room Habitability System Review Models). The NRC staff reviewed the applicant's methodology for determining compliance with the environmental consequences requirement and finds it to be acceptable. As discussed in Section 9.3.3 of this SER, the NRC staff determined that environmental consequences may occur only if uncontrolled, intermediate or high consequences to workers are also present. The NRC staff did not identify any accident sequence that would fail to meet the environmental performance requirements of 10 CFR 70.61(c)(3). The NRC staff also made confirmatory calculations for a subset of potential accident scenarios as discussed in the non-public Appendix B of this SER.

3.3.2.7 ISA Methodology Conclusion

Based on the above information, the staff concludes that the applicant has used a methodology consistent with NUREG-1513 (NRC, 2001) to identify hazards related to this type of facility and credible events that could exceed the performance requirements of 10 CFR 70.61. The applicant has also established definitions of likelihood consistent with Section 3.4.3.2 of NUREG-1520 and applied those definitions to demonstrate that intermediate-consequence events are unlikely and high-consequence events are highly unlikely.

3.4 Evaluation

The staff finds that the applicant's maintenance of process safety information is in accordance with the guidance of NUREG-1520 (NRC, 2002) (described in Section 3.3.1.1 of this SER).

The staff finds that the applicant's commitment to conduct and maintain an ISA is in accordance with the requirements of 10 CFR 70.62(c)(1) and the guidance in NUREG-1520 (NRC, 2002). (described in Sections 3.3.1.1 and 3.3.1.2 of this SER).

The staff considers the ISA methodology to be complete by its use of the appropriate accident identification methodology from NUREG-1513 (NRC, 2001). The staff considers the consequence determinations to be acceptable and in accordance with the guidance in NUREG/CR-6410 (NRC, 1998). The staff has also evaluated the consequence determination methodology, and the staff's evaluation is presented in Appendix B of this SER. The staff considers the likelihoods to have been derived using acceptable methods and to comply with acceptable definitions of "not unlikely," "unlikely," and "highly unlikely" as evaluated in Appendix A of this SER. The staff concludes that these descriptions (described in Section 3.3.2 of this SER) conform with the guidance provided in NUREG-1520 (NRC, 2002), and meet the requirements of 10 CFR 70.65(b)(4).

Area boundaries, including the controlled area boundary and the locations of restricted areas, are adequately described for the purpose of determining consequences (described Section 3.3.2.2 of this SER) and meet the requirements of 10 CFR 70.65(b)(2).

The likelihood evaluation, including the definition of "not credible," conforms to the guidance provided in NUREG-1520 (NRC, 2002) and is acceptable (described in Section 3.3.2.3 of this SER).

The determination of chemical consequences (described in Section 3.3.2.4 of this SER) conforms to the guidance provided in NUREG-1520 (NRC, 2002) and is acceptable.

3.5 References

(AES, 2010a) AREVA Enrichment Services LLC, "Eagle Rock Enrichment Facility, Safety Analysis Report," 2010.

(AES, 2010b) AREVA Enrichment Services LLC, "Integrated Safety Analysis Summary for the Eagle Rock Enrichment Facility," 2010.

(ANSI, 1974) American National Standards Institute/Institute of Electrical and Electronics Engineers, "IEEE Standard for Type Test of Class 1E Electrical Cables, Field Splices, and Connections for Nuclear Power Generating Stations," ANSI/IEEE-383-1974, 1974.

(NRC, 2002) U.S. Nuclear Regulatory Commission, NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," 2002.

(NRC, 2001) U.S. Nuclear Regulatory Commission, NUREG-1513, "Integrated Safety Analysis Guidance Document," 2001.

(NRC, 1998) U.S. Nuclear Regulatory Commission, NUREG/CR-6410, "Nuclear Fuel Cycle Accident Analysis Handbook," 1998.

(NRC, 1991) U.S. Nuclear Regulatory Commission, NUREG-1391, "Chemical Toxicity of Uranium Hexafluoride Compared to Acute Effects of Radiation," February 1991.

(NRC, 1990) U.S. Nuclear Regulatory Commission and Science Applications International Corporation, "Control Room Habitability System Review Models," December 1990.

CHAPTER 4.0 RADIATION PROTECTION

The purpose of this review is to determine whether the applicant's radiation protection (RP) program is adequate to protect the radiological health and safety of workers and to comply with the associated regulatory requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Parts 19, 20, 30, 40, and 70. Public and environmental protection is discussed in Chapter 9 of this Safety Evaluation Report (SER).

4.1 Regulatory Requirements

- Regulations applicable to the establishment of an RP program are presented in Part 20, Subpart B, "Radiation Protection Programs."

- Regulations applicable to the As Low As Is Reasonably Achievable (ALARA) program are presented in 10 CFR 20.1101, "Radiation Protection Programs."

- Regulations applicable to the organization and qualifications of the radiological protection staff are presented in 10 CFR 30.33, "General Requirements for Issuance of Specific Licenses;" 10 CFR 40.32, "General Requirements for Issuance of Specific Licenses;" and 10 CFR 70.22, "Contents of Applications."

- Regulations applicable to RP procedures and Radiation Work Permits (RWPs) are presented in 10 CFR 30.33; 10 CFR 40.32; and 10 CFR 70.22, "Contents of Applications."

- The following regulations apply to the radiation safety training program:

    1. 10 CFR 19.12 "Instructions to Workers"
    2. 10 CFR 20.2110 "Form of Records"

- Regulations applicable to the ventilation and respiratory protection programs are presented in 10 CFR Part 20, Subpart H, "Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas."

- Regulations in 10 CFR Part 20 are applicable to radiation surveys and monitoring programs:

    1. 10 CFR Part 20, Subpart F "Surveys and Monitoring"
    2. 10 CFR Part 20, Subpart C "Occupational Dose Limits"
    3. 10 CFR Part 20, Subpart L "Records"
    4. 10 CFR Part 20, Subpart M "Reports"

- The following NRC regulations are applicable to the additional program requirements:

    1. 10 CFR Part 20, Subpart L "Records"
    2. 10 CFR Part 20, Subpart M "Reports"
    3. 10 CFR 70.61 "Performance Requirements"
    4. 10 CFR 70.74 "Additional Reporting Requirements"

4.2 Regulatory Guidance and Acceptance Criteria

The acceptance criteria for the NRC staff's review of the RP program are outlined in Sections 4.4.1.3; 4.4.2.3; 4.4.3.3; 4.4.4.3; 4.4.5.3; 4.4.6.3; 4.4.7.3; and 4.4.8.3 of "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 (NRC, 2002).

4.3 Staff Review and Analysis

4.3.1 RP Program Implementation

The staff reviewed the applicant's RP program implementation against the acceptance criteria in Section 4.4.1.3 of NUREG-1520 (NRC, 2002). The following sections identify each acceptance criteria from NUREG-1520 and discuss the staff's analysis as to whether the information provided by the applicant in the Safety Analysis Report (SAR) (AES, 2009a) meets the criteria.

1. "[D]esign and implement a RP program that meets the regulatory requirements of 10 CFR Part 20, Subpart B"

   In Section 4.1 of the Eagle Rock Enrichment Facility (EREF) SAR (AES, 2009a), the applicant ensures that the RP program would meet the 10 CFR 20, Subpart B requirements by committing that the program would be consistent with the guidance provided in Regulatory Guide (RG) 8.2, "Guide for Administrative Practice in Radiation Monitoring (NRC, 1973)." The facility would develop, document, and implement its RP program commensurate with the risks posed by a uranium enrichment operation. The facility would use, to the extent practicable, procedures and engineering controls based upon sound RP principles to achieve occupational doses and doses to members of the public that are ALARA. Constraints on atmospheric releases would be established such that no member of the public would be expected to receive greater than 10 millirem per year (mrem/yr) total effective dose equivalent (TEDE) from exposure to the release. Also, the RP program content and implementation would be reviewed at least annually.

2. "[O]utline the RP program structure and define the responsibilities of key program personnel"

   The applicant outlined the RP program structure and defined the responsibilities of key program personnel in Section 4.1.1 of the EREF SAR (AES, 2009a). The AREVA Enrichment Services's (AES's) President would have overall responsibility for the operation of the EREF, including RP. The Plant Manager would report to the AES President and would be responsible for the protection of all persons against radiation exposure resulting from facility operations and materials and for compliance with applicable NRC regulations and the facility license. The Environmental, Health, Safety, and Licensing (EHS&L) Manager would report to the Plant Manager and would have overall responsibility for development and implementation of the RP program. The Radiation Protection/Chemistry Manager would report to the ESH&L Manager; would be responsible for implementing the RP program; and, in matters involving RP, would have direct access to the Plant Manager; and, would establish the RP program independent of operations.

The Operations Manager would report to the Plant Manager and would have the responsibility for the safe day-to-day operation of the facility including operating in accordance with procedures that incorporate RP practices. Facility personnel would also be required to work safely and follow rules, regulations, and procedures established for their protection and protection of the public.

3. "[S]taff the RP program with suitably trained people, provide sufficient resources, and implement the program"

   Staffing of the RP program is addressed in Sections 4.1.1.4 and 4.1.2 of the EREF SAR (AES, 2009a). The Radiation Protection/Chemistry Manager would be responsible to sufficiently staff the facility with suitably trained people to implement an effective RP program. The staff would be trained and qualified consistent with the guidance provided in the American National Standards Institute (ANSI)/American Nuclear Society (ANS) Standard N3.1-1993, "Selection, Qualification and Training of Personnel for Nuclear Power Plants" (ANSI/ANS, 1993).

4. "[C]ommit to the independence of the radiation protection function from the facility's operations"

   Section 4.1.3 of the EREF SAR (AES, 2009a) addresses the independence of the RP program. The RP program will be independent of operations because the Radiation Protection/Chemistry Manager has direct access to the Plant Manager, and the RP staff reports to the Radiation Protection/Chemistry Manager.

5. "[R]eview, at least annually, the content and implementation of the radiation protection program as required by 10 CFR 20.1101(c)"

   As previously stated, the applicant, in Section 4.1 of the SAR (AES, 2009a), commits to review, at least annually, the content and implementation of the RP program.

Based on the analysis as summarized above, the staff finds that the commitments in the EREF SAR (AES, 2009a) satisfactorily address the application acceptance criteria in Section 4.4.1.3 of NUREG-1520 (NRC, 2002).

4.3.2 ALARA Program

The staff reviewed the applicant's ALARA program commitments against the acceptance criteria in NUREG-1520 Section 4.4.2.3 (NRC, 2002). The following sections identify each acceptance criteria from NUREG-1520 and discuss the staff's analysis as to whether the information provided by the applicant in the SAR (AES, 2009a) meets the criteria.

1. "[E]stablish a comprehensive, effective, and written ALARA program"

   Section 4.2 of the EREF SAR addresses the proposed ALARA program. The applicant would implement an ALARA program with the objective to make every reasonable effort to maintain exposures to radiation as far below the regulatory dose limits as is practical. The program would be written and policies documented to ensure the ALARA goal is met. The design and implementation of the ALARA program would be consistent with RGs: 8.2, "Guide for Administrative Practice in Radiation Monitoring" (NRC, 1973a); 8.13 "Instruction Concerning Prenatal Radiation Exposure," (NRC, 1999); 8.29, "Instruction Concerning Risks from Occupational Radiation Exposure," (NRC, 1996); and 8.37 "ALARA Levels for Effluents from Materials Facilities," (NRC, 1993). The operation of the facility would be consistent with Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable," (NRC, 1977). Facility procedures would incorporate the ALARA philosophy into the routine facility operations. Also, the Radiation Protection/Chemical Manager would prepare an annual ALARA program evaluation report.

2. "[P]repare policies and procedures to ensure occupational exposures are maintained ALARA, and that such exposures are consistent with the requirements of 10 CFR 20.1101"

   As described in Section 4.2 of the EREF SAR (AES, 2009a), the applicant would maintain exposures to members of the public such that they are not expected to exceed the limits in 10 CFR 20.1101(d) which established a 10 mrem/yr dose constraint to the public from atmospheric releases.

3. "[O]utline specific ALARA program goals, establish an ALARA program organization and structure, and have written procedures for its implementation in the facility design and operations"

   Section 4.2 of the EREF SAR (AES, 2009a) commits to specific goals for the ALARA program such as maintaining occupational exposures (both individual and collective exposures), as well as environmental releases, as far below regulatory limits as is reasonably achievable. The Radiation Protection/Chemistry Manager would be responsible for implementing the ALARA program and ensuring adequate resources are available to make the program effective. As previously noted, the applicant has committed that facility procedures would incorporate the ALARA philosophy into the routine facility operations.

   The size and number of areas at the facility with higher dose rates would be minimized, consistent with accessibility for performing necessary services, while areas where facility personnel spend significant amounts of time would be designed to maintain the lowest dose rates reasonably achievable. Radiological zones, such as "Radiation Area," "Airborne Radioactivity Area," and "Contaminated Area," will be established to minimize the spread of contamination and reduce unnecessary radiation exposure to personnel.

4. "[E]stablish an ALARA Committee, or equivalent organization, with sufficient staff, resources, and clear responsibilities to ensure that the occupational radiation exposure dose limits specified in 10 CFR Part 20 are not exceeded under normal operations"

Sections 2.2.3 and 4.2.1 of the EREF SAR (AES, 2009a) include commitments to establish a Safety Review Committee (SRC) which would meet at least quarterly and within 30 days of any NRC-reportable incident and would fulfill the duties of the ALARA Committee. The SRC would be composed of at least five members including the Chairman. The five members would include experts on operations and all safety disciplines (criticality, radiological, chemical, and industrial). The Chairman, members, and alternate members, of the SRC would be formally appointed by the AES President. The SRC would conduct at least one facility audit per year that would review and audit the following subject areas as pertain to the facility.

- Radiation protection
- Nuclear criticality safety
- Hazardous chemical safety
- Industrial safety including fire protection
- Environmental protection
- ALARA policy implementation
- Changes in facility design or operations.

As part of its duties, the SRC would review the effectiveness of the ALARA program and determine if exposures, releases, and contamination levels are in accordance with the ALARA concept. It also would evaluate the results of assessments made by the RP organization, reports of facility radiation levels, and employee exposures for identified categories of workers and types of operations. The committee is responsible for ensuring that occupational radiation exposures do not exceed the dose limits in 10 CFR 20.1201 under normal operations.

5. "[U]se the ALARA program as a mechanism to facilitate interaction between RP and operations personnel"

   The applicant would assure interaction between RP and operations personnel because both operations and radiation safety management will be participating on the SRC and making joint recommendations for improvements. Recommendations would be tracked to completion using the facility's Corrective Action Program.

6. "[R]egularly review and revise, when appropriate, the ALARA program goals and objectives and to incorporate, when appropriate, new approaches, technologies, operating procedures or changes that could reduce potential radiation exposures at a reasonable cost"

   As discussed in Section 4.2.1 of the EREF SAR, the SRC would review the effectiveness of the ALARA program and determine if exposures, releases and contamination levels are in accordance with the ALARA concept. The SRC would also periodically review the goals and objectives of the ALARA program. These goals and objectives would be revised to incorporate, as appropriate, new technologies or approaches and operating procedures or changes that could cost-effectively reduce potential radiation exposures.

Based on the analysis as summarized above, the staff finds that the commitments in the EREF SAR (AES, 2009a) satisfactorily address the application acceptance criteria in Section 4.4.2.3 of NUREG-1520 (NRC, 2002).

4.3.3 Organization and Personnel Qualifications

The staff reviewed the applicant's organization and personnel qualifications against the acceptance criteria in NUREG-1520, Section 4.4.3.3 (NRC, 2002). The following sections identify each acceptance criteria from NUREG-1520 and discuss the staff's analysis as to whether the information provided by the applicant in the SAR (AES, 2009a) meets the criteria.

1. "[A]ppoint suitably trained RP personnel and identify their authority and responsibilities"

   The applicant described the qualifications, authority, and responsibilities of RP personnel in Sections 2.2.1, 4.1.1, and 4.3 of the EREF SAR (AES, 2009a). The Radiation Protection/Chemistry Manager would be responsible for establishing and implementing the RP program, which includes training personnel in use of equipment, control of radiation exposure of personnel, continuous determination and evaluation of the radiological status of the facility, and conducting the radiological environmental monitoring program. The Radiation Protection/Chemistry Manager would have direct access to the Plant Manager regarding all matters involving RP, would be skilled in the interpretation of RP data and regulations, would be familiar with the operation of the facility and RP concerns of the site, and would be a resource in radiation safety management decisions.

   The Radiation Protection/Chemistry Manager would have, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and 4 years of responsible nuclear experience associated with the implementation of an RP program. At least one member of the RP staff would have a minimum of 2 years experience at a facility that processes uranium, including uranium in soluble form. The RP staff would be trained and qualified consistent with guidance provided in ANSI/ANS Standard 3.1 "Selection, Qualification, and Training of Personnel for Nuclear Power Plants," (ANSI/ANS, 1993).

2. "[E]stablish clear organizational relationships among the individual positions responsible for the radiation protection program and other line managers"

   The organizational relationships between the various managers are described in Section 2.2.1 of the SAR (AES, 2009a). Of these relationships, the Radiation Protection/Chemistry Manager reports up through the ESH&L Manager to the Plant Manager. The operations line managers organizational relationships also lead to the Plant Manager through an unrelated managerial path.

3. "[A]ppoint a suitably trained RP program director (typically referred to as the radiation safety officer) who has direct access to the facility manager, who is skilled in the interpretation of data and regulations pertinent to radiation protection, who is familiar with the operation of the facility and RP concerns of the site, who is used as a resource in radiation safety management decisions, and who will be responsible for establishing and implementing the RP program"

   In Section 4.3 of the EREF SAR (AES, 2009a), the applicant designated the Radiation Protection/Chemistry Manager as the individual who would have responsibility for establishing and implementing the RP Program. This individual would be skilled in the interpretation of RP data and regulations, familiar with the operation of the facility and relevant RP concerns, and be a resource for radiation safety management decisions.

4. "[A]ssign responsibility to the RP program staff for implementation of the RP program functions"

   The applicant would assign responsibility for implementing the RP program functions to the RP staff as stated in Section 4.3 of the EREF SAR (AES, 2009a).

5. "[D]escribe the minimum training requirements and qualifications for the RP staff"

   As described in Sections 2.2.4, 4.1.2 and 4.3 of the EREF SAR (AES, 2009a), the Radiation Protection/Chemistry Manager's qualifications would minimally include a bachelor's degree and at least four years of nuclear experience. At least one member of the staff would have at least two years of experience in a uranium processing facility. Other members of the staff would be trained and qualified consistent with the guidance provided in ANSI standard 3.1, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants," (ANSI 1993).

Based on the analysis as summarized above, the staff finds that the information provided in the EREF SAR (AES,2009a) satisfactorily address the application acceptance criteria in Section 4.4.3.3 of NUREG-1520 (NRC, 2002).

4.3.4 Written Procedures

The staff reviewed the applicant's written procedure commitments against the acceptance criteria in NUREG-1520, Section 4.4.4.3 (NRC, 2002). The following sections identify each acceptance criteria from NUREG-1520 and discuss the staff's analysis as to whether the information provided by the applicant in the SAR (AES, 2009a) meets the criteria.

1. "[P]repare written, approved RP procedures to carry out activities related to the RP program"

   The applicant has committed to using written procedures to conduct operations involving licensed materials in Section 4.4 of the EREF SAR (AES, 2009a). RP procedures would be prepared, reviewed, and approved to carry out activities related to the RP program. They would also be used to ensure RP activities are conducted in a safe, effective, and consistent manner. The RP procedures are reviewed and revised, as necessary, to incorporate any facility or operational changes to the facility's Integrated Safety Analysis (ISA).

2. "[S]pecify how the RP procedures will be prepared, authorized, approved, and distributed"

   As discussed in Section 4.3 of the EREF SAR (AES, 2009a), the applicant committed that RP procedures would be prepared by qualified personnel, reviewed by members of the facility staff (i.e., personnel with enrichment plant operating experience and other staff members, as appropriate), and approved by the Radiation Protection/Chemistry Manager (or designee) and the Plant Manager (or designee).

3. "[S]pecify written, approved RWPs for activities involving licensed material that are not covered by written RP procedures"

The applicant discussed RWPs in Section 4.4.1 of the EREF SAR (AES, 2009a). The applicant commits to perform all work in Restricted Areas in accordance with a RWP. The applicant would also issue RWPs for activities involving licensed materials not covered by operating procedures, where radioactivity levels are likely to exceed airborne radioactivity limits, or whenever deemed as necessary by the Radiation Protection/Chemistry Manager. RWPs would provide a description of the work or authorized activities; summary results of dose rate, contamination, and airborne radioactivity surveys; and precautions, which may include personnel protective equipment, stay-times or dose limits, recordkeeping requirements, and coverage by an RP technician. RWPs would require approval by the Radiation Protection/Chemistry Manager or qualified designee, and would have a predetermined period of validity with a specified expiration or termination time.

Based on the analysis as summarized above, the staff finds that the commitments in the EREF SAR (AES, 2009a) satisfactorily address the application acceptance criteria in Section 4.4.4.3 of NUREG-1520 (NRC, 2002).

4.3.5 Training

The staff reviewed the applicant's training commitments against the acceptance criteria in NUREG-1520, Section 4.4.5.3 (NRC, 2002). The following sections identify each acceptance criteria from NUREG-1520 and discuss the staff's analysis as to whether the information provided by the applicant in the SAR (AES, 2009a) meets the criteria.

1. "[D]esign and implement an employee RP training program that complies with the requirements of 10 CFR Parts 19 and 20"

   The applicant has incorporated the provisions of 10 CFR 19.12 into the RP training program, as discussed in Section 4.5, of the EREF SAR (AES, 2009a). The requirements in 10 CFR 19.12 address required health physics information the applicant must make available to workers likely to receive exposures greater than 1 milliSievert (mSv) (100 mrem) per year. The applicant's RP training program would ensure that workers likely to receive such exposures are:

   - Kept informed of the storage, transfer, or use of radioactive material
   - Instructed in the health protection problems associated with exposure to radiation and radioactive material, in precautions or procedures to minimize exposure, and in the purposes and functions of protective devices employed
   - Required to observe, to the extent within the worker's control, the applicable provisions of the NRC regulations and licenses for the protection of personnel from exposure to radiation and radioactive material
   - Instructed of their responsibility to report promptly to the facility management, any condition which may cause a violation of NRC regulations and licenses or unnecessary exposure to radiation and radioactive material
   - Instructed in the appropriate response to warnings made in the event of any unusual occurrence or malfunction that may involve exposure to radiation and radioactive material
   - Advised of the various notifications and reports to individuals that a worker may request in accordance with 10 CFR 19.13

2. "[P]rovide training, to all personnel and visitors entering restricted areas, that is commensurate with the health risk to which they may be exposed, or provide trained escorts"

   The applicant has committed in Section 4.5 of the EREF SAR (AES, 2009a) to an RP training program that will be designed and implemented to provide training to all personnel and visitors, unless provided with trained escorts, who enter Restricted Areas or Controlled Areas, and which would be commensurate with the radiological hazard to which they may be exposed. As per Section 11.3.3.1.1 of the SAR (AES, 2009a), "personnel access procedures ensure the completion of formal nuclear safety training prior to permitting unescorted access into the Controlled Access Area."

3. "[P]rovide a level of training based on the potential radiological health risks associated with that employee's work responsibilities"

   In Section 4.5 of the EREF SAR (AES, 2009a), the applicant committed to provide a level of training based on the potential radiological health risks associated with the individual's work responsibilities.

4. "[I]ncorporate, in the RP training program, the provisions of 10 CFR 19.12 and topics such as: correct handling of radioactive materials; minimization of exposures to radiation and/or radioactive materials; access and egress controls and escort procedures; radiation safety principles, policies, and procedures; monitoring for internal and external exposures; monitoring instruments; contamination control, including protective clothing and equipment; ALARA and exposure limits; radiation hazards and health risks; and, emergency response"

   As discussed in Section 11.3.3.1.1 of the EREF SAR (AES, 2009a), topics covered in the applicant's training program would include:

   - Notices, reports and instructions to workers
   - Practices designed to keep radiation exposures ALARA
   - Methods of controlling radiation exposures
   - Contamination control methods (including decontamination)
   - Use of monitoring equipment
   - Emergency procedures and actions
   - Nature and sources of radiation
   - Safe use of chemicals
   - Biological effects of radiation
   - Use of personnel monitoring devices
   - Principles of nuclear criticality safety
   - Risk to pregnant females
   - Radiation protection practices
   - Protective clothing
   - Respiratory protection
   - Personnel surveys

5. "[R]eview the RP training program at least every 3 years and to conduct refresher training at least every 3 years to address changes in policies, procedures, requirements, and the facility ISA"

   As discussed in Section 4.5.1 of the EREF SAR (AES, 2009a), individuals requiring unescorted access to a Restricted Area would receive annual retraining. Retraining for individuals would be scheduled and reported by means of a computerized tracking system.

   Contents of the formal training program would be reviewed and updated at least annually by the EHS&L Manager or the Radiation Protection/Chemistry Manager to ensure the program is current and up to date. In Sections 4.5 and 11.3.3.1.1 of the EREF SAR (AES, 2009a), the applicant more conservatively commits to evaluate the RP sections of the training program at least annually.

6. "[E]valuate the effectiveness and adequacy of the training program's curriculum and instructors"

   The applicant would evaluate the effectiveness and adequacy of the training program through initial examination, audits, and assessments of operations and maintenance personnel responsible for following the requirements of criticality safety and RP. This is discussed in Section 11.3.3.1.1 of the EREF SAR (AES, 2009a).

Based on the analysis as summarized above, the staff finds that the commitments in the EREF SAR (AES, 2009a) satisfactorily address the application acceptance criteria in Section 4.4.5.3 of NUREG-1520 (NRC, 2002).

4.3.6 Ventilation and Respiratory Protection Programs

The staff reviewed the applicant's ventilation and respiratory protection program commitments against the acceptance criteria in NUREG-1520, Section 4.4.6.3 (NRC, 2002). The following sections identify each acceptance criteria from NUREG-1520 and discuss the staff's analysis as to whether the information provided by the applicant in the SAR (AES, 2009a) meets the criteria.

1. "[I]nstall appropriately sized ventilation and containment systems in areas of the facility identified in the ISA Summary as having potential airborne concentrations of radionuclides that could exceed the occupational, derived air concentration values specified in 10 CFR Part 20, Appendix B, during normal operations"

   The applicant discussed ventilation commitments in Section 4.6.1 of the EREF SAR (AES, 2009a). The confinement of uranium and the attenuation of its associated radiation are considered a design requirement for the facility. The internal radiation exposure of workers would be controlled primarily by the containment of uranium hexafluoride (UF6) within process equipment. The entire UF6 enrichment process, except for liquid sampling, would be operated under a partial vacuum so that leaks are into the system and not into work areas. This indicates that the entire process will be fully contained such that little airborne radioactivity would be present.

The applicant stated that the design of the ventilation would be consistent with the guidance contained in the documents ANSI N510-1989, "Testing of Nuclear Air Cleaning Systems" (ANSI, 1989) and "DOE Nuclear Air Cleaning Handbook" (DOE, 2003). Ventilation systems serving potentially contaminated areas include design features that provide for confinement of radiological contamination. These systems exhaust 100% of the air handled to the environment through exhaust vents after being filtered to remove radioactive particulates.

2. "[D]escribe management measures, including preventive and corrective maintenance and performance testing, to ensure that the ventilation and containment systems designated as IROFS operate when required, and are within their design specifications"

   The applicant's ISA Summary (AES, 2009b) Tables 3.8-1 and 3.8-2 list all items relied on for safety (IROFS) identified by the facility. Staff observed no IROFS identified as resulting from potential radiological exposure consequences. This is not unexpected as the licensed material is natural and low enriched uranium compounds and, typically, the consequences resulting from the toxicity of reaction products outweigh the radiological exposure consequences for a release of materials. As such, this specific criterion is not applicable to this evaluation. Discussion of chemical and criticality IROFS and their management measures is presented in Chapters 3, 5, and 6 and in Appendices A, B, and G of this SER.

3. "[D]escribe the design criteria for the ventilation and containment systems, including minimum flow velocity at openings in these systems, maximum differential pressure across filters and types of filters to be used"

   The applicant addressed the design of the facility ventilation in Sections 4.6, 4.6.1, and 4.8 of the EREF SAR (AES, 2009a). The applicant committed for ventilation design to be consistent with the "DOE Nuclear Air Cleaning Handbook" (DOE, 2003) and ANSI 510-1989, "Testing of Nuclear Air Cleaning Systems" (ANSI, 1989b).

   All effluents released from potentially contaminated areas would be filtered to remove radioactive particulates before it is released. The ventilation systems would be designed to maintain the potentially contaminated areas at a slightly negative pressure relative to the uncontaminated areas. This ensures that the airflow direction is from areas of little or no contamination to areas of higher contamination. The systems would operate slightly below atmospheric pressure to remove potentially hazardous vapors and particulates from confined areas of the plant. The systems would contain high efficiency particulate air (HEPA) and carbon adsorption filters to remove radioactive materials from the gas stream prior to release from the plant. Effluent air streams would be continuously monitored for alpha activity and hydrogen fluoride (HF) concentrations and automatically alarm if concentrations above the set point are detected. Differential pressure across high efficiency particulate air filters in potentially contaminated exhaust systems would be monitored monthly or automatically. Automatic monitors would have alarm features. Filters would be replaced when differential pressure exceeds the manufacturers' ratings or if the filters fail to function properly.

   Gloveboxes would be designed to maintain a negative differential pressure of about 0.623 millibar (mbar) (0.25 inch [in] water [H2O]). This differential pressure would be maintained anytime that the glovebox is in use. If the differential pressure is lost, use of the glovebox would be suspended until the required differential pressure is restored.

Air flow rates at exhausted enclosures and close-capture points, when in use, would be adequate to preclude escape of airborne uranium and minimize the potential for intake by workers. Air flow rates would be checked monthly when in use and after modification of any hood, exhausted enclosure, close-capture point equipment or ventilation system serving these barriers.

4. "[D]escribe the frequency and types of tests to measure ventilation and containment systems performance, the acceptance criteria, and actions to be taken when the acceptance criteria are not satisfied"

   As discussed in Section 4.6.1 of the EREF SAR (AES, 2009a), filter inspection, testing, maintenance and change out criteria would be specified in written procedures approved by the Operations Manager, or a designated alternate. Change out frequency would be based on considerations of filter loading, operating experience, differential pressure data and any UF6 releases indicated by HF alarms.

   Gloveboxes would be designed to maintain a negative differential pressure of about 0.623 mbar (0.25 in H2O). This differential pressure would be maintained anytime that the glovebox is in use. If the differential pressure is lost, use of the glovebox would be suspended until the required differential pressure is restored.

5. "[E]stablish a respiratory protection program that meets the requirements of 10 CFR Part 20, Subpart H"

   Section 4.6.2 of the EREF SAR (AES, 2009a) discusses the respiratory protection program. In that section, the applicant stated that, if the decision is made to permit the use of respiratory protection equipment to limit the intake of radioactive material, only National Institute of Occupational Safety and Health (NIOSH) certified equipment would be used. The respiratory protection program would meet the requirements of 10 CFR 20, Subpart H, "Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas."

   In Section 4.6 of the SAR (AES, 2009a), the applicant states the design of the respiratory protection program would be consistent with RG 8.15, "Acceptable Programs for Respiratory Protection," (NRC 1999), and ANSI Z88.2-1992, "Practices for Respiratory Protection," (ANSI, 1992).

6. "[P]repare written procedures for the selection, fitting, issuance, maintenance, maintenance, testing, training of personnel, monitoring, and recordkeeping for individual respiratory protection equipment, and for specifying when such equipment is to be used"

   In Section 4.6.2 of the EREF SAR (AES, 2009a), the applicant committed to develop written procedures for the respiratory protection program to address the following subjects:

   - Monitoring, including air sampling and bioassays;
   - Supervision and training of respirator users;
   - Fit testing;
   - Respirator selection;
   - Breathing air quality;
   - Inventory and control;

- Storage, issuance, maintenance, repair, testing, and quality assurance of respiratory protection equipment;
- Record keeping; and,
- Limitations on periods of respirator use and relief from respirator use.

7. "[R]evise the written procedures for use of individual respiratory protection equipment as applicable, when processing, facility, or equipment changes are made"

   Section 4.6.2 of the EREF SAR (AES, 2009a) states that respiratory protection procedures would be revised as necessary whenever changes are made to the facility, processing, or equipment.

8. "[M]aintain records of the respiratory protection program, including training for respirator use, and maintenance"

   In Section 4.6.2 of the EREF SAR (AES, 2009a), the applicant stated that records for the respiratory protection program (including training for respirator use and maintenance) would be maintained in accordance with the facility records management program as described in Section 11.7 of the EREF SAR (AES, 2009a). Section 11.7 discusses control of records and maintenance of the master file. It states that records related to health and safety would be maintained in accordance with the requirements of Title 10, Code of Federal Regulations.

Based on the analysis as summarized above, the staff finds that the commitments in the EREF SAR (AES, 2009a) satisfactorily address the application acceptance criteria in Section 4.4.6.3 of NUREG-1520 (NRC, 2002).

4.3.7 Radiation Survey and Monitoring Programs

The staff reviewed the applicant's radiation survey and monitoring program commitments against the acceptance criteria in NUREG-1520, Section 4.4.7.3 (NRC, 2002). The following sections identify each acceptance criteria from NUREG-1520 and discuss the staff's analysis as to whether the information provided by the applicant in the SAR (AES, 2009a) meets the criteria.

1. "[H]ave radiation surveys and monitoring programs consistent with the requirements of 10 CFR Part 20, Subpart F"

   The applicant committed to written procedures that will assure compliance with 10 CFR Part 20, Subpart F, in Section 4.7 of the EREF SAR (AES, 2009a). 10 CFR Part 20, Subpart F, has multiple requirements which are satisfied by the applicant's commitments meeting the other acceptance criteria addressed in this section.

2. "[P]repare written procedures for the radiation survey and monitoring program that include an outline of the program objectives, sampling procedures, data analysis methods, types of equipment and instrumentation to be used, frequency of measurements, recordkeeping and reporting requirements, and actions to be taken when measurements exceed 10 CFR Part 20 occupational dose limits or administrative levels established by the applicant"

In Section 4.7 of the EREF SAR (AES, 2009a), the applicant committed that the radiation survey and monitoring programs would be consistent with multiple guidance documents including the ones below:

- RG 8.2, "Guide for Administrative Practice in Radiation Monitoring" (NRC, 1973a)
- RG 8.4, "Direct-Reading and Indirect-Reading Pocket Dosimeters" (NRC, 1973b)
- RG 8.7, "Instructions for Recording and Reporting Occupational Radiation Exposure Data," Rev. 2 (NRC, 2005)
- RG 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program" (NRC, 1993e)
- RG 8.24, "Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication," (NRC, 1979)
- RG 8.25, "Air Sampling in the Workplace" (NRC, 1992a)
- RG 8.34, "Monitoring Criteria and Methods To Calculate Occupational Radiation Doses" (NRC, 1992b)
- NUREG-1400, "Air Sampling in the Workplace" (NRC, 1993f)
- ANSI/HPS N13.1-1999, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities," (ANSI/HPS, 1999)
- ANSI N323-1978, "Radiation Protection Instrumentation Test and Calibration" (ANSI, 1978)
- ANSI/HPS N13.11-2001, "Dosimetry-Personnel Dosimetry Performance-Criteria for Testing" (ANSI/HPS, 2001)
- ANSI/HPS N13.22-1995, "Bioassay Program for Uranium" (ANSI/HPS, 1995)
- ANSI/HPS N13.27-1981, "Performance Requirements for Pocket-Sized Alarm Dosimeters and Alarm Ratemeters," (ANSI/HPS, 1981)
- ANSI/HPS N13.30-1996, "Performance Criteria for Radiobioassay," (ANSI,1996)
- ANSI N13.6-1966 (R1989), "Practice for Occupational Radiation Exposure Records Systems" (ANSI, 1989)

The applicant stated that the procedures that would be developed to implement the programs would include an outline of the program objectives, sampling procedures, data analysis methods, types of equipment and instrumentation to be used, frequency of measurements, recordkeeping and reporting requirements, and actions to be taken when measurements exceed the occupational dose limits in 10 CFR Part 20 or the administrative levels established by the applicant.

3. "[D]esign and implement a personnel monitoring program for external occupational radiation exposures that outlines methods or procedures to
    a. Identify the criteria for worker participation in the program
    b. Identify the types of radiation to the monitored
    c. Specify how exposures will be measured, assessed and recorded
    d. Identify the type and sensitivity of personal dosimeters to be used, when they will be used, and how the collected data will be processed and evaluated
    e. Identify the facility's administrative exposure levels or action levels at which actions are taken to investigate the cause of exposures exceeding these levels"

In Section 4.7.5 of the EREF SAR (AES, 2009a), the applicant committed to having all personnel whose duties require them to enter Restricted Areas wear individual external dosimetry devices, e.g., passive dosimeters such as thermoluminescent dosimeters (TLDs) that are sensitive to beta, gamma and neutron radiation. External dosimetry devices would be evaluated at least quarterly to ascertain external exposures. The applicant proposes an administrative limit of 1 rem/y in Table 4.1-1 of the EREF SAR (AES, 2009a). If 25 percent of the annual administrative limit (i.e., 2.5 mSv or 0.250 rem) is exceeded in any quarter, then an investigation would be performed and documented to determine what types of activities may have contributed to the worker's external exposure. If the administrative limit is exceeded, the Radiation Protection/Chemistry Manager would be informed. The Radiation Protection/Chemistry Manager would be responsible for determining the need for and recommending investigations or corrective actions to the responsible manager(s).

4. "[D]esign and implement a personnel monitoring program, for internal occupational radiation exposures, based on the requirements of 10 CFR 20.1201, 20.1204, and 20.1502(b), that outlines methods or procedures to:

    a. Identify the criteria for worker participation in the program
    b. Identify the type of sampling to be used, the frequency of collection and measurement, and the minimum detection levels
    c. Specify how worker intakes will be measured, assessed, and recorded
    d. Specify how the data will be processed, evaluated, and interpreted
    e. Identify the facility's administrative exposure levels or the levels at which actions are taken to investigate the causes of exposures exceeding these levels"

    The applicant made commitments in Section 4.7.6 of the EREF SAR (AES, 2009a) that internal exposures for all personnel wearing external dosimetry devices would be evaluated via direct bioassay (e.g., in vivo body counting), indirect bioassay (e.g., urinalysis), or an equivalent technique. For soluble (Class D) uranium, regulations in 10 CFR 20.1201(e) require worker intake be no more than 10 milligrams (mg) of soluble uranium in a week. This limit is to protect workers from the toxic chemical effects of uranium. The facility annual administrative limit for the TEDE would be 1.0 rem. Internal doses would be evaluated at least annually. If the facility annual administrative limit is exceeded as determined from bioassay results, then an investigation would be performed and documented to determine what types of activities may have contributed to the worker's internal exposure.

5. "[C]omply with the requirements of 10 CFR 20.1202 for summation of external and internal occupational radiation exposures through the use of procedures such as those outlined in RGs 8.7 or 8.34"

    In Section 4.7.7 of the EREF SAR (AES, 2009a), the applicant stated that the internal and external exposure values would be summed in accordance with 10 CFR 20.1202. Procedures for the evaluation and summation of doses would be based on the guidance contained in RGs 8.7 and 8.34.

6. "[D]esign and implement an air sampling program in areas of the facility identified as potential airborne radioactivity areas, to conduct air surveys, and to calibrate and maintain the airborne sampling equipment in accordance with the manufacturer's recommendations"

    Section 4.8.8.2 of the EREF SAR (AES, 2009a) addresses the facility air monitoring and sampling program. In this section, the applicant states that active on-line monitors for airborne alpha emitters that would be used to measure representative airborne concentrations of radionuclides that may be due to facility operation. On-line monitoring for gross alpha activity would be performed assuming all the alpha activity is due to uranium. When airborne activity data would be used for dose calculations, the assumption would be that all the activity is due to uranium-234, class D material. The lower limit of detection would be either 0.02 mg of uranium in the total sample or a concentration of 3.7 nanoBecquerel per milliliter (nBq/ml) (1E-13 microCurie per milliliter [µCi/ml]) gross alpha. An action level would be established at 1 mg of total uranium likely to be inhaled by a worker in seven days.

In addition, permanent monitors would be operated in the restricted areas to collect continuous samples. The filters in these monitors would be changed and analyzed weekly and following any indication of release that might lead to airborne concentrations of uranium likely to exceed (1) 10 percent of the values listed in 10 CFR 20.1003 or (2) the total uranium action level of one milligram of total uranium inhaled in one week. The filters would also be changed each shift following changes in process equipment or process control or following detection of any event (e.g., leakage, spillage or blockage of process equipment) that would likely exceed (1) 10 percent of the values listed in 10 CFR 20.1003, or (2) the total uranium action level of one milligram inhaled by a worker in one week. The representativeness of workstation air samplers would be checked annually and when significant process or equipment changes have been made.

In Section 4.7 of the EREF SAR (AES, 2009a), the applicant stated that calibration will be performed in accordance with written established procedures and documented prior to the initial use of each airflow measurement instrument (used to measure flow rates for air or effluent sampling) and each radioactivity measurement instrument. Periodic operability checks would be performed in accordance with written established procedures. Calibrations would be performed and documented on each airflow measurement and radioactivity measurement instrument at least annually (or according to manufacturers' recommendations, whichever is more frequent) or after failing an operability check, or after modifications or repairs to the instrument that could affect its proper response, or when it is believed that the instrument has been damaged.

7. "[I]mplement additional procedures, as may be required by 10 CFR Part 20 and the ISA Summary, to control the concentration of airborne radioactive material (e.g., control of access, limitation of exposure times to licensed materials, and use of respiratory protection equipment)"

   The applicant addressed practices contained in this criteria citation in Section 4.6.2 of the EREF SAR (AES, 2009a). The applicant commits that when it is not possible to control the concentrations of radioactive material in the air to values below those that define an airborne radioactivity area, other means would be implemented to maintain the total effective dose equivalent ALARA. In these cases, the ALARA goal would be met by an increase in monitoring and the limitation of intakes by one or more of the following means: control of access; limitation of exposure times; use of respiratory protection equipment; or other controls, as available and appropriate. Respiratory protection procedures are discussed in Section 4.3.6(6) of this SER.

8. "[C]onduct a contamination survey program in areas of the facility identified in the ISA Summary most likely to be radiologically contaminated (the program must include the types and frequencies of surveys for various areas of the facility and the action levels and actions to be taken when contamination levels are exceeded)"

   Section 4.8.5.1 of the EREF SAR (AES, 2009a) states that contamination survey monitoring would be performed for all UF6 process areas and areas in which uranic materials are handled or stored. Surveys would include routine checks of non-UF6 process areas, including areas normally not contaminated. Monitoring would include direct radiation and removable contamination measurements. Survey procedures would be based on the potential for contamination of an area and operational experience. The Restricted Areas would be surveyed at least weekly. The lunch room and change rooms would be surveyed at least daily.

   If surface contamination levels exceed the following action levels, clean-up of the contamination would be initiated within 24 hours of the completion of the analysis:

   - Removable contamination: 83.3 Becquerel per 100 square centimeters (Bq/100 cm^2) (5000 disintegrations per minute per 100 square centimeters [dpm/100 cm^2]) alpha or beta/gamma
   - Fixed contamination: 4.2 kBq/100 cm^2 (250,000 dpm/100 cm^2) alpha or beta/gamma.

   In addition, in Section 4.7 of the EREF SER (AES, 2009a), the applicant committed to developing procedures consistent with guidance found in RG 8.24, "Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication," (NRC, 1979). This guidance establishes frequencies for contamination surveys and administrative limits for different areas of the facility.

9. "[I]mplement the facility's corrective action program when the results of personnel monitoring or contamination surveys exceed the applicant's administrative personnel contamination levels"

   The applicant stated in Sections 4.1 and 4.7 of the EREF SAR (AES, 2009a) that the facility corrective action process would be implemented if: (1) personnel dose monitoring results or personnel contamination levels exceed the administrative personnel limits; or if an incident results in airborne occupational exposures exceeding the administrative limits, or (2) the dose limits in 10 CFR 20, Appendix B, or 10 CFR 70.61 are exceeded. In the event the occupational dose limits given in 10 CFR 20, Subpart C, would be exceeded, notification of the NRC would be in accordance with the requirements of 10 CFR 20, Subpart M, "Reports."

10. "[I]mplement the facility's corrective action program when any incident results in airborne occupational exposures to radiation exceeding the facility's administrative limits, or the dose limits in 10 CFR Part 20, Appendix B, or 10 CFR 70.61"

    See the discussion under number 9 above.

11. "[U]se equipment and instrumentation with sufficient sensitivity for the type or types of radiation being measured and to calibrate and maintain equipment and instrumentation in accordance with manufacturers' recommendations"

    Section 4.7 of the EREF SAR (AES, 2009a) states that portal monitors, hand and foot monitors, and friskers would have the required sensitivity to detect alpha contamination on personnel to ensure that radioactive materials do not spread to the areas outside the Restricted Areas. Instruments would be calibrated with sources that are within ±5 percent of the reference value and are traceable to the National Institute of Standards and Technology (NIST) or equivalent. The background and efficiency of laboratory counting instruments, when used for radiation protection purposes, would be determined daily. This determination would be less frequent only if necessary due to long counting intervals.

    Discussion of air sampling and monitoring equipment and calibration/maintenance is provided in Section 4.3.7(6) of this SER.

12. "[E]stablish policies to ensure equipment and materials removed from restricted areas to unrestricted areas are not contaminated above the specified release levels in NRC Branch

    Technical Position, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material," April 1993"

    The applicant stated in Section 4.7.8 of the EREF SAR (AES, 2009a) that monitor stations will be established at the entry and exit points for Restricted Areas. Monitors would be provided to detect radioactive contamination on personnel and their personal items, including hard hats. All personnel would be required to monitor themselves, any hand-carried personal items, and hard hats prior to exiting a Restricted Area. In Section 4.7.2 of the EREF SAR (AES, 2009a), the applicant states that action levels for skin and personal clothing contamination at the point of egress from Restricted Areas and any additional designated areas within the Restricted Area (e.g., a Contaminated Area which is provided with a step-off pad and contamination monitor) would not exceed 2.5 Bq/100 cm^2 (150 dpm/100 cm^2) alpha or beta/gamma contamination (corrected for background). Clothing contaminated above egress limits shall not be released unless it can be laundered to within these limits. If skin or other parts of the body are contaminated above egress limits, reasonable steps that exclude abrasion or other damage will be undertaken to effect decontamination.

    The applicant further stated in Section 4.10 of the EREF SAR (AES, 2009a) that the facility will follow NRC Branch Technical Position (BTP): "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material" (NRC, 1993a). This guide would apply to the abandonment or release for unrestricted use, of surfaces, premises and equipment.

13. "[L]eak-test all sealed sources in accordance with the following NRC Branch Technical Positions: (1) "License Condition for Leak-Testing Sealed Byproduct Material Sources," April 1993, (2) "License Condition for Leak-Testing Sealed Plutonium Sources," April 1993, (3) "License Condition for Plutonium Alpha Sources," April 1993, (4) "License Condition for Leak-Testing Sealed Source Which Contains Alpha and/or Beta-Gamma Emitters," April 1993, and (5) "License Condition for Leak-Testing Sealed Uranium Sources," April 1993"

Section 4.11.1 of the EREF SAR (AES, 2009a) states that leak-testing of sources would be performed in accordance with the following NRC BTPs:

- License Condition for Leak-Testing Sealed Byproduct Material Sources (NRC,1993b)
- License Condition for Leak-Testing Sealed Source Which Contains Alpha and/or Beta-Gamma Emitters (NRC,1993c)
- License Condition for Leak-Testing Sealed Uranium Sources (NRC, 1993d).

14. "[E]stablish and implement an access control program that ensures that (a) signs, labels, and other access controls are properly posted and operative, (b) restricted areas are established to prevent the spread of contamination and are identified with appropriate signs, and (c) step-off pads, change facilities, protective clothing facilities, and personnel-monitoring instruments are provided in sufficient quantities and locations"

    The applicant stated in Section 4.7.2 of the EREF SAR (AES, 2009a) that the facility would establish and implement an access control program that ensures that (a) signs, labels, and other access controls are properly posted and operative, (b) restricted areas are established to prevent the spread of contamination and are identified with appropriate signs, and (c) step-off pads, change facilities, protective clothing facilities, and personnel monitoring instruments are provided in sufficient quantities and locations.

    Because there will be no High Radiation Areas in the facility, there would be no areas where access is physically prevented due to radiation level. Access control would be by administrative methods. Access to certain areas may be physically prevented for security reasons. Personnel who have not been trained in radiation protection procedures would not be allowed access to a Restricted Area without escort by other trained personnel.

    Access to and egress from a Restricted Area would be through one of the monitor stations at the particular Restricted Area boundary. Access to and egress from each Radiation Area, Contaminated Area or Airborne Radioactivity Area within a Restricted Area may also be individually controlled. A contamination monitor (e.g., frisker, hand and foot monitor or portal monitor), step-off pad, and container for any discarded protective clothing may be provided at the egress point from certain of these areas to prevent the spread of contamination.

15. "[H]ave a radiation reporting program consistent with the requirements of 10 CFR Parts 19 and 20"

    The applicant states in Section 4.7 of the EREF SAR (AES, 2009a) that the written procedures for the radiation survey and monitoring programs would assure compliance with the requirements of 10 CFR 20, Subparts L, "Records," and M, "Reports." Procedures would also be consistent with guidance in RG 8.7, "Instructions for Recording and Reporting Occupational Radiation Exposure Data," Rev. 2 (NRC, 2005). The applicant further elaborated in Section 4.11.2 (AES, 2009a) that the facility would meet the regulations for the additional program commitments applicable to records and reports: 10 CFR 20 Subpart L, "Records;" Subpart M, "Reports;" Section 70.61, "Performance requirements," and Section 70.74, "Additional Reporting Requirements." The facility would maintain complete records of the Radiation Protection Program for at least the life of the facility.

By procedure, the facility would report to the NRC, within the time specified in 10 CFR 20.2202 and 10 CFR 70.74, any event that results in an occupational exposure to radiation exceeding the dose limits in 10 CFR Part 20. The facility would prepare and submit to the NRC an annual report of the results of individual monitoring, as required by 10 CFR 20.2206(b).

Section 11.7 of the EREF SAR (AES, 2009a) discusses record management for the facility. This section states that records related to health and safety would be maintained in accordance with the requirements of Title 10, Code of Federal Regulations.

The applicant stated in Section 2.3.7 of the EREF SAR (AES, 2009a) that employees would have access to various resources to ensure their safety or quality concerns are addressed including NRC's requirements under 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations."

Based on the analysis as summarized above, the staff finds that the commitments in the EREF SAR (AES, 2009a) satisfactorily address the application acceptance criteria in Section 4.4.7.3 of NUREG-1520 (NRC, 2002).

4.3.8 Additional Program Requirements

The staff reviewed the applicant's additional program commitments against the acceptance criteria in NUREG-1520, Section 4.4.8.3 (NRC, 2002). The following sections identify each acceptance criteria from NUREG-1520 and discuss the staff's analysis as to whether the information provided by the applicant in the SAR (AES, 2009a) meets the criteria.

1. "[M]aintain records of the RP program (including program provisions, audits, and reviews of the program content and implementation), radiation survey results (air sampling, bioassays, external-exposure data from monitoring of individuals, internal intakes of radioactive material), and results of its corrective action program referrals, RWPs, and planned special exposures"

   The applicant stated in Section 4.11.2 of the EREF SAR (AES, 2009a) that the facility would maintain complete records of the RP Program for at least the life of the facility. This would specifically include: RP program provisions, audits, and reviews of the program content and implementation, radiation survey results (air sampling, bioassays, external-exposure data from monitoring of individuals, internal intakes of radioactive material), and results of corrective action program referrals, RWPs and planned special exposures.

2. "[E]stablish a program to report to the NRC, within the time specified in 10 CFR 20.2202 and 10 CFR 70.74, any event that results in an occupational exposure to radiation exceeding the dose limits in 10 CFR Part 20"

   As stated in Section 4.11.2 of the EREF SAR (AES, 2009a), the applicant would develop procedures such that the facility would report to the NRC, within the time specified in 10 CFR 20.2202 and 10 CFR 70.74, any event that results in an occupational exposure to radiation exceeding the dose limits in 10 CFR 20.

3. "[P]repare and submit to the NRC an annual report required by 10 CFR 20.2206(b)"

   Also in Section 4.11.2 of the EREF SAR (AES, 2009a), the applicant committed to prepare and submit to the NRC an annual report of the results of individual monitoring, as required by 10 CFR 20.2206(b).

4. "[R]efer to its 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 is required to be reported per 10 CFR 70.74, and to report to the NRC both the corrective action taken (or planned) to protect against a recurrence and the proposed schedule to achieve compliance with the applicable license condition or conditions"

   The applicant stated in Sections 4.1 and 4.11.2 of the EREF SAR (AES, 2009a) that the facility corrective action process would be implemented if: (1) personnel dose monitoring results or personnel contamination levels exceed the administrative personnel limits; or if an incident results in airborne occupational exposures exceeding the administrative limits; or (2) the dose limits in 10 CFR 20, Appendix B or 10 CFR 70.61 are exceeded.

   If the dose limits in 10 CFR 20, Appendix B, or 10 CFR 70.61 would be exceeded, or an event is required to be reported per 10 CFR 70.74, the NRC would be informed of the corrective action taken or planned to prevent recurrence and the schedule established by the facility to achieve full compliance.

Based on the analysis as summarized above, the staff finds that the commitments in the EREF SAR (AES, 2009a) satisfactorily address the application acceptance criteria in Section 4.4.8.3 of NUREG-1520 (NRC, 2002).

4.4 Evaluation Findings

The applicant has established and will maintain an acceptable RP program that includes:

1. An effective, documented program to ensure that occupational radiological exposures are ALARA;

2. An organization with adequate qualification requirements for the RP personnel;

3. Approved, written RP procedures and RWPs for RP activities;

4. RP training for all personnel who have access to restricted areas;

5. A program to control airborne concentrations of radioactive material with engineering controls and respiratory protection;

6. A radiation survey and monitoring program that includes requirements for controlling radiological contamination within the facility and monitoring of external and internal radiation exposures; and

7. Other programs to maintain records, the applicant will report to the NRC in accordance with Parts 20 and 70—and correct for upsets at the facility.

As discussed in its application and based on the staff's analysis, the applicant's RP program would meet the applicable requirements of Parts 19, 20, 30, 40, and 70.

4.5 References

(AES, 2009a) AREVA Enrichment Services LLC, "Eagle Rock Enrichment Facility Safety Analysis Report," Revision 1, 2009.

(AES, 2009b) AREVA Enrichment Services LLC, "Eagle Rock Enrichment Facility Integrated Safety Analysis Summary," Revision 1, 2009.

(ANSI, 1992) American National Standards Institute, American National Standard for Respiratory Protection, ANSI Z88.2, "Practices for Respiratory Protection," 1992.

(ANSI, 1989) American National Standards Institute, ANSI N13.6-1966 (R1989), "Practice for Occupational Radiation Exposure Records Systems," 1989.

(ANSI, 1978) American National Standards Institute, ANSI N323-1978, "American National Standard Radiation Protection Instrumentation Test and Calibration" 1978.

(ANSI/ANS, 1993) American National Standards Institute/American Nuclear Society (ANSI/ANS), ANSI/ANS 3.1, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants," 1993.

(ANSI/ASME, 1989) American National Standards Institute/American Society of Mechanical Engineers,. ANSI/ASME N510-1989, "Testing of Nuclear Air Cleaning Systems," 1989.

(ANSI/HPS, 2001) American National Standards Institute/Health Physics Society, ANSI/HPS N13.11-2001, "Dosimetry-Personnel Dosimetry Performance-Criteria for Testing," 2001.

(ANSI/HPS, 1999) American National Standards Institute/Health Physics Society, ANSI/HPS N13.1-1999, "Sampling and Monitoring Releases of Airborne Radioactive Substances From the Stacks and Ducts of Nuclear Facilities," 1999.

(ANSI/HPS, 1996) American National Standards Institute/Health Physics Society, ANSI/HPS N13.30-1996, "Performance Criteria for Radiobioassay," 1996.

(ANSI/HPS, 1995) American National Standards Institute/Health Physics Society, ANSI/HPS N13.22-1995, "Bioassay Program for Uranium," 1995.

(ANSI/HPS, 1981) American National Standards Institute/Health Physics Society, ANSI/HPS N13.27-1981, "Performance Requirements for Pocket-Sized Alarm Dosimeters and Alarm Ratemeters," 1981.

(DOE, 2003) U.S. Department of Energy, DOE-HDBK-1169-2003, "DOE Handbook Nuclear Air Cleaning Handbook," 2003.

(NRC, 2005) U.S. Nuclear Regulatory Commission, Regulatory Guide 8.7, Rev. 2, "Instructions for Recording and Reporting Occupational Radiation Exposure Data," 2005.

(NRC, 2002) U.S. Nuclear Regulatory Commission, NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," 2002.

(NRC 1999) U.S. Nuclear Regulatory Commission, Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection," 1999.

(NRC, 1992a) U.S. Nuclear Regulatory Commission, Regulatory Guide 8.25, "Air Sampling in the Workplace," 1992.

(NRC, 1992b) U.S. Nuclear Regulatory Commission, Regulatory Guide 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses," 1992.

(NRC, 1993a) U.S. Nuclear Regulatory Commission, Branch Technical Position, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material," 1993.

(NRC, 1993b) U.S. Nuclear Regulatory Commission, Branch Technical Position, "License Condition for Leak-Testing Sealed Byproduct Material Sources," 1993.

(NRC, 1993c) U.S. Nuclear Regulatory Commission, Branch Technical Position, "License Condition for Leak-Testing Sealed Source Which Contains Alpha and/or Beta-Gamma Emitters," 1993.

(NRC, 1993d) U.S. Nuclear Regulatory Commission, Branch Technical Position, "License Condition for Leak-Testing Sealed Uranium Sources," 1993.

(NRC, 1993e) U.S. Nuclear Regulatory Commission, Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program, 1993.

(NRC, 1993f) U.S. Nuclear Regulatory Commission, NUREG-1400, "Air Sampling in the Workplace," 1993.

(NRC, 1991) U.S. Nuclear Regulatory Commission, NUREG-1400, "Air Sampling in the Workplace," 1991.

(NRC, 1979) U.S. Nuclear Regulatory Commission, Regulatory Guide 8.24, "Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication," 1979.

(NRC, 1977) U.S. Nuclear Regulatory Commission, Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," 1977.

(NRC, 1973a) U.S. Nuclear Regulatory Commission, Regulatory Guide 8.2, "Guide for Administrative Practice in Radiation Monitoring," 1973.

(NRC, 1973b) U.S. Nuclear Regulatory Commission, Regulatory Guide 8.4, "Direct-Reading and Indirect-Reading Pocket Dosimeters," 1973b.

CHAPTER 5.0 NUCLEAR CRITICALITY SAFETY

The purpose of this review is to determine whether the applicant's nuclear criticality safety (NCS) program is adequate to support safe operation of the facility, as required by Title 10 of the *Code of Federal Regulations* (CFR) Part 70. The applicant's Safety Analysis Report (SAR), Integrated Safety Analysis (ISA) Summary, and responses to requests for additional information (RAI) provided the basis for this determination. The information and commitments which are significant to this determination are described in Section 5.3 of this chapter.

The NCS programmatic review determines whether: (1) the applicant provided for the appropriate management of the NCS program; (2) the applicant identified, and committed to, the responsibilities and authorities of individuals for developing and implementing the NCS program; (3) the facility management measures described in 10 CFR 70.62 have been committed to and will support implementing and maintaining the NCS program; and (4) an adequate NCS program is described, which includes identifying and committing to the NCS methods, and NCS technical practices used to ensure the safe operation of the facility, as required by 10 CFR Part 70.

5.1 Regulatory Requirements

The review of AREVA Enrichment Service's (AES's) NCS program should verify that the information the applicant provided meets the requirements of 10 CFR 70.22 and 70.65, which, respectively, specify the general and additional content of an application. In addition, the NCS review should verify compliance with the regulatory requirements in 10 CFR 70.24; 70.52; 70.61; 70.62; 70.64; 70.65; 70.72; and Appendix A to Part 70.

5.2 Regulatory Acceptance Criteria

The acceptance criteria for the U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's NCS program are outlined in NUREG-1520, Section 5.4 (NRC, 2002). This includes the commitment to use Regulatory Guide (RG) 3.71, Revision 1 (NRC, 2005), which endorses the use of the American National Standards Institute/American Nuclear Society (ANSI/ANS) Series-8 NCS standards with certain modifications. In addition, the acceptance criteria for the NRC's review of the applicant's ISA methodology and ISA Summary are outlined in NUREG-1520, Section 3.4 (NRC, 2002).

5.3 Staff Review and Analysis

5.3.1 Industry Standards

The acceptance criteria for NRC's review of industry standards are contained in Sections 5.4 and 5.4.2 of NUREG-1520 (NRC, 2002).

AES commits to RG 3.71, Revision 1 (NRC, 2005).[5] Thus, AES will follow the requirements (i.e., "shall" statements) in the ANSI/ANS Series-8 standards, as endorsed by the NRC. A commitment to these standards is an acceptable means of meeting many of the acceptance criteria in Section 5.4 of NUREG-1520.

In committing to RG 3.71, AES commits to comply with all the ANS standards as endorsed by NRC. However, some of the standards are not applicable because of the nature of facility operations. The staff determined that the following ANSI/ANS Series-8 standards do not apply to any activities that the Eagle Rock Enrichment Facility (EREF) will be conducting: ANSI/ANS-8.5-1996, -8.6-1983, -8.12-1987, -8.15-2004, -8.17-2004, and -8.21-1995. While AES has committed to these standards, this has no impact on its operations as AES would have to request a license amendment to conduct activities for which these standards apply. Although ANSI/ANS-8.10-1983, *Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement,* (ANSI/ANS, 1983), does not apply to any operations at EREF, because it is not a shielded facility, the technical information in this standard will be used to determine the consequences of a criticality accident.

5.3.2 Organization and Administration

The acceptance criteria for NRC's review of organization and administration are contained in Sections 5.4.3.2 and 2.4 of NUREG-1520 (NRC, 2002).

During all phases of facility operation (design, construction, operation, and decommissioning) EREF will comply with ANSI/ANS-8.1-1983, *Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors* (ANSI/ANS, 1998), and ANIS/ANS-8.19-2005, *Administrative Practices for Nuclear Criticality Safety* (ANSI/ANS, 2005), as they relate to organization and administration (AES, 2010b). The NCS function will be performed within the engineering organization during the design of the facility. When the facility transitions from design (including construction) to operations the NCS function will transition to an independent function within the Environmental, Health, Safety and Licensing (EHS&L) function.

The EHS&L function at the proposed EREF will be created during the transition to operations and will be independent of production responsibilities (AES, 2010b). During the operations phase, the manager of the NCS function reports to the EHS&L manager. The NCS function has the authority to shutdown potentially unsafe operations, and must approve restart of any operation it has requested to be shutdown. The minimum qualifications for the NCS manager are a bachelor's degree (or equivalent) in an engineering or scientific field and four years of nuclear experience. At least one year of direct experience with the administration of NCS analyses and evaluations is also required.

NCS engineers report to the engineering organization managers during the design phase and to the NCS Manager during the operations phase of the facility (AES, 2010b). The minimum qualifications for an NCS engineer are a bachelor's degree in an engineering or scientific field, two years experience in the implementation of a criticality safety program, and successful completion of an NCS training program. In addition to these requirements, individuals who

[5] This regulatory guide endorses, with some exceptions, specific nuclear criticality safety standards developed by the American Nuclear Society's Standards Subcommittee 8 (ANS-8), "Operations with Fissionable Materials Outside Reactors."

perform NCS analyses or evaluations for facility changes shall have at least two years experience with NCS programs and performing NCS analyses.

The staff has reviewed AES's organizational structure and finds that it is acceptable because the applicant has adequately addressed the acceptance criteria in NUREG-1520, Sections 2.4 and 5.4.3.2 as they relate to NCS. The NCS function will be independent from the production staff, NCS evaluations are performed by qualified reviewers, with independent review for quality assurance, and AES's administrative practices are consistent with the requirements in ANSI/ANS-8.19-2005.

5.3.3 Management of the NCS Program

The acceptance criteria for NRC's review of the management of the NCS program are contained in Section 5.4.3.1 of NUREG-1520 (NRC, 2002).

As described in Section 5.1.1 of the SAR (AES, 2010b), the NCS program will evaluate changes to operations, recommend changes to processes to maintain safe operations, and select appropriate items relied on for safety (IROFS) and management measures. Section 5.1.5 of the SAR (AES, 2010b) describes the responsibilities of the NCS manager for the development and implementation of the NCS program. These responsibilities include:

- Performing NCS analyses and evaluations
- Establishing limits and controls
- Assuring limits and controls are properly implemented
- Monitoring plant compliance with NCS requirements.

As described in Section 5.2.1.4 of the SAR (AES, 2010b), during both the design and operating phases, qualified NCS engineers will prepare and independently review NCS analyses. Once operating, NCS engineers are also responsible for conducting and documenting NCS evaluations and assessments. During the operating phase, the NCS Manager must approve NCS analyses and evaluations (AES, 2010b).

At least one NCS engineer will be available when the facility is operating to respond to any routine request or emergency condition. As stated in Section 4.2.2.1 of the Emergency Plan (AES, 2010b), in the event of a fire, at least one responder will be assigned as the criticality safety officer who is responsible for ensuring that NCS is not compromised during firefighting activities.

As described in Section 5.1.1 of the SAR, the NCS program will be updated to reflect changes in the ISA or NCS methodologies (AES, 2010b). NCS program records, which include NCS analyses and evaluations, will be retained in accordance with the AES document control and records management procedures (See Chapter 11 of this SER) (AES, 2010b).

The staff has reviewed the applicant's management of the NCS program and finds that it is acceptable because the applicant has adequately addressed the acceptance criteria in NUREG-1520, Section 5.4.3.1.

5.3.4 NCS Management Measures

The acceptance criteria for NRC's review of NCS management measures are contained in Sections 5.4.3.3 and 5.4.3.4.7 of NUREG-1520 (NRC, 2002).

5.3.4.1 Training

The training program for personnel who handle nuclear material is based upon ANSI/ANS-8.20-1991, *Nuclear Criticality Safety Training* (ANSI/ANS, 1991). The training will include the required response to an NCS deviation and to the activation of the criticality accident alarm (AES, 2010b). Successful completion of nuclear safety training will be required to gain unescorted access into the Controlled Access Area (CAA) of the facility; retraining must be completed annually to maintain access (AES, 2010b).

The staff has reviewed the applicant's commitments regarding personnel training as they relate to NCS and finds them acceptable because the applicant has adequately addressed the acceptance criteria in NUREG-1520, Section 5.4.3.3(1).

5.3.4.2 Procedures

AES commits to ANSI/ANS-8.19-2005 (ANSI/ANS, 2005) as it relates to procedures. Personnel will only perform actions in accordance with approved written procedures. Procedures will be written such that no single, inadvertent departure from a procedure could cause a nuclear criticality accident (AES, 2010b). If a particular situation is not covered by procedure, then personnel shall report the issue and take no further action until the situation has been evaluated and recovery procedures are provided.

In addition to procedures, postings will be used to identify administrative controls in applicable work areas (AES, 2010b). These postings will be maintained current (AES, 2010b).

The staff has reviewed the applicant's commitments regarding procedures as they relate to NCS and finds them acceptable because the applicant has adequately addressed the acceptance criteria in NUREG-1520, Section 5.4.3.3(2).

5.3.4.3 Audits and Assessments

The NCS program will use operational inspections, audits, and investigations to promptly detect NCS deficiencies (AES 2010b). NCS deficiencies will be entered into the corrective action program to prevent recurrence (AES, 2010b). Records of the corrective actions will be retained (AES, 2010b).

AES commits to ANSI/ANS-8.19-2005 (ANSI/ANS, 2005) as it relates to NCS audits and assessments (AES, 2010b). Audits will be conducted to verify compliance with regulations, procedures, and the license (AES, 2010b). NCS audits will be conducted and documented quarterly such that the entire NCS program will be audited at least every two years (Section 11.5.2, AES, 2010b). Weekly NCS walkthroughs of uranium hexafluoride (UF_6) process areas will also be conducted and documented (AES, 2010b).

NCS assessments are performed under the direction of the NCS staff by personnel who are not directly involved in the function or area being assessed (AES, 2010b). Assessments are focused on the effectiveness of activities and ensuring that Quality Assurance Level 1 and 2 items are available and reliable (AES, 2010b). The Operations Department will be periodically assessed to ensure that NCS procedures are being followed and process conditions have not adversely changed (AES, 2010b). Assessments will be conducted at least semi-annually, but the exact frequency will be based on the identified NCS controls (AES, 2010b).
The staff has reviewed the applicant's commitments regarding audits and assessments as they relate to NCS and finds them acceptable because the applicant has adequately addressed the acceptance criteria in NUREG-1520, Section 5.4.3.3(3).

5.3.5 NCS Methodologies and Technical Practices

The acceptance criteria for NRC's review of NCS methodologies and technical practices are contained in Sections 5.4.3.4.1, 5.4.3.4.4, and 5.4.3.4.5 of NUREG-1520 (NRC, 2002).

AES commits to use the acceptance criteria in NUREG-1520, Section 5.4.3.4, "Methodologies and Technical Practices" (NRC, 2002), with the exception of Sections 5.4.3.4.2 (9), (13), and (15) (RAI NCS-1 and AES, 2010b), when analyzing criticality accidents. In the response to RAI NCS-1, AES stated that the acceptance criteria in Section 5.4.3.4.2(9) will not be used because there is no planned use of density as a control parameter; that the criteria in Section 5.4.3.4.2(13) will not be used because there is no planned use of concentration as a control parameter; and the criteria in Section 5.4.3.4.2(15) will not be used as there is no planned use of neutron absorption as a control parameter (AES, 2009a). In addition, the criteria in NUREG-1520, Section 5.4.3.4.1(4) does not apply to NCS analysis but does apply to the criticality accident alarm system (CAAS) (AES, 2009a). Further information about how AES will comply with these criteria is described in this section.

5.3.5.1 NCS Methodologies

<u>Double Contingency Principle</u>

AES commits to meet the double contingency principle for each process where a criticality accident is possible (AES, 2010b). AES will incorporate into process designs sufficient factors of safety to ensure that at least two unlikely, independent, and concurrent changes in process conditions are required before a criticality accident is possible (AES, 2010b). AES's commitment to NUREG-1520, Section 5.4.3.4, commits it to the policy that no single credible event or failure can result in a criticality accident (AES, 2010b).

The NCS analyses and evaluations will show how the double contingency principle is met for each process, and it will be used to determine NCS controls and items relied on for safety (IROFS) (AES, 2010b). AES commits to NUREG-1520, Section 5.4.3.4, thus double contingency will be assured by independent controls on at least two parameters or by multiple independent controls on a single parameter (AES, 2010b). The exception to the double contingency principle described in NUREG-1520, Section 5.4.3.4.4(7)(c) will not apply to EREF since AES explicitly commits to meet the double contingency principle as defined in ANSI/ANS-8.1-1998 for all processes (AES, 2010b).

NCS Determinations

AES will ensure that each process is adequately subcritical by demonstrating that the effective neutron multiplication (k_{eff}) is less than 0.95 for normal and credible abnormal conditions (AES, 2010b). k_{eff} will be calculated using the equation:

$$k_{eff} = k_c + 3\sigma,$$

where k_c is the calculated neutron multiplication factor and σ is the associated standard deviation.

NCS analyses and the safe values listed SAR Table 5.1-1 (see Table 5.3-1 of this chapter) form the nuclear criticality safety basis for the facility. NCS analyses describe the calculations performed to demonstrate that each process will be adequately subcritical under normal and credible abnormal conditions (AES, 2010b). Each NCS analysis will include a discussion of any assumptions used and will identify required limits and controls (AES, 2010b).

An NCS evaluation is prepared and approved whenever there is a change that involves or could affect uranium. The NCS evaluation determines if existing NCS analyses bound the change being evaluated or if new or revised NCS analyses are required (AES, 2010b). The basis for the evaluation is documented to allow the independent review by a second NCS engineer to confirm the analyst's conclusions (AES, 2010b). Controlled parameters and limits upon which NCS depends must be determined and identified in the NCS evaluation (AES, 2010b). NCS analyses and evaluations will be performed in accordance with ANSI/ANS-8.1-1998 and ANSI/ANS-8.19-2005 requirements (NRC, 2005). NCS analyses and evaluations will be used to determine operating limits on controlled parameters that will be as or more conservative than the safety limits derived from the NCS evaluations and analyses to ensure operations will be subcritical. NCS operating limits will be derived from safety limits and provide an additional margin of safety (AES, 2010b). This additional safety margin will account for the variability and uncertainty in processes and take into consideration changes in process parameters.

The facility will be designed and operated in accordance with SAR Table 5.1-2, which specifies safety criteria for many of the systems and components that will be used (AES, 2010b). These criteria are primarily based on the EREF safe values for 6 weight percent (wt%) uranium-235 (^{235}U) (See Table 5.3-1).

Table 5.3-1: Comparison of the EREF Safe Values for 6 wt% ^{235}U and the NRC Endorsed Single Parameter Limits and Safety Factors

Parameter	Review Comparison			EREF Safe Value for 6 wt%
	Single Parameter Limit	SRP Safety Factor	Calculated Safety Limits [4]	
Volume	30.6 L [1]	0.75	22.95 L	19.3 L
Cylinder Diameter	26.6 cm [1]	0.90	23.94 cm	22.4 cm
Slab Thickness	12.6 cm [1]	0.85	10.71 cm	10.1 cm
Uranium Mass	1.64 kg ^{235}U [1]	N/A	32.8 kg U	20.1 kg U
-no double batching		0.75	24.6 kg U	19.4 kg U
-double batching		0.45	14.76 kg U	12.2 kg U
Areal Density	0.40 g $^{235}U/cm^2$ [2]	N/A	8.0 g U/cm^2	7.9 g U/cm^2
Water Mass [3]	N/A	N/A	N/A	11.9 kg H_2O

[1] ANSI/ANS-8.1-1998, 5 wt% ^{235}U;
[2] ANSI/ANS-8.1-1998, 100 wt% ^{235}U;
[3] ANSI/ANS-8.1-1998 does not have a single parameter limit for water;
[4] Calculated safety limits assume 5 wt% ^{235}U; they are the single parameter limit multiplied by the SRP-recommended safety factor, and are expressed in the same units as EREF safe values.

The details for how the safe values for the parameters listed in SAR Table 5.1-1 were determined are described in Section 5.1.2 of the SAR (AES, 2010b). AES calculated the safe values for the parameters in the table above assuming optimally moderated UO_2F_2 solution with full water reflection (i.e., 30 cm of tight-fitting water reflection around fissile units), for both 4 and 6 wt% ^{235}U. The safe values correspond to a calculated k_{eff} of 0.95. Although AES will only enrich uranium up to 5 wt% ^{235}U, it will establish all its process safety limits, except for the contingency dump system traps and tails cylinders, assuming 6 wt% ^{235}U which provides added safety margin. Thus, an independent review of only the safe values listed for 6 wt% ^{235}U was performed by NRC staff. Except for water mass, Table 5.3-1 of this SER shows that the values calculated by AES at 6 wt% ^{235}U are conservative in comparison to well-established single parameter limits at 5 wt% ^{235}U when combined with an acceptable safety factor. The staff also determined, based on its previous experience and independent calculations for a spherical UO_2F_2-water system, that the difference between analyzing at 5 and 6 wt% ^{235}U usually results in conservatism in k_{eff} of about 3%. This is significant because it represents additional margin beyond that provided by the use of a 0.95 k_{eff} limit, and provides a high degree of assurance that processes calculated to be subcritical will be subcritical.

Only product cylinders (30B and 48Y) will use moderation as a controlled parameter (AES, 2010b). SAR Table 5.1-2 indicates that product cylinders will be limited to less than 9.3 kg of water, which represents a safety factor of about 65% when compared to the safe value EREF calculated for 5 wt% ^{235}U of 14.2 kg of water (AES, 2010b) which represents a substantial margin of safety.

The values listed in SAR Table 5.1-1 and 5.1-2 are adequately subcritical for single units (AES, 2010b). However, AES has also committed to demonstrate that each process is adequately subcritical (i.e., k_{eff} < 0.95 for normal and credible abnormal conditions) when multiple units can interact (see Section 5.3.5.2 of this SER) (AES, 2010b).

Computer Code Validation

NCS analyses are conducted using MONK8A, a well-established and well-known Monte Carlo computer code used to determine k_{eff}. Details about the code and a summary of its validation are described in SAR Section 5.2.1 (AES, 2010b). Portions of the MONK8A validation report (AES, 2008) were reviewed by NRC staff to verify that the validation was consistent with the SAR and was relevant to facility operations. AES commits to report any change in the validation report to the NRC by letter. Use of computer codes for NCS analysis other than MONK8A would require NRC approval since the SAR would need to be amended to authorize additional codes.

The MONK8A validation established a bias, bias uncertainty, and upper subcritical limit (USL) in accordance with NRC guidance (NRC, 2001). The validation does not use a positive bias, and the bias uncertainty was determined to be 0.0092. Using a 0.05 margin of subcriticality, which is widely accepted in the nuclear industry for low-enriched fuel applications, the validation report established a USL of 0.9408 for everything except the contingency dump system traps and tails cylinders (AES, 2010b). For the contingency dump system traps and tails cylinders, an additional penalty of 0.0014 was taken since the area of applicability was extrapolated to include 1.5 wt% ^{235}U (AES, 2010b). The USL for the contingency dump system traps and tails cylinders was therefore determined to be 0.9394 (AES, 2008). Despite establishing the USL as described in the MONK8A validation report, AES will use 0.95 as its subcritical limit in all its NCS analyses (Section 5.2.1.2 of (AES, 2010b)). This is equivalent to having a reduced subcritical margin of 0.0394 for the contingency dump system traps and tails cylinders and 0.0408 for all other systems. The justification for this reduced margin is discussed below.

The validation used well-known benchmark experiments, which consisted of 34 experiments with uranium enrichment between 4.46–5.64 wt% ^{235}U (Group 1), 30 experiments with uranium enriched to 9.97 wt% ^{235}U (Group 2), and 29 experiments with uranium enriched to 29.83 wt% ^{235}U (Group 3). Based solely upon enrichment, the first group of experiments would be most applicable to the proposed operations at the proposed EREF. The staff questioned whether the higher enrichment experiments should be included in the validation study, because it was not readily apparent that they were applicable to operations at the proposed EREF. The inclusion of inapplicable experiments could decrease the calculated bias or extend the validated area of applicability inappropriately. Because the applicability of these experiments was not readily apparent, the staff examined the impact of including the higher enriched experiments in the validation, by using the 0.95 subcritical limit to calculate an effective subcritical margin assuming no extrapolation penalty is taken. The goal of this was to determine whether the inclusion of these experiments would materially change the validation results. When only Group 1 is used in the validation, the subcritical margin is reduced to 0.0358. When Group 1 and 2 are used, the subcritical margin is increased to 0.0416. As stated above, when all three groups are used, the subcritical margin is 0.0408. AES has not provided justification for including experiments with uranium enrichments well outside the range required for the application; however, the inclusion or exclusion of the higher enrichment benchmarks produces only a very small difference in the USL.

A subcritical margin of 0.05 has long been accepted without extensive justification for low-enriched fuel facility processes (as discussed in (NRC, 2006). However AES is requesting a smaller margin and has included experiments well outside the credible operating parameters (AES, 2008). Performing NCS analyses at 6 wt% ^{235}U provides a significant additional safety margin (about 3% in k_{eff}) that is sufficient to compensate for the effect of including or excluding the higher enriched benchmark experiments, and is also large compared to the difference between the USL and the 0.95 subcritical limit. The subcritical limit of 0.95 is therefore justified by the presence of this additional conservatism. The staff also determined that the bias and bias uncertainty are slowly varying quantities as a function of enrichment, and therefore any reasonably anticipated change in the USL resulting from extrapolating the area of applicability (AOA) from the range of Group 1 (up to 5.64 wt% ^{235}U) to 6 wt% would be more than compensated for by this same conservative margin.

For the contingency dump system traps and tails cylinders, which are the only items analyzed with 1.5 wt% ^{235}U, significant changes in multiple operating parameters would be required before the system could approach the subcritical limit. This large difference between operating conditions and the safety limit for the contingency dump system traps and tails cylinders provides an additional safety margin that is sufficient to justify the 0.95 subcritical limit for these systems. Based on the above review, the staff concluded that there will be an adequate margin of subcriticality for safety to satisfy 10 CFR 70.61(d).

5.3.5.2 NCS Technical Practices

NCS Controls and Controlled Parameters

As described in Section 5.1.1 of the SAR (AES, 2010b), all NCS controls will be designed to prevent a criticality accident; there are no mitigative NCS controls (AES, 2010b). Where possible, AES will use passive engineered controls to ensure NCS (AES, 2010b). Through its commitment to NUREG-1520, Section 5.4.3.4.2, the applicant has committed to select controls in accordance with the following order of preference: (1) passive engineered; (2) active engineered; (3) augmented administrative; and (4) simple administrative (AES, 2010b).

AES commits to establish NCS controls to ensure that under normal and credible abnormal conditions all nuclear processes are adequately *subcritical (AES, 2010b)*. This means that NCS controls must be capable of maintaining the parameter limits established in the NCS analyses and evaluations (AES, 2010b). AES commits to NUREG-1520, Section 5.4.3.4, thus:

- NCS limits will be derived assuming credible optimal condition (i.e., most reactive conditions physically possible or limited by written commitment to the NRC) unless controls are implemented to limit a parameter to a certain range.
- Heterogeneous effects will be considered when evaluating controlled parameters.
- NCS analyses will show that each controlled parameter will be maintained during both normal and credible abnormal conditions.

The parameters which AES may control for NCS purposes are: mass, geometry/volume, enrichment, reflection, moderation, and interaction. Commitments associated with these controlled parameters are discussed in more detail below.

Mass

When mass is used as an NCS controlled parameter, analysis or sampling will be used to verify the mass of material (AES, 2010b). Records will be kept for mass transfers into and out of containers which are mass controlled (AES, 2010b). Double batching is assumed to be a credible upset condition whenever only administrative controls are used to limit the mass

(AES, 2010b). AES explicitly commits to limit the mass of uranium in any single tank in the Technical Support Building to less than or equal to 12.2 kg uranium (AES, 2010b).

Geometry/Volume

When geometry is used as a controlled parameter, subcriticality for a single unit (i.e., a component or vessel in isolation) will be demonstrated to be independent of all other parameters assuming 6 wt% ^{235}U. Geometry alone is not sufficient to ensure that a particular system is subcritical since it is generally not possible to install a single, isolated unit. Thus, AES will also consider credible interaction and reflection conditions in its NCS determinations (AES, 2010b). AES will verify all dimensions relied on for NCS prior to the beginning of operations (AES, 2010b). The safe geometry values, as demonstrated in Table 5.3-1 of this SER, provide a substantial margin of safety when geometry is used as a controlled parameter. The applicant based this conclusion on a determination that this margin is sufficient to account for credible manufacturing tolerances, deformation (e.g., from bulging and corrosion), and uncertainties in the calculational methods. AES explicitly commits to implement and maintain centrifuge, chemical trap, and product cold trap diameters and second stage UF_6 pump, uranic liquid container, vacuum cleaner, and oil container volumes to less than the safe values in Table 5.3-1 of this SER. AES also commits to demonstrate that an isolated first stage UF_6 pumps is subcritical for 6 wt% ^{235}U based only upon controlling its geometry (AES, 2010b).

Enrichment

Although AES is requesting authorization to only enrich uranium up to 5 wt% ^{235}U, it commits to conduct all its NCS analyses, except for the contingency dump system traps and tails cylinders, using 6 wt% ^{235}U (AES, 2010b). The NCS analysis for the contingency dump system traps and tails cylinders will be based upon 1.5 wt% ^{235}U (AES, 2010b). Feed and uranium byproduct cylinders are limited to less than 0.72 wt% ^{235}U, which ensures they will be subcritical under all credible conditions. In the original SAR, AES had stated that the only exception to conducting analyses at 6 wt% ^{235}U was for the contingency dump system traps (AES, 2009b). The staff observed that ISA documents indicated that portions of the primary (tails take-off) dump system and other parts of the contingency dump systems had been analyzed at 1.5 wt% ^{235}U, which was inconsistent with the commitment in the SAR (AES, 2010b). In response to the staff's RAI ISA-14 (AES, 2009a), AES revised the SAR (AES, 2010b) to include the tails cylinders in the exception but otherwise maintained its commitment to conduct all other NCS analyses using 6 wt% ^{235}U.

Reflection

AES considers full water reflection, but has determined that this is highly unlikely to occur from a source internal to the facility (AES, 2010b). Partial reflection of 2.5 cm of water is assumed where limited moderating materials may be present (AES, 2010b). This means that calculations will assume that the outside surface of a component is covered by water 2.5 cm thick. AES indicated (RAI NCS-2) (AES, 2009a) that using 2.5 cm of water is a common practice in NCS for simulating room return and human presence. Where present, concrete is modeled in the analysis since it is a better reflector than water. AES commits to NUREG-1520, Section 5.4.3.4, thus the controls to prevent potential reflectors will be identified as IROFS in the ISA Summary (AES, 2010b).

In areas where fire sprinkler systems are installed, the analyses will consider credible interstitial moderation and sheeting assuming the system activates (AES, 2010b). In addition, standpipes will be installed in areas where their failure will not result in flooding of areas containing enriched uranium greater than a critical mass (AES, 2010b).

Moderation

Moderation control will be established in accordance with ANSI/ANS-8.22-1998 (NRC, 2005). Where moderation is the only parameter used for criticality control, two independent controls will be used to verify moderator content (AES, 2010b). AES commits to NUREG-1520, Section 5.4.3.4, thus when process variables can affect moderation, the process variables will be shown in the ISA Summary to be controlled by IROFS (AES, 2010b). As stated previously, control of moderation will only be applied to product cylinders (AES, 2010b).

Optimum moderation or the worst case credible hydrogen to uranium (H/U) ratio will be used in NCS analyses when moderation is not controlled. AES demonstrated that an H/U ratio of 7 is the maximum credible moderation that can result from moisture in-leakage (AES, 2010b). This demonstration assumes the hydrogen is entirely from water molecules. Therefore, when AES uses the worst case credible moderation (or credible optimum moderation), it will select the H/U ratio between 0 and 7 that is most reactive for the system being analyzed A(ES, 2010b). Optimum moderation is the moderating condition that is most reactive without any limits on the H/U ratio.

UF_6 and vacuum pumps will use fully fluorinated (non-hydrogenous) perfluorinated polyether (PFPE) type lubricant (AES, 2010b). AES indicated that it will analyze these pumps with water, instead of PFPE, which will bound all credible abnormal moderating conditions (AES, 2010b).

Fire sprinkler systems may be installed in some areas of the facility. AES will not install sprinklers in any area where a critical mass could accumulate in an unsafe geometry (e.g., drains) or near sub-atmospheric process systems that could contain a critical mass and require moderator control (AES, 2010b).

Interaction

NCS analysis and evaluations will consider the effects of interaction (RAI NCS-10), including in-transit material, using MONK8A to ensure that $k_{eff} < 0.95$. Spacing requirements will be determined separately for each system (AES, 2010b). AES commits to NUREG-1520, Section 5.4.3.4, thus it will use either engineered controls or augmented administrative controls to maintain the physical separation between different units (AES, 2010b). In this case, an administrative control is considered to be augmented if a physical device significantly aids the operator to ensure that the required action will be performed correctly.

AES will establish controls to maintain the spacing between items in movement and other fissile material (AES, 2010b). AES commits to have only one item of each type (e.g., one pump and one trap) in movement at the same time (AES, 2010b). The only time the criticality analysis allows the spacing requirements to be relaxed is when vessels are moved in or out of fixed positions.

5.3.5.3 *Findings*

The staff has reviewed the NCS methodologies and technical practices and finds that they are acceptable because the applicant has adequately addressed the acceptance criteria in NUREG-1520, Sections 5.4.3.4.1, 5.4.3.4.2, 5.4.3.4.4, and 5.4.3.4.5. The applicant commits to the double contingency principle as required by 10 CFR 70.64(a)(9) (AES, 2010b). The staff finds that the NCS methodologies and technical practices, if applied as described in the SAR, will provide an adequate margin of subcriticality for safety as required by 10 CFR 70.61(d).

5.3.6 CAAS

5.3.6.1 *SAR Commitments*

The acceptance criteria for NRC's review are contained in Section 5.4.3.4.3 of NUREG-1520 (NRC, 2002).

AES commits to maintain documents demonstrating that its CAAS meets the requirements of 10 CFR 70.24 (AES, 2010b). The applicant will follow ANSI/ANS-8.3-1997, "Criticality Accident Alarm System," (ANSI/ANS, 1998) as modified by NRC RG 3.71, Revision 1 (NRC, 2005) (AES, 2010b). The CAAS will be uniform throughout the facility for the type of radiation detected and the alarm signals (AES, 2010b).

AES commits to have an emergency plan and will comply with ANSI/ANS-8.23-1997, "Nuclear Criticality Accident Emergency Planning and Response" [Ref. 0] (AES, 2010b). Fixed criticality accident dosimeters will be located within the buildings for accident reconstruction (AES, 2010b). AES and emergency response personnel will be provided with dosimeters prior to entering the radiological controlled areas (AES, 2010b).

The CAAS is backed up by emergency power and will be designed to remain operational during credible events and conditions (AES, 2010b). If CAAS coverage is lost, compensatory measures [such as limiting access and restricting movement of special nuclear material (SNM)] will be implemented (AES, 2010b). If coverage is lost and cannot be restored within a specified number of hours, or if an equivalent level of coverage is not provided, operations in the affected area will be shut down or quarantined as necessary (AES, 2010b). AES will develop on-site guidance on when an operation should be safely shut down if CAAS coverage cannot be restored within the specified number of hours (AES, 2010b).

In the original SAR submittal, AES only committed to provide CAAS coverage to those areas with greater than a critical mass of fissile material (AES, 2010b). However in response to the staff's concern that this was not sufficient to meet 10 CFR 70.24 (RAI NCS-5) (AES, 2009a), AES committed to provide CAAS coverage for each area where fissile material is handled, used, or stored (AES, 2010b).

5.3.6.2 *ISA Summary*

The ISA Summary provided additional information on the planned CAAS (AES, 2010b). CAAS coverage map was included which shows that each area at EREF where SNM will be handled, used, or stored is covered by at least two detectors. The coverage map is based off the planned use of gamma ray detectors with an alarm set point of 1 milligray/hour (mGy/hr). The proposed distances in areas with no shielding were compared with hand calculations described in Appendix B to ANSI/ANS-8.3-1997 (ANSI/ANS, 1997), and found to be consistent. The proposed range for CAAS detectors near autoclaves was much shorter to account for shielding.

5.3.6.3 Findings

The staff has reviewed the applicant's commitment to the CAAS requirements in 10 CFR 70.24 and finds that it is acceptable because the applicant maintains a CAAS that is capable of energizing a clearly audible alarm signal if accidental criticality occurs, and the applicant maintains emergency procedures for each area in which SNM is handled, used, or stored to ensure prompt personnel evacuation upon the sounding of the alarm. Additionally, the staff finds that the applicant has adequately addressed the acceptance criteria in NUREG-1520, Sections 5.4.3.4.3 and 3.4.3.2(4)(c).

5.3.7 Reporting Requirements

AES will establish a program to evaluate the significance of events, in terms of NCS, using qualified individuals (AES, 2010b). The criteria for reporting events to the NRC Operations Center (10 CFR 70.50 and Appendix A to 10 CFR 70) will be incorporated into the facility emergency procedures. Reports will be made based upon IROFS failure regardless of whether or not safety limits are exceeded. An event will be reported within one hour of discovery, when the reporting criteria in 10 CFR 70, Appendix A, Paragraph (a) apply, or when it cannot be determined that these criteria do not apply.

The staff has reviewed the applicant's commitments to report NCS events to the NRC Operations Center and finds that they are acceptable because the applicant has adequately addressed all the acceptance criteria in NUREG-1520, Section 5.4.3.4.7(7).

5.3.8 ISA

The acceptance criteria for the NRC's NCS review of the ISA are contained in Sections 5.4.3.4.6 and 3.4 of NUREG-1520 (NRC, 2002).

The purpose of the ISA review is to determine that there is reasonable assurance that an ISA is being conducted in accordance with 10 CFR 70.62(c) which will ensure that the requirements of 10 CFR 70.61, as they relate to NCS, will be met. This review is divided into two parts: (1) review of commitments regarding the ISA and ISA Summary, and (2) review of the ISA Summary. The conclusions reached in this section are supported by an on-site review of ISA documents. The non-public portions of the ISA Summary and on-site ISA document review (horizontal and vertical slice) are discussed in Appendix A.

5.3.8.1 ISA Commitments

The acceptance criteria for the NRC's review of the NCS ISA commitments are contained in Sections 3.4.3.1.

ISA Methodology

The ISA conducted to support the design and application was performed by two teams – one for classified processes and one for unclassified processes. Some ISA team members are on both teams to ensure consistency. AES commits to have ISA teams which include at least one member with expertise in NCS that is appropriately trained and qualified as described in Section 5.3.2. During the operations phase, the NCS manager will be responsible for providing NCS support for the ISA (AES, 2010b).

The ISA teams use the hazard and operability (HAZOP) method to identify process hazards (AES, 2010b). Each process is divided into nodes. Guidewords are used to identify potential hazards associated with each node. A representative list of guidewords is found in Table 3.1-1 of the SAR (AES, 2010b). For each hazard, the ISA team will identify potential causes, consequences, and safeguards or design features which may prevent or mitigate the hazard. During the on-site review, AES stated (Ref. 0) that these safeguards and design features are not credited when evaluating the risk for the uncontrolled situation. Some of these safeguards and design features are credited when evaluating the controlled accident sequence, in which case they are designated as IROFS (AES, 2010b).

AES considers criticality accidents to be high consequence events per 10 CFR 70.61(b), and thus must be determined to be either not credible or highly unlikely. An event is considered to be credible unless it is determined to meet one of the following criteria:

1. An external event with a frequency of occurrence of less than once in a million years.
2. A process deviation that consists of a sequence of many unlikely human actions or errors for which there is no reason or motive.
3. A process deviation for which there is a convincing argument based on physical laws that it is not possible or unquestionably extremely unlikely.

In applying these criteria, AES will not rely on any control or design feature that could credibly fail to function.

For each credible criticality accident, AES will demonstrate that the accident is highly unlikely as defined in NUREG-1520 (NRC, 2002). An event is highly unlikely if the frequency of occurrence is less than 10^{-5} events per year (based on an approximate order of magnitude). IROFS are required to meet this criterion unless the uncontrolled event is highly unlikely. More details on the determination of event frequencies and IROFS failure rates are found in Chapter 3 of this SER.

The staff concludes that the method used for the HAZOP analysis is sufficient to ensure that all credible conditions which could impact NCS are considered.

Safe-by-Design IROFS

The SAR defines a special class of NCS controls as safe-by-design IROFS (AES, 2010b). A safe-by-design IROFS is a passive engineered control which by its geometry and configuration alone will prevent a criticality accident from occurring (AES, 2010b). This means that all NCS parameters, except geometry, enrichment, and interaction, are assumed to be in the optimum or worst case credible condition. By definition, it must be highly unlikely for a safe-by-design IROFS to fail in a manner that would cause a criticality accident (AES, 2010b). The most significant requirements to qualify as a safe-by-design IROFS are that:

1. The only credible failure mechanism that could result in a criticality would be to implement an improper design change.
2. Credible process deviations or events do not adversely impact performance of the safety function.
3. Quality Assurance Level 1 is applied to the feature.
4. No human actions are required for the component to perform its safety function.

The applicant considers failure of a safe-by-design IROFS due to a loss of configuration control to be highly unlikely (AES, 2010b). The staff determined that such an event could lead to criticality if an improperly modified safe-by-design IROFS were installed, used, and if an adequate margin of subcriticality (i.e., $k_{eff} < 0.95$) were not maintained for all credible process deviations or events. The staff determined, based on commitments made in the SAR (AES, 2010b), that such a failure would require several mistakes to be made in the configuration management process, which is sufficient to justify that this event is highly unlikely. Thus, the staff concludes that the configuration management program will ensure that failure or improper modification of safe-by-design IROFS will be highly unlikely, in accordance with 10 CFR 70.61. A HAZOP analysis is performed by the ISA team on each safe-by-design IROFS to support the conclusion that it is not credible for a process deviation or other event to adversely impact the safety function. The primary focus is on those events which could either change the geometry or configuration (i.e., physical arrangement and spacing) of the component. The information provided in Appendices B and C of the ISA Summary (AES, 2010b) indicates that AES will consider, as appropriate, the worst case credible process deviations and events. Process deviations considered in the HAZOP analysis include pressure changes, temperature changes, corrosion, erosion, bulging, leakage, and rupture. Other events considered include impacts, construction activities, maintenance activities, fires, earthquakes, flooding, and loss of utilities (AES, 2010b).

The analysis of safe-by-design components assumes the operator is using the required component and mobile vessels are adequately controlled. Events which could cause one of these assumptions to be violated are considered to be credible and are analyzed separately in the ISA.

Safe-by-design IROFS must also have *significant margin*, as defined in the SAR (AES, 2010b). Significant margin is defined as follows: (1) for components based on safe-by-design attributes of diameter, volume, or slab thickness, a margin of at least 10% during normal and credible abnormal conditions, and (2) for components based on a calculation, $k_{eff} < 0.95$. The staff determined that AES's application of significant margin does not provide any additional benefit to safety, in that it does not provide any actual safety margin beyond what is required for other passive controls, and it is therefore not relied upon to make any conclusions for the staff's review.

Based on the above, the staff concludes that components that meet the safe-by-design criteria will be capable of (1) preventing a criticality accident under normal and credible abnormal conditions, (2) maintaining an adequate margin of subcriticality for safety, and (3) ensuring that a criticality accident is highly unlikely to occur.

5.3.8.2 ISA Summary Review

The acceptance criteria for the NRC's NCS review of the ISA Summary are contained in Section 3.4.3.2 of NUREG-1520 (NRC, 2002).

The purpose of the ISA Summary review is to verify that the applicant complied with the requirements of 10 CFR 70.65(b) as it relates to NCS. The acceptance criteria (NRC, 2002) for the ISA Summary regarding completeness and description of processes, accident sequences, and IROFS is based upon the current stage of the design and not on the final design.

The ISA Summary requirements which are applicable to NCS are 10 CFR 70.65(b)(3) – (9). Since the ISA Summary contains proprietary, security-related, and export controlled information, the review of information required to meet 10 CFR 70.65(b)(3), (4), (6), and (8) is discussed in Appendix A and summarized in this Section. Review of CAAS information required by 70.65(b)(4) is discussed in Section 5.3.6 of this SER. The requirements of 70.65(b)(5) and (9) are redundant with the commitments discussed in Section 5.3.8.1.

The applicant was unable to provide classified information during the review. The ISA Summary does not have a classified section; some safety analyses which support the ISA Summary are classified. The applicant was able to provide the reviewer with an unclassified summary of this analysis, when it was requested.

10 CFR 70.65(b)(3): Process Description

The staff has reviewed each of the process descriptions in the ISA Summary and finds them acceptable because they adequately address the acceptance criteria in NUREG-1520, Section 3.4.3.2(3) as it relates to NCS. Specifically, the applicant provided sufficient information in the ISA Summary to determine where criticality hazards exist and how operations might impact NCS.

10 CFR 70.65(b)(4): Performance Requirements

The regulation in 10 CFR 70.61(d) requires that the risk of nuclear criticality accidents be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical. At the proposed EREF, a criticality hazard will exist only in those areas where enriched uranium is present under normal or credible abnormal conditions (AES, 2010b). For the uranium enrichment levels that will be present at the proposed EREF, a moderator is also required for a criticality to occur. Common moderators include water, hydrogen fluoride (HF), and hydrocarbon oils or lubricants. At low enriched fuel facilities like the proposed EREF, operations where a moderator is intentionally added to enriched uranium require the most NCS scrutiny. However, possible external sources of a moderator must be considered as a credible abnormal condition, in accordance with 10 CFR 70.61(d), in the NCS analysis for all operations. At the proposed EREF, the only operations where significant moderation of fissile material will normally occur are in the decontamination workshop and the liquid effluent collection and treatment system (AES, 2010b).

Large portions of the proposed facility rely upon safe-by-design attributes to ensure that processes are subcritical under normal and credible abnormal conditions (AES, 2010b). The staff determined that subcriticality in equipment with safe-by-design attributes relies upon controlling equipment geometry (i.e., one or more physical dimensions), enrichment, and interaction. All other parameters are assumed to be in the worst case credible condition for NCS purposes. 10 CFR 70.61(e) requires that controls on parameters needed to ensure subcriticality be designated as IROFS. The staff determined during its ISA Summary review that IROFS have been designated which will ensure that the enrichment limits will not be exceeded for any the safe-by-design equipment. The staff also determined that geometry control is part of the safe-by-design IROFS designations. The staff found that some interaction controls were part of a safe-by-design IROFS, and that some were captured by administrative IROFS. The staff did not assess whether all interaction controls had been designated as IROFS because the staff's review of the ISA Summary was on a sampling basis.

AES has identified loss of configuration control as an initiating event that could cause a safe-by-design IROFS to fail (AES, 2010b). The staff notes that configuration management is a management measure that is applied to all IROFS to ensure their availability and reliability. Typically, failure of configuration management is not identified as a separate accident initiator in the ISA since it is already factored into the availability and reliability of the IROFS. Therefore, consideration of its failure is already covered in determining that failure of safe-by-design components is highly unlikely. The applicant has made commitments regarding its configuration management program (some specific to safe-by-design IROFS) in its SAR (AES, 2010b). Upon review, the staff determined that those commitments are sufficient to ensure that the failure of these safe-by-design IROFS is at least highly unlikely. Thus, the staff determined that it is not necessary for AES to specifically identify failure of configuration management as an accident sequence.

Because AES considered improper design changes as separate accident sequences, it had identified safe-by-design IROFS as sole IROFS. However, the staff disagrees, based on the determination that such failures of configuration management are not required to be considered as separate accident sequences. In addition, in sequences involving the failure of configuration management, it is the configuration management program that ensures safety, and not the safe-by-design IROFS itself. Designating these controls as sole IROFS is conservative, so this does not constitute a safety or regulatory concern.

The staff has reviewed the information provided in the ISA Summary and supporting ISA documents demonstrating compliance with the performance requirements of 10 CFR 70.61 and finds it is acceptable because it adequately addresses the acceptance criteria in NUREG-1520, Sections 3.4.3.2(4)(a) and (b) as they relate to NCS. Specifically, the staff has concludes that for the current state of facility design (1) each process has been evaluated for NCS, (2) criticality accidents will be at least highly unlikely, and (3) each process will be adequately subcritical under normal and credible abnormal conditions.

<u>10 CFR 70.65(b)(4): Baseline Design Criteria and Defense-in-Depth</u>

As part of the defense-in-depth requirements, the applicant must show a preference for engineered over administrative controls in the design of the facility. The widespread application of safe-by-design IROFS demonstrates this preference since these are passive engineered controls that require no human actions to prevent a criticality accident.

Another part of the defense-in-depth requirements is that the facility design should incorporate features which reduce challenges to IROFS. For many reasons, the normal facility operations requirements greatly reduce the challenges to IROFS for most operations. The parameter values at which the safe-by-design components are analyzed represent very conservative process conditions which are unlikely to ever be approached. For example, the feed, product, and main processes all use UF_6, which necessarily requires that moderator be excluded. Without a moderator, a criticality accident is not possible for the proposed enrichment limit. This ensures that the geometry control associated with safe-by-design components will seldom be relied on to ensure subcriticality. The staff has reviewed the information provided in the SAR and ISA Summary as it relates to the defense-in-depth practices required by 10 CFR 70.64(b). The staff finds this information is acceptable because it adequately addresses the acceptance criteria in NUREG-1520, Sections 3.4.3.2(4)(d) as it relates to NCS. Specifically, the NRC staff concludes that the applicant has designed the process with a preference for engineered controls and will operate it in a manner which limits the challenges to NCS IROFS.

<u>10 CFR 70.65(b)(6) and (8): IROFS and Sole IROFS</u>

The ISA Summary identifies four generic safe-by-design IROFS. The first three are for components where NCS is assured because the diameter, slab thickness, or volume of the component is less than the safe geometry values for 6 wt% ^{235}U (SAR Table 5.1-1). The other safe-by-design IROFS is for components which are demonstrated to be subcritical with an explicit NCS analysis (AES, 2010b). Appendix B and C of the ISA Summary describes the specific components that will be categorized as safe-by-design IROFS.

The safe-by-design IROFS have been designated as sole IROFS, because they are the only IROFS listed for a criticality accident initiated by a loss of configuration control. The staff determined that the loss of configuration control is not required to be listed as an accident initiator in the ISA summary, for reasons stated above. Therefore, the staff does not concur that safe-by-design IROFS are necessarily sole IROFS. They are not sole IROFS if the only basis for calling them sole IROFS is that a hypothetical design change could theoretically result in criticality. Nevertheless, the staff finds that there is no safety or regulatory concern with the applicant's conservatively designating them as sole IROFS, since this would result in a higher level of regulatory oversight. The ISA Summary identifies several enhanced administrative controls which prevent criticality accidents (AES, 2010b). Each of these IROFS requires an independent verification, which means that a second qualified individual will independently ensure that the required task is performed correctly. A failure probability index of -3 is assigned to these IROFS, which is consistent with what has been approved for other facilities.

The ISA Summary identifies six enhanced administrative controls as sole IROFS preventing certain criticality accident sequences (AES, 2010b). For five out of these six sole IROFS, several repeat failures would have to occur before a criticality is possible. The remaining sole IROFS is sufficient, along with an initiating event with a frequency index of at least -2, for the sequence to meet the criteria for demonstrating that an event is highly unlikely. (The applicant's risk index methodology is discussed and evaluated in Chapter 3 of this SER.)

The staff has reviewed the descriptive lists of NCS IROFS and sole IROFS in the ISA Summary and finds them acceptable because they adequately address the acceptance criteria in NUREG-1520, Sections 3.4.3.2(6) and (8) as it relates to NCS. For each criticality accident sequence identified in the ISA Summary, the list included the IROFS needed to render the sequence highly unlikely. The descriptions were sufficient to understand how the IROFS would prevent a criticality from occurring.

5.3.8.3 Conclusion

The staff has reviewed the EREF ISA methodology and ISA Summary and has reasonable assurance that the applicant has conducted an ISA, based upon the current level of design, that:

(1) Identified all credible criticality accident scenarios.

(2) Identified controls to prevent each credible criticality accident sufficient to meet the performance requirements of 10 CFR 70.61(b) and (d) and provide for double contingency protection and defense-in-depth as required by 10 CFR 70.64(a)(9) and 70.64(b).

(3) Designated as IROFS, as required by 10 CFR 70.61(e), the controls relied on to meet the performance requirements as they relate to NCS.

(4) Identified management measures which will be applied to NCS-related IROFS to ensure they will be available and reliable.

Based upon the review of the ISA commitments, ISA Summary, and on-site ISA documents, the staff concludes that an ISA will be performed and maintained in accordance with 10 CFR 70.62(c) such that the requirements of 10 CFR 70.61, as they relate to NCS, will be met. The staff concludes that the ISA methodology committed to in the SAR is sufficient to identify all credible nuclear criticality accidents and the IROFS necessary to make such events highly unlikely.

5.3.9 Facility Changes and Configuration Management

Configuration management will be provided during all phases of facility operation (AES, 2010b). In accordance with approved procedures, design changes undergo formal review which will include assessing impacts to the ISA. As discussed in Section 5.2.1.6 of the SAR, any change to the facility or activities of personnel that may impact the handling, use, or storage of uranium requires an NCS evaluation and, if necessary, an NCS analysis to be prepared by an NCS engineer and approved by an NCS manager. Prior to implementing a change it must be determined that the entire process is subcritical under normal and credible abnormal conditions. As part of configuration management, the applicant must assess whether NRC approval is required before a change is implemented. Changes to the site, structures, processes, systems, equipment, components, computer systems, and activities of personnel may be changed without prior NRC approval if the criteria in 10 CFR 70.72(c) are met.

NRC approval is required before AES can make a change which alters a sole IROFS identified in the ISA Summary that is necessary to prevent a criticality accident (10 CFR 70.72(c)(3)). Altering a sole IROFS consists of making a change that modifies, positively or negatively, any of the attributes associated with the safety function of the IROFS.

NRC staff has determined that the SAR contains the information and commitments required by 10 CFR 70.22(a) and 70.65(a) to be included as part of the license application. In addition, 10 CFR 70.61(d) requires that the margin of subcriticality for safety to be approved by the NRC. The information and commitments regarding NCS methodologies and technical practices define the margin of subcriticality that AES will use. Other NCS, ISA, and management measure program information and commitments are necessary to ensure that the margin of subcriticality will be maintained. As discussed in Section 1.2.4.2.2 of this SER, the applicant has requested authorization to make changes to license commitments. The authorization has two parts covering changes requiring prior approval and changes not requiring prior approval. Changes to the license application that will require prior NRC approval are those that decrease the effectiveness of any safety commitments. For a change that decreases the effectiveness of any safety commitment, the applicant will submit the change to the NRC as a license amendment and will not implement the change until NRC has reviewed and approved the amendment. Changes to the license application that could be made without prior NRC approval are those for which there is no degradation in the safety commitments in the license and for which the change, test, or activity does not conflict with any condition specifically stated in the license. The staff will impose the license condition stated in Section 1.2.4.2.2 of this SER so that commitments in the SAR, including those required by 10 CFR 70.22(a) and 70.65(a), cannot be changed except by license amendment.

The staff has reviewed the applicant's commitments for configuration management as it relates to NCS and finds that it is acceptable because the applicant has adequately addressed the acceptance criteria in NUREG-1520, Section 5.4.3.4.7. In addition, the staff finds the applicant's method for assessing changes to safe-by-design IROFS to determine if they are alterations per 10 CFR 70.72(c)(3) is acceptable. The staff does not find the applicant's initial proposal to maintain the SAR as a "living document" under the configuration management to be acceptable. Instead, the staff will impose a license condition covering changes requiring prior NRC approval and changes not requiring prior approval to ensure that commitments in the SAR, including those required by 10 CFR 70.22(a) and 70.65(a), cannot be changed except by license amendment. This license condition is provided in Section 1.2.4.2.2 of this SER.

5.4 Evaluation Findings

The staff has reviewed the SAR, ISA Summary, and on-site ISA documents for the proposed EREF as it relates to NCS. Provided that the applicant implements and maintains its safety programs as described in the SAR, the staff has reasonable assurance that:

(1) The applicant will have a staff of managers, supervisors, engineers, process operators, and other support personnel who are qualified to develop, implement, and maintain the NCS program.

(2) The applicant's conduct of operations will be based on NCS methodologies and technical practices, which will ensure that fissile material will be possessed, stored, and used safely according to the requirements in 10 CFR Part 70.

(3) The applicant's NCS methodologies and technical practices will provide an adequate margin of subcriticality for safety as required by 10 CFR 70.61(d).

(4) The applicant's safety programs and management measures will ensure that the margin of subcriticality is maintained such that processes will meet the subcriticality requirements of 10 CFR 70.61(d) and the requirements for new facilities specified in 10 CFR 70.64.

(5) The applicant will develop, implement, and maintain a CAAS in accordance with the requirements in 10 CFR 70.24 and the facility emergency management program.

(6) The applicant will perform and maintain an ISA in accordance with 10 CFR 70.62(c) such that the requirements of 10 CFR 70.61, as they relate to NCS, will be met.

NRC staff has determined that the SAR contains commitments required by 10 CFR 70.22(a) and 70.65(a) and needed to define the approved margin of subcriticality to be included as part of the license application. The staff will impose the license condition provided in Section 1.2.4.2.2 of this SER so that commitments cannot be changed except by license amendment.

The NRC staff concludes that if the proposed EREF NCS program is implemented in accordance with the statements made in the SAR it will meet the requirements of 10 CFR Part 70 and provide reasonable assurance for the protection of public health and safety, including workers and the environment.

5.5 References

(AES, 2010a) AREVA Enrichment Services LLC, Eagle Rock Enrichment Facility, "Supplemental Response to Requests for Additional Information," May 12, 2010.

(AES, 2010b) AREVA Enrichment Services, Eagle Rock Enrichment Facility, "Revision 2 to License Application for the Eagle Rock Enrichment Facility," April 30, 2010.

(AES, 2009a) AREVA Enrichment Services, Eagle Rock Enrichment Facility, "Response to Requests for Additional Information—AREVA Enrichment Services LLC License Application for the Eagle Rock Enrichment Facility," September 28, 2009.

(AES, 2009b) AREVA Enrichment Services, Eagle Rock Enrichment Facility, "Revision 1 to License Application for the Eagle Rock Enrichment Facility," April 23, 2009.

(AES, 2008a) AREVA Enrichment Services, Eagle Rock Enrichment Facility, "Application for a Uranium Enrichment Facility," Rev. 0, December 30, 2008.

(AES, 2008b) AREVA Enrichment Services, Eagle Rock Enrichment Facility, "MONK8A Validation and Verification," December 2008.

(AES, undated) AREVA Enrichment Services, Eagle Rock Enrichment Facility, "MONK8A Validation and Verification," Rev. 2, undated.

(ANSI/ANS, 2005) American National Standards Institute/American Nuclear Society, ANSI/ANS-8.19-2005, "Administrative Practices for Nuclear Criticality Safety."

(ANSI/ANS, 1998) American National Standards Institute/American Nuclear Society, ANSI/ANS-8.1-1998, "Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors."

(ANSI/ANS, 1997a) American National Standards Institute/American Nuclear Society, ANSI/ANS-8.3-1997, "Criticality Accident Alarm System."

(ANSI/ANS, 1997b) American National Standards Institute/American Nuclear Society, ANSI/ANS-8.22-1997, "Nuclear Criticality Safety Training Based on Limiting and Controlling Moderators."

(ANSI/ANS, 1997c) American National Standards Institute/American Nuclear Society, ANSI/ANS-8.23-1997, "Nuclear Criticality Accident Emergency Planning and Response."

(ANSI/ANS, 1991) American National Standards Institute/American Nuclear Society, ANSI/ANS-8.20-1991, "Nuclear Criticality Safety Training"

(ANSI/ANS, 1995) American National Standards Institute/American Nuclear Society, ANSI/ANS-8.21-1995, "Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors."

(ANSI/ANS, 1983) American National Standards Institute/American Nuclear Society, ANSI/ANS-8.10-1983, "Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement."

(NRC, 2010a) U.S. Nuclear Regulatory Commission, Teleconference Summary, February 25, 2010.

(NRC, 2010b) U.S. Nuclear Regulatory Commission, Teleconference Summary, March 8, 2010.

(NRC, 2010c) U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-3037, "Guidance for Fuel Cycle Facility Change Processes," June 2009.

(NRC, 2009) U.S. Nuclear Regulatory Commission, July 13–15, 2009, Meeting Summary, "AREVA Enrichment Services Eagle Rock License Application - Integrated Safety Analysis In-Office Review," 2009.

(NRC, 2006) U.S. Nuclear Regulatory Commission, FCSS-ISG-10, "Justification for Minimum Margin of Subcriticality for Safety," 2006.

(NRC, 2005) U.S. Nuclear Regulatory Commission, Regulatory Guide 3.71, "Nuclear Criticality Safety Standards for Fuels and Material Facilities," Rev. 1, 2005.

(NRC, 2002) U.S. Nuclear Regulatory Commission, NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," March 2002.

(NRC, 2001) U.S. Nuclear Regulatory Commission, NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," January 2001.

CHAPTER 6.0 CHEMICAL PROCESS SAFETY

The purpose of the U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's chemical safety program and the design of the proposed Eagle Rock Enrichment Facility (EREF) is to evaluate whether the applicant's chemical process safety program will adequately protect workers, the public, and the environment during normal operations against chemical hazards of licensed material and its byproducts. The chemical process safety program and the facility's design must also protect against facility conditions and/or operator actions that can affect the safety of licensed materials and thus, present an increased chemical risk.

6.1 Regulatory Requirements

The regulatory bases for the review are the general and additional contents of an application that addresses chemical process safety, as required by Title 10 of the *Code of Federal Regulations* (10 CFR) 70.22 and 70.65. In addition, the chemical process safety review should provide reasonable assurance of compliance with 10 CFR 70.61, 70.62, and 70.64.

6.2 Regulatory Guidance and Acceptance Criteria

The guidance applicable to NRC's review of chemical process safety for the proposed facility is contained in Chapter 6 of "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 (NRC, 2002). This chapter is applicable in its entirety. The staff also uses the following as guidance documents for this review: NUREG-1601, "Chemical Process Safety at Fuel Cycle Facilities," (NRC, 1997a), NUREG-1513, "Integrated Safety Analysis Guidance Document," (NRC, 2001). The acceptance criteria applicable to this review are contained in Section 6.4.3 of NUREG-1520 (NRC, 2002).

6.3 Staff Review and Analysis

The NRC staff reviewed the Safety Analysis Report (SAR) (AES, 2009a) and the Integrated Safety Analysis (ISA) Summary (AES, 2009b), submitted by the applicant and considered the following areas:

1. Process Description;
2. Chemical Accident Sequences;
3. Chemical Accident Consequences;
4. Chemical Process items relied on for safety (IROFS)
5. Management Measures;
6. Emergency Management; and
7. Baseline Design Criteria (BDC).

The staff reviewed the applicant's responses to requests for additional information and the ISA documents during an in-office review (NRC, 2009), as necessary, to have a better understanding of the process and safety requirements. The staff evaluated the information to determine if the facility's design complied with the BDC and defense-in-depth requirements specified in 10 CFR 70.64(a) and 70.64(b), respectively. Compliance with these regulations is discussed in more detail in Chapter 3, Appendix A, and Appendix B of this Safety Evaluation Report (SER). The staff's evaluation and general information about the proposed EREF process are summarized in the following sections.

6.3.1 Process Description

The applicant describes the gas centrifuge process in Section 1.1.3 of the SAR (AES, 2009a). The process is entirely physical in nature and mechanically separates the uranium isotopes using a fast rotating cylinder (centrifuge). This separation occurs because there is a difference in centrifugal forces between the isotopes of uranium since they have different molecular weights. Neither nuclear reactions nor significant chemical changes are expected to occur during normal operations. The feed, product, and tails streams are all in the form of uranium hexafluoride (UF_6).

6.3.1.1 Gas Centrifuge Process

The enrichment process proposed by the applicant, housed in the Separations Building, is comprised of four major systems: a UF_6 Feed System, a Cascade System, a Product Take-off System, and a Tails Take-off System. Other product related functions include the Product Liquid Sampling System and the Product Blending System. Supporting functions include sample analysis, equipment decontamination and rebuild, liquid effluent treatment, and solid waste management (AES, 2009b).

The major equipment used in the UF_6 Feed System are the Solid Feed Stations. UF_6 will be delivered to the proposed EREF in the American Nuclear Standards Institute (ANSI) N14.1 (ANSI, 1995) standard 48Y international transit cylinders (AES, 2009a). Feed cylinders will be loaded into Solid Feed Stations; vented for removal of light gases, primarily air and hydrogen fluoride (HF); and heated air will be circulated around the feed cylinder to sublime the UF_6.

The light gases and UF_6 gas generated during feed purification will be routed to the Feed Purification Subsystem (AES, 2009b). The major pieces of equipment in the Feed Purification Subsystem will be UF_6 Cold Traps, a Vacuum Pump/Chemical Trap Set, and a Low-Temperature Take-off Station (LTTS) (AES, 2009b). The Feed Purification Subsystem will remove any light gases, such as air and HF from the UF_6 feed prior to introduction into the cascades (AES, 2009b). The UF_6 will be captured on UF_6 Cold Traps and ultimately recycled as feed, while HF will be captured on chemical traps (AES, 2009b).

After purification, the gaseous UF_6 will be flow controlled through a pressure control system for distribution to the Cascade System at subatmospheric pressure. Individual centrifuges are not able to produce the desired product and tails concentration in a single step. Therefore, the centrifuges are grouped in series and in parallel to form arrays known as cascades. A typical cascade is compromised of many centrifuges. Each centrifuge has a thin-walled, vertical, cylindrically shaped rotor that spins around a central post within an outer casing. Feed enters, and product and tails leave the centrifuge through the central post. Control valves, restriction orifices, and controllers provide uniform flow of product and tails.

Depleted UF_6 exiting the cascades will be transported from the high vacuum of the centrifuge for desublimation into cylinders at subatmospheric pressure. Tails material will be desublimed into

48Y cylinders. The primary equipment of the Tails Take-off System is the vacuum pump and the Tails LTTS. Chilled air will flow over cylinders in the Tails LTTS to effect desublimation. Filling of the cylinders will be monitored with a load cell system, and filled cylinders will be transferred outdoors to the Full Tails Cylinder Storage Pad.

Enriched UF_6 from the cascades will be desublimed in a Product Take-off System comprised of vacuum pumps, Product LTTS, UF_6 Cold Traps, and Vacuum Pumps/Chemical Trap Sets (AES, 2009b). Product material will be desublimed into 48Y or smaller 30B cylinders (AES, 2009b). The vacuum pumps will transport the UF_6 from the cascades to the Product LTTS at subatmospheric pressure (AES, 2009b). The heat of desublimation of the UF_6 will be removed by cooling air routed through the LTTS (AES, 2009b). The product stream normally contains small amounts of light gases that may have passed through the centrifuges. Therefore, a UF_6 Cold Trap and Vacuum Pump/Chemical Trap Sets will be provided to vent these gases from the product cylinder (AES, 2009b). Any UF_6 captured in the cold trap will be periodically transferred to another product cylinder for use as product or blending stock. Filling of the product cylinders will be monitored with a load cell system, and filled cylinders will be transferred to the Product Liquid Sampling System for sampling.

Sampling will be performed to verify product assay level (weight percentage ^{235}U) (AES, 2009b). The Product Liquid Sampling Autoclave is an electrically heated, closed pressure vessel used to liquefy and allow collection of a sample (AES, 2009b). After the UF_6 is liquefied, the contents of the product cylinder will be allowed to homogenize for a period of time. The autoclave will be fitted with a hydraulic tilting mechanism that elevates one end of the autoclave so that liquid UF_6 will pour into a sampling manifold connected to the cylinder valve. After sampling, the autoclave will be brought back to the horizontal position; and the autoclave and cylinder will be cooled down by a chiller unit mounted on the interior of the pressure vessel with the refrigerant compression and heat rejection components on the exterior.

The facility will have the capability to blend enriched UF_6 from donor cylinders of different assays into a product receiver cylinder. The Product Blending System will be comprised of Blending Donor Stations for two donor cylinders and Blending Receiver Stations for the receiver cylinder. The Donor Stations are similar to the Solid Feed Stations and the Receiver Stations are similar to the LTTS. The Solid Feed Stations and the LTTS are described above.

The entire enrichment and product blending processes will operate at subatmospheric pressure with the exception of product liquid sampling operations. This safety feature helps ensure that releases of UF_6 or HF are minimized because leakage would be typically inward into the system. During sampling operations, the UF_6 will be liquefied within an autoclave. The autoclave is an American Society of Mechanical Engineers (ASME), Section VIII, Division I, "Boiler and Pressure Vessel Code" rated pressure vessel that serves as a secondary containment for the UF_6 product cylinder while the UF_6 is in a liquid state.

6.3.1.2 Chemical Screening and Classification

The applicant classifies all site chemicals into one of three categories: Chemicals of Concern (EREF Class 1), Interaction Chemicals (EREF Class 2), or Incidental Chemicals (EREF Class 3).

Chemicals of Concern (EREF Class 1)

The applicant determined the Chemicals of Concern (EREF Class 1) based on one or more characteristics of the chemical and/or the quantity in storage/use at the facility. For licensed materials or hazardous chemicals produced from licensed materials, chemicals of concern are those that, in case of release, have the potential to exceed any of the concentrations defined in 10 CFR 70.61(b) and 70.61(c). UF_6 is the only licensed material-related chemical of concern (EREF Class 1) that will be used at the facility.

Chemicals of concern that are not related to licensed materials are those that are listed and handled above threshold quantities set forth by either of the following standards: 29 CFR 1910.119, "Occupational Safety and Health Administration (OSHA) Process Safety Management," or 40 CFR Part 68, "U.S. Environmental Protection Agency (EPA) Risk Management Program."

These chemicals represent, based on their inherent toxic, reactive, or flammable properties, a potential for a severe chemical release or acute chemical exposure to an individual that:

1. Could endanger the life of a worker or

2. Could lead to irreversible or other serious, long-lasting health effects to any individual located outside the controlled area.

The applicant states in Section 6.1.1.1 of the SAR (AES, 2009a) that there are no non-licensed chemicals of concern at the facility.

Interaction Chemicals (EREF Class 2)

Interaction chemicals (EREF Class 2) are those chemicals/chemical systems that require evaluation for the potential to precipitate or propagate accidents in chemical of concern (EREF Class 1) systems, but by themselves are not chemicals of concern. The EREF Class 2 chemicals are listed below (AES, 2009a):

- Perfluorinated polyether (PFPE) oil
- Activated carbon (C)
- Alumimum oxide (Al_2O_3)
- Sodium fluoride (NaF)
- Citric acid ($C_6H_8O_4$)
- Nitrogen (N_2)
- Polydimethylsiloxane (silicone oil)

Incidental Chemicals (EREF Class 3)

The facility will use other chemicals that are neither chemicals of concern nor interaction chemicals. Some of these incidental chemicals (EREF Class 3) include those that have the potential to result in injurious occupational or environmental exposure - but represent no potential for acute exposure to the public; and which, via their nature, quantity, and/or use have no potential for impacting chemicals of concern (EREF Class 1). EREF Class 3 chemicals are listed below (AES, 2009a), but do not impact the performance requirements of 10 CFR 70.61.

- Paper, polymers
- Potassium hydroxide (KOH)
- Phosphate
- Scrap metals
- Sodium hydroxide (NaOH)
- Hydrocarbon oils / greases
- Hydrocarbon sludges
- Methylene chloride (CH_2Cl_2)
- Hydrocarbon / polar solvents and liquids
- Nitric acid (HNO_3)
- Hydrofluoric acid ($HF_{(aq)}$)
- Hydrogen peroxide (H_2O_2)
- Sulfuric acid (H_2SO_4)
- Phosphoric acid (H_3PO_4)
- Diesel fuel
- Deionized water (H_2O)
- Hydrofluorocarbons
- Propane (C_3H_8)
- Hydrogen (H_2)
- Acetylene (C_2H_2)
- Oxygen (O_2)
- Argon (Ar)
- Helium (He)

6.3.1.3 Hazardous Chemicals and Chemical Interactions

The only chemical present in significant quantities in the facility is UF_6, and it constitutes the main hazard in the EREF. Any UF_6 that is released to the environment will react exothermically with moisture in the air, producing solid uranyl fluoride (UO_2F_2) and HF gas. The reaction of gaseous UF_6 with water vapor at elevated temperatures is shown in Equation 1.

$$UF_{6\,(gas)} + 2H_2O_{(vapor)} \rightarrow UO_2F_{2\,(solid)} + 4HF_{(gas)} + \text{heat} \qquad \text{(Eq. 1)}$$

At room temperature, depending on the relative humidity of the air, the products of this reaction are UO_2F_2, hydrates and HF-H_2O fog, which will be seen as a white cloud. A typical reaction with excess water is given in Equation 2.

$$UF_{6\,(gas)} + (2+4x)H_2O_{(vapor)} \rightarrow UO_2F_2 * 2H_2O_{\,(solid)} + 4HF * xH_2O_{(fog)} + \text{heat} \qquad \text{(Eq. 2)}$$

These reactions, if occurring in the gaseous phase at ambient or higher temperatures, are very rapid, near instantaneous. Usually, UO_2F_2 compounds are deposited or precipitated close to the point of the release.

6.3.1.4 UF_6 and Interaction Chemicals

The following discussion describes the potential interactions between UF_6 and EREF Class 2 chemicals, halocarbon refrigerants, and centrifuge cooling water.

Perfluorinated Polyether (PFPE) Oil

Besides the interaction with moisture in the air, UF_6 can react exothermically with hydrocarbons. Gaseous UF_6 reacts with hydrocarbons to form a black residue of uranium-carbon compounds. Hydrocarbons can be explosively oxidized if they are mixed with UF_6 in the liquid phase or at elevated temperatures. Fluorinated hydrocarbons do not react with UF_6, thus the applicant will not be using non-fluorinated hydrocarbon lubricants in any UF_6 system at the EREF.

The UF_6 vacuum pumps are lubricated using PFPE oil. PFPE oil is inert, fully fluorinated, and does not react with UF_6 under any operating conditions.

Small quantities of uranium compounds and traces of hydrocarbons may be contained in the PFPE oil. The UF_6 degrades in the oil or reacts with trace hydrocarbons to form crystalline compounds - primarily UO_2F_2 and uranium tetrafluoride (UF_4) particles - that gradually thicken the oil and reduce pump capacity.

The used PFPE oil will not be recovered for reuse at the proposed EREF. Instead, the used PFPE oil will be collected, packaged, and shipped offsite for disposal at a licensed, low-level radioactive waste facility (AES, 2009b).

Activated Carbon, Aluminum Oxide, and Sodium Fluoride

The applicant will use chemisorption for the removal of UF_6 and HF from gaseous effluent streams. Chemisorption occurs when a gas is captured on the surface of an activated solid caused by the chemical reaction between the gas and the activated solid. The applicant also will use chemisorption to remove oil mist from vacuum pumps operating upstream gaseous effluents ventilation systems. The applicant places absorbent materials on stationery beds in chemical traps downstream of the various cold traps. These materials capture HF and the trace amounts of UF_6 that escape desublimation during feed purification or during venting of residual UF_6 contained in hoses or piping that is bled down before disconnection (AES, 2009a).

The applicant will be using two different types of traps. The first type of trap contains a charge of activated carbon to capture small amounts of UF_6 that escape desublimation (AES, 2009a). A second type of trap is used to absorb HF, since HF is not fully absorbed on carbon at low pressure (AES, 2009a). The second type of trap contains a charge of aluminum oxide (Al_2O_3) (AES, 2009a).

Activated carbon cannot be used in the Dump System because the relatively high UF_6 flow rates during this non-routine operation could lead to severe overheating. A chemical trap containing sodium fluoride (NaF) will be installed in the contingency dump flow path to trap UF_6. NaF will be used because the heat of UF_6 chemisorption on NaF is significantly lower than the heat of UF_6 chemisorption on activated carbon.

Citric Acid

The applicant will use citric acid to decontaminate components (e.g., pumps, valves, and piping), sample bottles, and flexible connectors from residual UO_2F_2 compound layers that are present in the surfaces once these are removed from the process areas (AES, 2009a). The reaction of the uranium compounds with the citric acid solution produces various uranyl citrate complexes. The applicant will use personnel protective features for the safe handling of decontamination chemicals and byproducts.

Nitrogen

The applicant will use gaseous nitrogen in the UF_6 systems for purging and filling lines that have been exposed to the atmosphere for any of several reasons, including: connection and disconnection of cylinders, preparing lines/components for maintenance, providing an air-excluding gaseous inventory for system vacuum pumps, and filling the interstitial space of the liquid sampling autoclave (secondary containment) before cylinder liquefaction (AES, 2009a). Nitrogen is not reactive with UF_6 in any operational condition at the facility.

Silicone Oil

The applicant will use silicone oil as a heat exchange medium for heating/chilling of various cold traps. In all cases, this oil will be external to the UF_6 process stream and is not expected to interact with UF_6 (AES, 2009a).

Halocarbon Refrigerants

The applicant will use halocarbon refrigerants (including R23 trifluoromethane, R404A fluoromethane blend, and R507 penta/trifluoromethane) in individual package chillers that will provide cooling of UF_6 cylinders and/or silicone oil heat exchange media for take-off stations and cold traps (AES, 2009a). The applicant will use these halocarbon refrigerants because they have good heat transfer properties, they satisfy environmental restrictions regarding ozone depletion, and they are non-flammable. In all cases, the halocarbon refrigerants will be external to the UF_6 process stream and are not expected to interact with UF_6.

Centrifuge Cooling Water

The applicant will provide centrifuge cooling water from the Centrifuge Cooling Water Distribution System (AES, 2009a). The applicant will use this system to provide a supply of deionized cooling water to the cooling coils of the centrifuge (AES, 2009a). This system will provide stringent control over the operating temperature of the centrifuge to enable its efficient operation. In all cases, centrifuge cooling water will be external to the UF_6 process stream and is not expected to interact with UF_6.

6.3.1.5 UF_6 and Construction Materials

UF_6 is a fluorinating agent that reacts with most metals. The reaction between UF_6 and metals such as nickel, copper, and aluminum will produce a protective fluoride film over the metal that inhibits further reaction. These materials are therefore relatively inert to UF_6 corrosion after passivation and are suitable for UF_6 service.

Resistant metals such as stainless steel will be used in valve bellows and flex hoses. Aluminum piping will be bent to minimize the use of fittings. Connections will be welded to minimize the use of flanges and gaskets. The use of sealant materials will be minimized to reduce the number of potential leak paths.
Non-metallic materials are required to seal connections in UF_6 systems to facilitate valve and instrument replacement as well as cylinder connections. The applicant will confirm that all gasketing and packing material are appropriate for UF_6 services. Typical materials that are resistant to UF_6 through the range of the facility operating conditions include butyl rubber, Teflon, Viton, and Kel-F.[6]

The applicant selected materials of construction compatible with the process. The material compatibility is demonstrated by the corrosion rates provided in Table 6.2-2 of the SAR (AES, 2009a). For instance, take the corrosion rate of stainless steel at 100°C (i.e, 0.03 millimeters (mm) per year) (AES, 2009a) and the nominal wall thickness of the UF_6 piping from Table 6.2-3 of the SAR (i.e., 3.7 mm) which will be constructed of aluminum and stainless steel (AES, 2009a). These data suggest that it would take approximately 123 years to corrode the nominal wall thickness of the UF_6 piping. Therefore, these corrosion rates indicate that these materials are acceptable for UF_6 service over the life of the facility.

The cylinders used to store and transport UF_6 are made out of carbon steel and are standard Department of Transportation (DOT)-approved containers, designed and fabricated in accordance with ANSI N14.1 (ANSI, 1995) (AES, 2009a). Tails cylinders will be stored outside in open air where they will be exposed to the elements. Feed and product cylinders will be subject to short duration exterior storage (months) and will be inspected in accordance with requirements of DOT regulations upon receipt and prior to shipment.

The nominal and minimum wall thicknesses for cylinders are listed in Table 6.2-3 of the SAR (AES, 2009a). The carbon steel storage cylinders are painted to provide a corrosion barrier to external elements. Tails cylinders will be periodically inspected to assess corrosion and corrosion rate. This provides reasonable assurance that there is not undue risk to members of the public from the storage of tails cylinders.

6.3.1.6 Process Description Conclusion

The staff finds that the applicant has provided process descriptions that are sufficiently detailed to allow an understanding of the chemical process hazards, and the chemical hazards that could result from potential chemical interactions. The rational for these findings is that the applicant provided expected process operating conditions, interactions between chemicals, and interactions between process chemicals and the material of the equipment that will be used to contain these process chemicals. Also, the information provided by the applicant allowed the

[6]Teflon, Viton, and Kel-F are tradenames for different fluorocarbon-based polymers.

development of potential accident sequences. Therefore, the information that the applicant provided, as described above, meets the guidance in Section 6.4.3.1 bullets (1) and (2) of NUREG-1520 (NRC, 2002) and is acceptable.

6.3.2 Chemical Accident Sequences

The ISA Summary Section 3.7, Table 3.7-1 lists the chemical accident sequences, and Table 3.7-2 provides a narrative description (AES, 2009b). The applicant determined that the consequences of the chemical accident sequences identified in these tables have the potential to exceed the performance requirements of 10 CFR 70.61. The chemical accident sequences covered the following areas:

(a) UF_6 Feed System,
(b) Tails Take-off System,
(c) Product Take-off System,
(d) Product Blending System,
(e) Product Liquid Sampling System,
(f) Cylinder Preparation Room,
(g) Dump System,
(h) Decontamination System,
(i) Sub-sampling System,
(j) Separation Building Module Gaseous Effluent Ventilation System,
(k) Centrifuge Test/Centrifuge Post Mortem, and
(l) Ventilated Room System.

The applicant identified a total of 42 chemical accident sequences. The chemical accident sequences covered the range of events that could result in a loss of confinement of UF_6 and the hazardous chemicals produced from UF_6 (i.e., UO_2F_2 and HF). The accident sequences addressed both high- and intermediate-consequence events (AES, 2009b). High- and intermediate-consequence events are defined in 10 CFR 70.61 (b) and 70.61(c), respectively. A high-consequence event is the release of licensed material or hazardous chemicals produced from licensed material that, if an individual is exposed, it could endanger the life of a worker, or lead to irreversible or other serious, long-lasting health effect to a member of the public. Also, a high-consequence event includes an intake of 30 milligrams of soluble uranium by a member of the public. An intermediate-consequence event is the release of licensed material or hazardous chemicals produced from licensed material that, if an individual is exposed, it could lead to irreversible or other serious, long-lasting health effects to a worker, or could cause mild transient health effects to a member of the public. The staff performed a risk-informed review of selected chemical accident sequences and also performed four vertical slice reviews. Chemical accident sequences are discussed in Chapter 3, and Appendix A of this SER.

The staff concludes that the applicant has identified appropriate chemical accident sequences based on the applicant's use of one of the recommended hazard evaluation methods (i.e., Hazard and Operability [HAZOP] Analysis) contained in NUREG-1513 (NRC, 2001) to identify those sequences and the results of the above staff review. In addition, the information provided by the applicant, as described above and in Chapter 3, and Appendix A of this SER, includes a list of the accident sequences with their respective consequence and likelihood that were identified in the ISA Summary (AES, 2009b) that involve hazardous chemicals produced from licensed material and chemical risks of plant conditions that affect the safety of licensed material. Also, the applicant described the postulated high-consequence events, how they will be detected, and the mitigative measures on Section 2.0 of the Emergency Plan (AES, 2009d). The actions described on the Emergency Plan (AES, 2009d) are consistent with the consequences of the accident sequences identified in the ISA Summary (AES, 2009b). Based on the above, the information provided by the applicant meets the guidance in Section 6.4.3.1 bullet (2) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

6.3.3 Chemical Accident Consequences

The chemical exposure limits proposed by the applicant for HF and UF_6 are based on the Acute Exposure Guideline Limit (AEGL) values. Since UO_2F_2 is a soluble uranium compound, the applicant used the values presented in Table 2 of NUREG-1391 (NRC, 1991) to evaluate soluble uranium exposure in terms of both chemical toxicity and radiological dose. The AEGL values for HF and UF_6, and the values for soluble uranium intake are provided in Chapter 3 of this SER. These exposure limits and the categorization of the severity of the chemical accident consequences proposed by the applicant are consistent with NUREG-1520, Table A-5, "Consequence Severity Categories Based on 10 CFR 70.61." The applicant proposed to use the 10-minute AEGL values for exposures to workers whose duration was 10 minutes or less. For the public, the exposure duration was assumed to be 30 minutes. This is consistent with self-protective criteria for UF_6/HF plumes listed in NUREG-1140 (NRC, 1988). The staff finds that the use of AEGL values is consistent with the guidance in Section 6.4.3.1 bullet (6) of NUREG-1520 (NRC, 2002) and is, therefore, acceptable for the determination of compliance with the performance requirements of 10 CFR 70.61.

The applicant used the methods prescribed in NUREG/CR-6410 (NRC, 1998) to determine the source terms. The source term was estimated using a five factor formula contained in section 3.2.5.2 of NUREG/CR-6410 (NRC, 1998):

$$ST = MAR \times DR \times ARF \times RF \times LPF$$

ST	= Source term, UF_6 or HF available to cause consequences, grams
MAR	= Material at risk, the amount of UF_6 that potentially could be impacted by the accident, grams
DR	= Damage ratio, fraction of the MAR actually impacted by the accident, unitless
ARF	= Airborne release fraction, the fraction of material affected by the accident that becomes airborne, unitless
RF	= Respirable fraction, the fraction of material that becomes airborne with a particle size that can be inhaled, unitless
LPF	= Leak path factor, the fraction of the respirable material that leaves any confinement or containment barrier, unitless

The staff's review of the ISA and supporting documentation found the source terms values to be reasonable because the applicant provided the MAR values for each building and followed the methodology described in NUREG/CR-6410 (NRC, 1998). The values for the MAR can be found in Table 2-14 of Appendix E to the ISA Summary (AES, 2009b). Site boundary atmospheric dispersion factors were generated based on Regulatory Guide 1.145 (NRC, 1982). Meteorological data collected at Argonne National Lab-West (EBR) - which is now identified as MFC (Materials and Fuels Complex) - a mesonet station on the Idaho National Laboratory property that is located 18 kilometers (11 miles) west of the EREF site, during 2003-2007, were used. The applicant also used modeling methods for source term determination, release fraction, dispersion factors, and meteorological conditions as prescribed in Regulatory Guide 1.145 (NRC, 1982). The information provided by the applicant meets the guidance of Section 6.4.3.1 bullets (3) and (5) of NUREG-1520 (NRC, 2002) because, as described in Appendix E to the ISA Summary (AES, 2009b), the applicant identified and used appropriate techniques in estimating the concentration of hazardous chemicals produced from licensed material, used the performance requirements criteria of 10 CFR 70.61, and the consequence analysis conform to the guidance in NUREG/CR-6410 (NRC, 1998). Therefore, the staff finds the applicant's proposed methodology for source term determination to be acceptable.

6.3.4 Items Relied On For Safety (IROFS) and Management Measures

6.3.4.1 Chemical Process IROFS

ISA Summary Table 3.7-2 (AES, 2009b) describes each of the accident sequences identified by the applicant and the specific IROFS that are applied to prevent the accident sequence or mitigate its consequences. The ISA Summary Table 3.8-1 (AES, 2009b) describes the safety functions of all identified IROFS and the specific accident sequences that take credit for each IROFS. The staff reviewed the listed IROFS and the process descriptions and process flow diagrams provided in Sections 3.4 and 3.5 of the ISA Summary to identify where each IROFS would be used and how the IROFS would function to prevent the accident sequence or mitigate its consequences. The identified IROFS provide protection to prevent a loss of confinement of licensed material during operation of the facility. Based on this system level review, the staff concludes that the applicant has met the guidance of Section 6.4.3.2 bullet (2) of NUREG-1520 (NRC, 2002) because the ISA Summary (AES, 2009b) identified chemical process IROFS to prevent or mitigate the consequences of accident sequences that involve the chemical hazards of licensed material and hazardous chemicals produced from licensed material. In addition, the ISA Summary (AES, 2009b) identified the hazards being mitigated and the risk category of each accident sequence.

6.3.4.2 Management Measures

The applicant identified management measures to ensure that chemical safety IROFS would be available and reliable to perform their safety function when needed. A Quality Assurance (QA) Level will be applied to IROFS based in the following criteria described in the Quality Assurance Program Description (QAPD) (AES, 2009c):

QA Level 1 Items whose failure or malfunction could directly result in a condition that adversely affects the public, the worker, and the environment as described in 10 CFR 70.61. The failure of a single QA Level 1 item could result in a high or intermediate consequence.

QA Level 2 Items whose failure or malfunction could indirectly result in a condition that adversely affect the public, the worker, and the environment as described in 10 CFR 70.61. The failure of a QA Level 2 item, in conjunction with the failure of an additional item, could result in a high or intermediate consequence.

QA Level 3 Items that are not classified as QA Level 1 or QA Level 2. QA Level 3 items are controlled in accordance with standard commercial practices.

The following sections briefly discuss the management measures applied to chemical safety. Chapter 11 of this SER provides an evaluation of the management measures applied to the proposed EREF.

6.3.4.2.1 Configuration Management

The configuration management proposed for the facility includes those controls that ensure that the facility design basis is thoroughly documented and maintained, and that changes to the design basis are controlled. This includes the following:

- That management commitment and staffing is appropriate to ensure that configuration management is maintained;

- That proper QA is in place for design control, document control, and records management; and

- That all structures, systems, and components, including IROFS, are under appropriate configuration management. (AES, 2009a)

6.3.4.2.2 Maintenance

The applicant proposes to help maintain chemical process safety through the implementation of administrative controls that ensure that process system integrity is maintained and that IROFS and other engineered controls are available and operate reliably (AES, 2009a). These controls include planned and scheduled maintenance of equipment and controls so that design features will function when required (AES, 2009a). Appropriate plant management is responsible for ensuring the operational readiness of IROFS under this control. For this reason, the maintenance function is closely coupled to operation. The maintenance function plans, schedules, tracks, and maintains records for maintenance activities.

Maintenance activities generally fall into the following categories:

1. Surveillance/monitoring,

2. Corrective maintenance,

3. Preventive maintenance, and

4. Functional testing.

6.3.4.2.3 Training

The applicant proposes to provide training to individuals who handle licensed materials and other chemicals at the facility (AES, 2009a). The training program is developed and implemented with input from the chemical safety staff, training staff, and management. The program includes the following:

1. Analysis of jobs and tasks to determine what a worker must know to perform tasks efficiently;

2. Design and development of learning objectives, based on the analysis of jobs and tasks that reflect the knowledge, skills, and abilities needed by the worker;
3. Design and development of qualification requirements for positions where a level of technical capability must be achieved and demonstrated for safe and reliable performance of the job function;

4. Development and implementation of standard and temporary operating procedures;

5. Development and implementation of proper inspection, test, and maintenance programs and procedures;

6. Development of chemical safety awareness throughout the facility so that all individuals know what their roles and responsibilities are in coordinating chemical release mitigation activities—in support of the Emergency Plan—in the event of a severe chemical release; and

7. Coordination of chemical process safety training curriculum with that of other areas including, radiological safety, criticality safety, facility operations, emergency response, and related areas (AES, 2009a).

6.3.4.2.4 Procedures

The applicant proposes to use four types of facility procedures to control activities: (1) operating procedures, (2) administrative procedures, (3) maintenance procedures, and (4) emergency procedures.

Operating procedures, developed for workstation and control room operators, are used to directly control process operations. Operating procedures include:

1. Directions for normal operations, including startup and some testing, operation, and shutdown—as well as off-normal conditions of operation—including alarm response;

2. Required actions to ensure radiological and nuclear criticality safety, chemical safety, fire protection, emergency planning, and environmental protection;

3. Operating limits, controls, and specific direction regarding administrative controls to ensure operational safety; and

4. Safety checkpoints such as hold points for radiological or criticality safety checks, QA verifications, or operator independent verification. (AES, 2009a)

The applicant uses administrative procedures to perform activities that support the process operations, including, but not limited to, management measures such as:

1. Configuration management;

2. Nuclear criticality, radiation, chemical, and fire safety;

3. QA;

4. Design control;
5. Plant personnel training and qualification;

6. Audits and assessments;

7. Incident investigations;

8. Recordkeeping and document control; and

9. Reporting (AES, 2009a).

Administrative procedures are also used for:

1. Implementing the Fundamental Nuclear Material Control Plan (FNMCP),

2. Implementing the Emergency Plan,

3. Implementing the Physical Security Plan, and

4. Implementing the Standard Practice Procedures Plan for the Protection of Classified Matter (AES, 2009a).

Maintenance procedures address:

1. Preventive and corrective maintenance of IROFS;

2. Surveillance (including calibration, inspection, and other surveillance testing);

3. Functional testing of IROFS; and

4. Requirements for pre-maintenance activity involving review of the work to be performed and review of procedures (AES, 2009a).

Emergency procedures address the pre-planned actions of operations and other facility personnel in case of an emergency (AES, 2009a).

6.3.4.2.5 Audits and Assessments

The applicant proposes to conduct audits to determine that facility operations are performed in compliance with regulatory requirements, license conditions, and written procedures (AES, 2009a). The applicant assesses activities related to radiation protection, criticality safety control, hazardous chemical safety, fire protection, and environmental protection.

The applicant will perform audits in accordance with a written plan, which will identify and schedule audits to be performed. Audit team members have no direct responsibility for the function area being audited. Team members will have technical expertise or experience in the function area being audited and will be trained in audit techniques. The applicant will conduct audits annually, on selected functions and areas, as defined above. The chemical process safety functions and areas will be audited at least once every 3 years.

Personnel, qualified by the applicant, who are not directly responsible for productions activities will be used to perform routine surveillance/assessments (AES, 2009a). Deficiencies noted during the inspection, requiring corrective action, will be forwarded to the manager of the applicable area or function for action (AES, 2009a). Future surveillance/assessments will include a review to evaluate if corrective actions have been effective (AES, 2009a).

6.3.4.2.6 Incident Investigation and Corrective Actions

The applicant has a facility-wide incident investigation process that includes chemical process related incidents (AES, 2009a). This process is available for use by any person at the facility for reporting abnormal events and potentially unsafe conditions or activities (AES, 2009a). Events that potentially threaten or lessen the effectiveness of health, safety, or environment protection will be identified, reported to, and investigated by the Environmental Health, Safety & Licensing Manager (AES, 2009a). Each event will be considered in terms of its requirements for reporting in accordance with regulations and will be evaluated to determine the level of investigation required (AES, 2009a). These evaluations and investigations will be conducted in accordance with applicant-approved procedures. The depth of the investigation will depend upon the severity of the classified incident in terms of the levels of uranium/chemical released and/or the degree of potential for exposure to workers, the public, or the environment.

6.3.4.2.7 IROFS and Management Measures Conclusion

The information the applicant provided, as described above and in Appendices A and B of this SER, meets the guidance in Section 6.4.3.2 bullet (2) of NUREG-1520 (NRC, 2002) and is acceptable because the applicant has adequately identified the administrative and engineered controls (IROFS) to prevent chemical accident sequences or mitigate their consequences at the proposed facility. Also, the applicant identified the hazards being mitigated and the risk category.

In addition, the information the applicant provided, as described above and in Chapter 11 of this SER, meets the guidance in Section 6.4.3.2 bullet (3) of NUREG-1520 (NRC, 2002) and is acceptable because the applicant provided sufficient description of its procedures to ensure the reliable operation of engineered controls and that administrative controls will be correctly implemented.

6.3.5 Emergency Management

The proposed facility has an emergency plan and program which include response to mitigate the potential impact of any process chemical release, including requirements for notification and reporting of accidental chemical releases (AES, 2009a). The EREF fire brigade/emergency response team will be outfitted, equipped, and trained for hazardous material response; and local agencies can supplement with additional response teams (AES, 2009a). The City of Idaho Falls, Idaho, Fire Department is the nearest offsite response agency that can supplement the proposed EREF with additional response teams. The applicant states in Section 4.4.2 of the Emergency Plan (AES, 2009d) that offsite response and support groups will be offered annual radiological and chemical response training specific to the proposed EREF.

In Chapter 10 of the Emergency Plan (AES, 2009d) for the proposed EREF, the applicant certifies that the facility has met all the responsibilities under the Emergency Planning and Community Right to Know Act of 1986, Title III, Public Law 99-499, as required by 10 CFR 70.22(i)(3)(xiii). The applicant states that the Material Safety Data Sheets are required for all received hazardous chemicals, controlled, and made available at one or more locations within the facility to ensure that they are readily accessible to all work shifts and for use in an emergency (AES, 2009d).

The information the applicant provided, as described above, meets the guidance in Section 6.4.3.1 bullet (2) of NUREG-1520 (NRC, 2002), and is acceptable because the applicant described the postulated high-consequence events, how they will be detected, and the mitigative measures on Section 2.0 of the Emergency Plan (AES, 2009d). The actions described on the Emergency Plan (AES, 2009d) are consistent with the consequences of the accident sequences identified in the ISA Summary (AES, 2009b).

6.3.6 Baseline Design Criteria

The applicant provides design bases information for chemical process safety IROFS for the proposed facility in the SAR (AES, 2009a) and the ISA Summary (AES, 2009b). For chemical protection, 10 CFR 70.64(a)(5) states:

> "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 only chemical of concern (EREF Class 1) is UF_6. Details of design and safety features of all chemical process systems are found in Chapter 3 of the ISA Summary (AES, 2009b). The applicant's design of the chemical process systems includes numerous controls, in addition to the IROFS, for maintaining safe conditions during operation. The applicant accomplishes this through several means, including:

- Managing the arrangement and size of material containers and processes;
- Selection and use of materials compatible with process chemicals;
- Providing inherently safer operating conditions (e.g., vacuum handling); and
- Providing process interlocks, controls, and alarms within the process.

The staff reviewed the applicant's proposed design of the gas centrifuge uranium enrichment facility contained in the ISA Summary Sections 3.3 through 3.5, and the process hazards description in Section 3.6 (AES, 2009b). The staff notes that the uranium enrichment process is basically a physical process that separates the ^{235}U isotope from the ^{238}U isotope based on their difference in mass. The uranium is in the chemical form of UF_6. The entire process, with the exception of the product liquid sampling step, will be conducted under a significant vacuum. Furthermore, the process design involves very limited inventories of UF_6 throughout the entire process. As a result, any process leak would result in air in-leakage into the system. Any uranium that could escape through a system breach would be limited by the available inventory and molecular diffusion. The chemical behavior of UF_6 and its reaction products (upon contact with moisture in the air, discussed in Section 6.3.1.3 of this SER) are such that most uranium-bearing material is likely to accumulate near any process breach. Because maintenance of the vacuum is necessary to operate the centrifuge machines, any significant leakage would be quickly detected, as operation with even relatively small amounts of air would result in damage to the centrifuges. Based on the need to operate at, and maintain, a significant vacuum throughout the gaseous portion of the process, and the limited inventories of licensed material contained in any portion of the gaseous process, the staff concludes that the design bases provide for adequate protection against chemical risks.

The staff notes that the applicant's design of product liquid sampling system uses an ANSI N14.1 qualified cylinder as the primary confinement vessel and an American Society of Mechanical Engineers Code pressure vessel as a secondary confinement system (AES, 2009b). The staff concludes that this design approach for the liquid portion of the process is acceptable because it uses recognized nuclear fuel cycle industry codes and standards.

The staff reviewed the results of the applicant's HAZOP analysis as discussed in Chapter 3 of this SER. This method is widely used in the chemical industry during the design phase to identify operability and safety issues, and is identified as an acceptable method in Section 2.4 of NUREG-1513 (NRC, 2001). As applied to the gas centrifuge uranium enrichment process, the applicant's HAZOP analysis considered a variety of internal process, facility, and external hazards that could breach the process and release licensed material and hazardous chemicals produced from licensed material. The results of the applicant's ISA are presented in Table 3.7-1 of the ISA Summary (AES, 2009b). The table contains information concerning the accident sequences identified as a result of the applicant's HAZOP analysis, the unmitigated risk of each applicant-identified accident sequence, and the IROFS applied to prevent the accident sequence or mitigate its consequences. The staff also reviewed selected high-consequence and intermediate-consequence accident scenarios to confirm that chemical events that could exceed the performance requirements of 10 CFR 70.61 were addressed.

The staff concludes that the information the applicant provided and the applicant's proposed design meets the guidance in Section 6.4.3.3 of NUREG-1520 (NRC, 2002) because the applicant addressed the list of items in Section 2.4 of NUREG-1601 (NRC, 1997). These items should be considered in an adequate facility design. Also, the applicant described in the ISA Summary how the ISA was performed and how it satisfies the performance requirements of 10 CFR 70.61. Based on the information in this section and the process descriptions of the ISA Summary (AES, 2009b), the applicant applied defense-in-depth to the entire process (i.e., several alarms before reaching IROFS set points, vacuum operation for most parts of the process and two independent confinement barriers for sampling operations). The applicant's information also provides for adequate protection against chemical risks produced from licensed materials, facility conditions which affect the safety of licensed material, and hazardous chemicals produced from licensed material; and meets the requirements of 10 CFR 70.64(a)(5).

6.4 Evaluation Findings

The staff evaluated the application using the criteria previously listed. Based on the review of the SAR (AES, 2009a), the NRC staff has concluded that the applicant has described and assessed accident sequences that can result from the handling, storage, or processing of licensed materials and that can potentially have significant chemical consequences and effects. The applicant has prepared a hazard analysis that identifies and evaluated those chemical process hazards and potential accidents and established safety controls providing reasonable assurance of safe facility operation. To ensure that the performance requirements in 10 CFR Part 70 are met, the applicant has stated that controls are maintained, available, and reliable to perform their safety-related functions when needed. The staff has reviewed and independently verified these safety controls and the applicant's plan to managing chemical process safety, and finds them acceptable.

The staff concludes that the applicant's plan for managing chemical process safety and chemical process safety controls meets the requirements for 10 CFR Part 70, and provides reasonable assurance that the public health and safety and the environment will be protected.

6.5 References

(AES, 2009a) AREVA Enrichment Services, "Eagle Rock Enrichment Facility Safety Analysis Report," Revision 1, April 2009.

(AES, 2009b) AREVA Enrichment Services, "Eagle Rock Enrichment Facility Integrated Safety Analysis Summary," Revision 1, April 2009.

(AES, 2009c) AREVA Enrichment Services, "Quality Assurance Program Description for Design, Construction, Operation and Decommissioning of the Eagle Rock Enrichment Facility," Revision 1, April 2009.

(AES, 2009d) AREVA Enrichment Services, "Eagle Rock Enrichment Facility Emergency Plan," Revision 1, April 2009.

(ANSI, 1995) American National Standards Institute, ANSI N14.1, "Uranium Hexafluoride – Packaging for Transportation," 1995.

(NRC, 1982) U.S. Nuclear Regulatory Commission, Regulatory Guide 1.145, Revision 1, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," November 1982.

(NRC, 1988) U.S. Nuclear Regulatory Commission, NUREG-1140, "A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees," January 1988.

(NRC, 1991) U.S. Nuclear Regulatory Commission, NUREG-1391, "Chemical Toxicity of Uranium Hexafluoride Compared to Acute Effects of Radiation," February 1991.

(NRC, 1997) U.S. Nuclear Regulatory Commission, NUREG-1601, "Chemical Process Safety at Fuel Cycle Facilities," August 1997.

(NRC, 1998) U.S. Nuclear Regulatory Commission, NUREG/CR-6410, "Nuclear Fuel Cycle Accident Analysis Handbook," March 1998.

(NRC, 2001) U.S. Nuclear Regulatory Commission, NUREG-1513, "Integrated Safety Analysis Guidance Document," May 2001.

(NRC, 2002) U.S. Nuclear Regulatory Commission, NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," March 2002.

(NRC, 2009) U.S. Nuclear Regulatory Commission, "Memorandum from M. Breeda Reilly to Thomas G. Hiltz Regarding the July 13-15, 2009, AREVA Enrichment Services Eagle Rock License Application-Integrated Safety Analysis In-Office Review," August 2009.

CHAPTER 7.0 FIRE SAFETY

The purpose of this review is to determine, whether the applicant has designed a facility that provides adequate protection against fires and explosions that could affect the safety of licensed materials and thus, present an increased radiological risk. The review should also establish that the applicant has considered the radiological consequences of fires and will institute suitable safety controls to protect workers, the public, and the environment.

7.1 Regulatory Requirements

The regulatory basis for the fire safety review should be the general and additional contents of application, as required by Title 10 of the *Code of Federal Regulations* (10 CFR) 30.33, 10 CFR 40.32, 10 CFR 70.22, and 10 CFR 70.65. In addition, the fire safety review should focus on providing reasonable assurance of compliance with 10 CFR 70.61, 70.62, and 70.64.

7.2 Regulatory Acceptance Criteria

The acceptance criteria the U.S. Nuclear Regulatory Commission (NRC) uses for reviews of fire safety are outlined in Sections 7.4.3.1 through 7.4.3.5 of NUREG-1520 (NRC, 2002). The following section, "Staff Review and Analysis," provides details on the acceptance criteria and describes how the applicant satisfies them.

7.3 Staff Review and Analysis

This chapter addresses the staff's review of facility fire protection, including fire safety management measures, fire hazards analysis, facility fire protection, process fire safety, and fire safety and emergency response, as presented in the Safety Analysis Report (SAR) (AES, 2009a) and supporting documents (AES, 2009b and AES, 2010).

The Eagle Rock Enrichment Facility (EREF) and its Fire Safety program were reviewed to determine applicability and level of compliance with the National Fire Protection Association Standard (NFPA) 801 (NFPA, 2008c) and applicable standards referenced within. Section 7.6 of the SAR (AES, 2009a) lists the fire codes and standards considered by the applicant to be applicable to the facility. The staff finds the use of these consensus codes and standards to be in accordance with the guidance of Section 7.4.3 of NUREG-1520 (NRC, 2002) in regard to nationally recognized codes and standards that may be used to measure reasonable assurance of fire safety. Therefore, the staff considers the use of the above codes and standards to satisfy the requirements of 10 CFR 70.64(a) Baseline Design Criterion (3), "Fire Protection."

7.3.1 Fire Safety Management Measures

The applicant will implement fire safety management measures as described in Chapter 11 of the SAR (AES, 2009a). Management measures applicable to fire safety include: configuration management; maintenance; training; procedures; audits and assessments; incident reporting and investigations; and records management. These measures will ensure that fire protection IROFS are available and reliable. The applicant will follow the codes and standards, as listed in

Section 7.6 of the SAR (AES, 2009a), which are applicable to the individual fire safety management measures. Management measures are evaluated in Chapter 11 of this Safety Evaluation Report (SER).

7.3.1.1 Management Policy and Direction

The Environmental Health, Safety and Licensing Manager is responsible for fire protection and is assisted by the Safety, Security, and Emergency Preparedness Manager, who is responsible for day-to-day safe operation of the facility, including fire safety. The Safety, Security, and Emergency Preparedness Manager is assisted by fire safety personnel who are trained in fire protection and have nuclear fire safety experience. The fire protection staff is responsible for:

- Fire protection program and procedural requirements;
- Fire prevention activities (i.e., administrative controls and training);
- Maintenance, surveillance, and quality of the facility fire protection features;
- Control of design changes, as related to fire protection;
- Documentation and recordkeeping, as related to fire protection;
- Organization and training of the fire brigade; and
- Pre-fire planning.

Fire prevention at the facility consists of administrative controls to: (a) govern the handling of transient combustibles, (b) control ignition sources, (c) ensure that open flames or combustion generated smoke is not used for leak-testing, (d) conduct periodic fire prevention inspections, (e) perform periodic housekeeping inspections, and (f) implement a system to control the disarming of the various types of fire detection or fire suppression systems. Further information concerning the fire detection and suppression systems can be found in Section 7.3.3 of this chapter. The inspection, testing, and maintenance of fire protection systems will comply with nationally recognized industry standards. The applicant's ISA Summary has adequately addressed fire safety management measures, in accordance with the guidance established in NUREG-1520 (NRC, 2002). In addition, the applicant's fire safety management measures meet the requirements of 10 CFR 30.33, 10 CFR 40.32, 10 CFR 70.22, 10 CFR 70.64, and 10 CFR 70.65 as they pertain to the fire protection aspects of the facility.

7.3.1.2 Fire Safety Management Measures Conclusions

Based on its review, the staff reached the following conclusions:

Consistent with the acceptance criteria from Section 7.4.3.1 of NUREG-1520 (NRC, 2002), the applicant's fire safety management measures identify a senior level manager who has the authority and staff to ensure that fire safety receives appropriate priority, and are, therefore, acceptable.

Consistent with the acceptance criteria from Section 7.4.3.1 of NUREG-1520 (NRC, 2002), the applicant's fire safety management measures identify a facility safety committee staffed by managers of different disciplines to integrate facility modifications, and are, therefore, acceptable.

Consistent with the acceptance criteria from Section 7.4.3.1 of NUREG-1520 (NRC, 2002), the applicant's fire safety management measures include fire prevention; inspection, testing, and maintenance of fire protection systems; Emergency Response Organization (ERO) qualifications, drills, and training; and pre-fire plans as recommended by NFPA 801 (NFPA, 2003e) and are, therefore, acceptable.

Consistent with the acceptance criteria from Section 7.4.3.1 of NUREG-1520 (NRC, 2002), the applicant's fire safety management measures are documented in sufficient detail to identify their relationship to, and functions for normal operations; anticipated (off-normal) events; and accident safety (i.e., IROFS), and are, therefore, acceptable.

Consistent with the acceptance criteria from Section 7.4.3.1 of NUREG-1520 (NRC, 2002), the applicant's fire safety management measures will ensure that the IROFS, as identified in the ISA Summary, are available and reliable; and will ensure that the facility maintains fire safety awareness among employees, controls transient ignition sources and combustibles, and maintains a readiness to extinguish or limit the consequences of fire. Therefore, these measures are acceptable.

Based on the above conclusions, the staff has reasonable assurance that the applicant's fire safety management measures would meet the requirements of 10 CFR 30.33, 40.32, 70.22, 70.61, 70.64, and 70.65 as they pertain to the fire protection aspects of the facility.

7.3.2 Fire Hazards Analysis (FHA)

The applicant's ISA Summary describes, qualitatively, the potential credible fire accident scenarios and associated risks for the facility. The applicant postulated and evaluated the following key fire accident scenarios:

- Fire in the Cylinder Receipt and Shipping Building (CRSB);
- Fire involving Cylinder Transporters/Movers;
- Fire inside the Cascade Halls;
- Fire inside the uranium hexafluoride (UF_6) handling area/Blending, Sampling, and Preparation Building;
- Fire inside the Technical Support Building (TSB); and
- Fire affecting cylinder storage pads.

Exterior and interior building explosions as initiating events to accident sequences were evaluated and found to be highly unlikely, without the need for IROFS. These initiating events had a frequency of less than 10^{-6}/year which satisfies the performance criteria requirements in 10 CFR 70.61, therefore no IROFS are required.

7.3.2.1 IROFS Related to Fire Safety

The applicant has identified a set of IROFS that would ensure that the likelihood of a fire causing high-consequence events is highly unlikely, and the likelihood of a fire causing intermediate-consequence events is unlikely. These IROFS are listed in Table 7.3-1.

The NRC staff considers the failure probability indices assigned to these IROFS to be achievable, with their respective bases as described in Section 3.8.3 of the ISA Summary. Section 3.8.3 of the ISA Summary provides proposed surveillance frequencies, safety margins, and other measures that will support the low-failure probabilities assigned to these measures, which are in accordance with the NRC guidance provided in Table A-9 of NUREG-1520. All of the listed IROFS will also be supported by the general management measures, as described in Section 3.1.8.3 of the ISA Summary. Further analysis of the ISA methodology can be found in Chapter 3 of this SER. In conclusion, the staff finds the selection of accident sequences and the determination of the IROFS related to fire protection to be acceptable to satisfy the performance requirements of 10 CFR 70.61.

The remaining features of fire protection that are described in the license application are fire protection measures that provide overall defense-in-depth protection of fire safety for operations. All fire protection measures are further discussed and evaluated in the following subsections discussing each fire related accident scenario. The applicant's ISA Summary has adequately addressed fire risks in accordance with the regulation and the guidance established in NUREG-1520 (NRC, 2002).

Table 7.3-1 Identification of IROFS

IROFS	Failure probability index	Description of IROFS
IROFS42	-3	Automatic closure of fire rated barrier opening protections (e.g., doors, dampers, penetration seals) to ensure that the integrity of area fire barriers prevents fires from propagating into areas containing uranic material.
IROFS43	-3	Administratively limit transient combustible loading in areas containing uranic material to ensure integrity of uranic material components/containers, and limit the quantity of uranic material at risk, to ensure that consequences to the public are low.
IROFS44	-3	Administratively limit fire exposure to 30B product cylinders containing > 0.1 kg of UF_6 from a semi-tractor trailer fire during the shipping process. This will be controlled by requiring that all outgoing 30B product cylinders are loaded into their required U.S. Department of Transportation (DOT) overpacks prior to placement on the semi-tractor trailer.
IROFS45	-3	Administratively limit storage of UF_6 cylinders in the CRSB, to ensure $\geq$ 1-m (3-ft) setback from the edge of the loading dock.
IROFS46	-2	Automatic hardwired, fail-safe trip of the Chemical Trap Workshop heating, ventilation, and air conditioning and isolation from Technical Support Building (TSB) gaseous effluent ventilation system on smoke detection with low exfiltration from the room to ensure low consequences to the public.
IROFS47	-3	Administratively limit vehicle approach to bare 30B UF_6 cylinders containing greater than heel quantity—11.3 kg (25 lb) for all exterior storage and handling areas.
IROFS48	-3	Administrative control of diesel fuel delivery vehicle route and staging.
IROFS49	-3	Administratively limit exposure by requiring worker action to evacuate the area(s) of concern to ensure worker consequences of inhalation of uranic material and Hydrogen Fluoride (HF) are low.
IROFS67	-3	Passive engineered control of intrasite cylinder movement vehicles to ensure they do not result in rupture of 30B or 48Y cylinders.
IROFS73	-3	Administrative controls to ensure that only authorized vehicles operate in areas proximate to exterior locations where bare 30B product cylinders (> heel quantity) are stored or handled.
IROFS100	-2	Automatic suppression of a fire to ensure consequences to the public is low. Included within the boundary of IROFS is an administrative control, independent from IRFOS43, to maintain combustibles at acceptable levels in accordance with NFPA 13 (NFPA, 2007b).

7.3.2.2 *Fire in the CRSB*

All UF_6 feed cylinders and empty product and byproduct cylinders will enter the facility through the CRSB. The applicant considered three separate fire scenarios in the CRSB. These fire scenarios were a fire at the CRSB loading dock, a fire in the CRSB general areas, and a fire involving a cylinder delivery vehicle.

At the CRSB loading dock, UF_6 is contained in 48Y and 30B cylinders on the loading dock and scales adjacent to the dock. The most severe fire was postulated to be a vehicle fire at the loading dock. The applicant evaluated the effect of this fire by calculation and showed that there could be a potential fire threat to UF_6 cylinders, but that this threat could be eliminated by assuring that cylinders were stored with a 1 m (3.3 ft) setback from the edge of the loading dock. The 1 m (3.3 ft) setback is an administrative control (IROFS45). Based on the above, the applicant has determined that the likelihood of a fire being initiated, and of the IROFS failing, so that a release exceeding the consequence threshold of 10 CFR 70.61(b) or (c) occurs is highly unlikely. The staff reviewed the applicant's calculation and agrees with this assessment, given the calculation was performed utilizing the guidance provided in NUREG-1805 (NRC, 2004).

The CRSB General Areas contain UF_6 in 48Y and 30B cylinders. Combustible loading is expected to be very low, and transient combustibles will be controlled with IROFS43. No liquid combustibles are listed for the CRSB general areas. Hence, a fire with the intensity and duration to heat a large UF_6 cylinder to its critical temperature is not considered credible. Fire propagation will be prevented by rated barriers (IROFS42) between adjacent fire areas and the CRSB general areas. Based on the above, the applicant has determined that the likelihood of a fire being initiated, and of the IROFS failing, so that a release exceeding the consequence threshold of 10 CFR 70.61(b) or (c) occurs is highly unlikely. The staff agrees with this assessment.

The applicant also evaluated a fire involving UF_6 cylinders present on a delivery vehicle. The Department of Transportation's (DOT's) regulations require thermal protection (e.g., overpack or other protective assembly) which will withstand the thermal test criteria, specified in 10 CFR 71.73(c)(4) without rupture of the containment system, for all offsite UF_6 shipments. Hence, both incoming cylinders and outgoing cylinders will be protected by approved thermal protection. The handling practice for incoming cylinders containing UF_6 will be to offload the integral cylinder in its protective assembly to the loading dock before opening or removal of the protective assembly. Outgoing cylinders will be individually loaded into a protective assembly before placement on truck trailers. The applicant determined that the worst postulated truck fire involving diesel fuel and other combustibles associated with a truck fire would burn for no more than 30 minutes. Approved protective assemblies (IROFS44) are designed to protect a cylinder for 30 minutes in an 800°C (1472°F) hydrocarbon fire. Because of the location of the cylinder in the assembly on the truck trailer and the duration of the fire, the UF_6 cylinders will be adequately protected from rupture from a truck fire. The staff also evaluated the applicant's calculations and agrees with the results, given the calculations were performed utilizing guidance previously approved by the NRC (NRC, 2005). Based on the above, the applicant has determined that the likelihood of a fire being initiated, and of the IROFS failing, so that a release exceeding the consequence threshold of 10 CFR 70.61(b) or (c) occurs is highly unlikely. The staff agrees with this assessment.

The staff concludes 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 fire in the CRSB.

7.3.2.3 Fire Involving Cylinder Transporters/Movers

The assumed inventory for a fire involving a cylinder transporter/mover is the amount of UF_6 in a UF_6 cylinder (48Y or 30B) in transit. Only electric drive cylinder transporters with preventative measures engineered to prevent a rupture of a cylinder will be used for cylinder transport at the proposed facility (IROFS67). When filled 30B cylinders are transported outside of the buildings, they are protected by DOT approved overpacks (IROFS44).

Based on the above, the applicant has determined that the likelihood of a fire being initiated, and of the IROFS failing, so that a release exceeding the consequence threshold of 10 CFR 70.61(b) or (c) occurs is highly unlikely.

The staff concludes 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 fire on a cylinder transporter either inside or outside of the proposed facility's buildings.

7.3.2.4 Fire in the Separations Building

The eight cascade halls containing the centrifuges are located in the Separations Building. The fire scenario inside the cascade halls is assumed to take place inside a module that holds twelve cascades, each cascade containing hundreds of centrifuges. If the entire module was engulfed in a fire and the total inventory released, a high-consequence event would result. However, a fire is prevented from propagating into the module by fire barriers (IROFS42) and by automatic fire suppression (IROFS100) in the Process Services Corridor of the Separations Building. A fire originating inside the module is presumed to involve the cables feeding the centrifuge drive motors. If transient combustibles are controlled (IROFS43), this fire, at worst, is expected to cause failures in the aluminum piping manifold releasing 100% of the inventory feeding one assay, resulting in consequences below the threshold of the 10 CFR 70.61(c) intermediate-consequence limit. This analysis was based upon the guidance provided in NUREG-1805 (NRC, 2004). Based on the above, the applicant has determined that the likelihood of a fire being initiated and the IROFS failing, so that a release exceeding the consequence threshold of 10 CFR 70.61(b) or (c) occurs is highly unlikely.

The staff concludes 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 fire in the Separations Building.

7.3.2.5 Fire in the UF_6 Handling Area or Blending, Sampling, and Preparation Building

The UF_6 Handling Area contains the UF_6 Feed System, Product Take-Off System, and the Tails Take-Off System. The Blending, Sampling, and Preparation Building contains the Product Liquid Sampling System and the Product Blending System. The UF_6 inventory in the blending and liquid sampling area of the UF_6 Handling Area is contained in cylinders, piping, manifolds, and hoses. The applicant states that additional uranic material may be present on the carbon/alumina traps that capture residual traces of UF_6 from the various feed, product, and tails system cold traps. A fire is prevented from propagating into the UF_6 Handling Area, and

Blending and Liquid Sampling Area, by fire barriers (IROFS42) and the automatic fire suppression system (IROFS100). A fire originating in these areas with improperly placed combustibles is considered capable of failing only a single-cylinder hose.

A fire, involving expected in-situ and transient combustibles, could cause failure in the aluminum piping manifold and release 50 percent of the inventory feeding one module. This would result in consequences that are below the 10 CFR 70.61(c) intermediate-consequence limit. Severe fires will be prevented by IROFS43, which will control the location and the amount of transient combustibles in the area, and by the automatic fire suppression system (IROFS100). Based on the above, the applicant has determined that the likelihood of a fire being initiated, and of the IROFS failing, so that a release exceeding the consequence threshold of 10 CFR 70.61(b) or (c) occurs, is highly unlikely.

The staff concludes 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 fire in the UF_6 Handling Area or the Blending, Sampling, and Preparation Building.

7.3.2.6 Fire in the TSB

The TSB contains support areas for the facility, such as the Solid Waste Collection Room, Chemical Trap Workshop, Sample Bottle Storage Room, various laboratories, and monitoring centers. In the TSB, fires were postulated in all uranic material areas; and IROFS were found to be needed in the Solid Waste Collection Room, Chemical Trap Workshop, and Sample Bottle Storage Room.

The uranium inventory in the Solid Waste Collection Room is contained in 12-L (3.2-gallon [gal]) metal containers and 210-L (55-gal) metal drums. A fire is prevented from propagating into the Solid Waste Collection Room by rated fire barriers (IROFS42) and automatic fire suppression (IROFS100). For a fire originating in the area, and involving expected in-situ and transient combustibles, the applicant postulates that only a few kilograms of uranic materials would be present in open containers or drums during transfer/packing operations, and driven off in case of a fire. Preventive measures are to administratively limit transient combustible loading in areas containing uranic material to ensure integrity of uranic components/containers and to limit the quantity of uranic material at risk to ensure that consequences to the public are low (IROFS43). Additionally, automatic fire suppression will be provided (IROFS100). Based on the above, the applicant has determined that the likelihood of a fire being initiated, and of the IROFS failing, so that a release exceeding the consequence threshold of 10 CFR 70.61(b) or (c) occurs is highly unlikely.

The uranium inventory in the Chemical Trap Workshop is contained in 12-L (3.2- gal) metal containers and 210-L (55-gal.) metal drums. A fire is prevented from propagating into the room by fire rated barriers (IROFS 42) and automatic fire suppression (IROFS100). For a fire originating in the area, and involving expected in-situ and transient combustibles, the applicant postulates that several kilograms of uranic materials would be present in open containers or drums during transfer/bulking operations, and driven off in case of a fire. Preventive measures are to administratively limit transient combustible loading in areas containing uranic material to ensure integrity of uranic components/containers and to limit the quantity of uranic material at risk to ensure that consequences to the public are low (IROFS43). Additionally, to mitigate the severity to low consequence, automatic fire suppression will be provided (IROFS100) and smoke detection interlocked to isolate the room's ventilation systems will also be provided (IROFS46). Based on the above, the applicant has determined that the likelihood of a fire being initiated, and of the IROFS failing, so that a release exceeding the consequence threshold of 10 CFR 70.61(b) or (c) occurs is highly unlikely.

The TSB Sample Bottle Storage Room also contains a uranium inventory that could potentially result in a high consequence. A fire is prevented from propagating into the Chemical Laboratory Sample Storage room by rated fire barriers (IROFS42) and automatic fire suppression (IROFS100). For a fire originating in the Sample Storage Room with controls in place (IROFS43), with expected insitu and transient combustibles, the consequences would be low. The fire would not have sufficient combustibles to fail a sample cylinder. Additionally, automatic fire suppression will be provided (IROFS100) consistent with criticality safety requirements. Based on the above, the applicant has determined that the likelihood of a fire being initiated, and of the IROFS failing, so that a release exceeding the consequence threshold of 10 CFR 70.61(b) or (c) occurs is highly unlikely.

The staff concludes 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 fire in the TSB.

7.3.2.7 Cylinder Storage Pads

The Cylinder Storage Pads provide storage for full product cylinders, full feed cylinders, tails cylinders, and 6 months of empty product and feed cylinders. The Storage Pads occupy approximately 17 hectares (42 acres) and are sized to accommodate enough cylinders for 30 years of operation.

For the full product cylinder storage pad, fires were postulated on the transporter and from other vehicles near the pad. Only electric drive cylinder transporters with preventative measures engineered to prevent a rupture of a cylinder will be used for cylinder transport at the proposed facility (IROFS67). In order to prevent exposure from a potential vehicle fire of fuel spill fire, preventative measures are to place cylinders on the pad behind a vehicle impact barrier (IROFS47). The barrier also provides separation distance to prevent fuel fires from migrating onto the cylinder storage pad and to provide radiant heat reduction from cylinders from vehicle and fuel spill fires. The combination of barriers and separation distance values was shown to prevent cylinder failure from fire exposure. The cylinder target wall temperature was less than 600°C (1112°F). Additionally, vehicle operation is administratively controlled (IROFS73) to allow only authorized vehicles and operators within exterior areas of bare 30B cylinders. Based on the above, the applicant has determined that the likelihood of a fire being initiated, and of the IROFS failing, so that a release exceeding the consequence threshold of 10 CFR 70.61(b) or (c) occurs is highly unlikely.

The applicant performed its own calculations, in accordance with the guidance provided in NUREG-1805 (NRC, 2004), to determine the possible effects of a diesel fuel delivery vehicle on the storage pads and staging areas. To prevent exposure to a potential diesel fuel delivery vehicle fire or fuel spill fire from this vehicle near the Full Tails Cylinder Storage Pad, the Full Feed Cylinder Storage Pad, the Empty Cylinder Storage Pad, the Truck Entry Staging Area, and the Cylinder Transporter Path, preventive measures are to designate a specific route for fuel deliveries that is remote from these areas and to require all fuel delivery vehicles to be accompanied onsite to ensure delivery only via the designated route (IROFS48). Based on the above, the applicant has determined that the likelihood of a fire being initiated, and of the IROFS failing, so that a release exceeding the consequence threshold of 10 CFR 70.61(b) or (c) occurs is highly unlikely.

The staff concludes 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 fire near the Cylinder Storage Pads.

7.3.2.8 FHA Conclusions

The NRC staff concludes that the applicant has identified and evaluated all fire related accident scenarios credible for the proposed centrifuge process. The applicant has reasonably identified and evaluated possible fire initiators and consequences and has identified IROFS for preventing or mitigating fire accident scenarios that could result in intermediate or large consequences leading to unacceptable performance, in accordance with 10 CFR Part 70, and as described in guidelines established in NUREG-1520 (NRC, 2002).

Consistent with the acceptance criteria from Section 7.4.3.2 of NUREG-1520 (NRC, 2002), an FHA was performed for each process area and each FHA included a description by fire area of the fuel loading, fire scenarios, methods of consequence analysis, the potential consequences, and a description of mitigative controls, and is, therefore, acceptable. In addition, based on its review of the above information, the staff concludes that the applicant's performance of their FHA meets the requirements of 10 CFR 30.33, 40.32, 70.22, 70.61, 70.64, and 70.65, as they pertain to the fire protection aspects of the facility.

7.3.3 Facility Design

7.3.3.1 Facility Passive-Engineered Fire Protection Systems

Buildings containing UF_6 are the TSB; CRSB; Cascade Halls; and the Blending, Sampling, and Preparation Building, which have protected structural steel columns and trusses with built-up composite roofing on metal deck. This construction is classified as Type II-222, in accordance with NFPA 220 (NFPA, 2006). This means that the exterior and bearing walls, interior-bearing walls supporting more than one floor, columns supporting more than one floor, and structural members supporting more than one floor, have a fire resistance of at least 2 hours. Floors have a resistance of at least 2 hours and roofs have a fire resistance of at least 1 hour.

The Centrifuge Assembly Building (including the Centrifuge Test and Centrifuge Post Mortem Area), which also contains UF_6, will have insulated metal panel exterior walls with built-up composite roofing on metal deck. This construction is classified as Type II-000 in accordance with NFPA 220 (NFPA, 2006). This means that all structural members are of non-combustible or limited combustible construction, but no fire rating is required.

The remaining utility and non-process-related areas are classified as Type II-000, in accordance with NFPA 220 (NFPA, 2006). This means that all structural members are of non-combustible or limited combustible construction, but no fire rating is required. The applicant's ISA Summary has adequately addressed passive-engineered fire protection systems in accordance with the guidance established in NUREG-1520 (NRC, 2002). In addition, the applicant's passive engineered fire protection systems meet the requirements of 10 CFR 30.33, 10 CFR 40.32, 10 CFR 70.22, 10 CFR 70.64, and 10 CFR 70.65 as they pertain to the fire protection aspects of the facility.

7.3.3.2 Facility Active-Engineered Fire Protection Systems

Electrical Installation and Fire Alarm System

All electrical systems at the facility are installed in accordance with NFPA 70 (NFPA, 2008b). Switchgear, motor control centers, panel boards, variable frequency drives, uninterruptible power supply systems, and control panels are mounted in metallic enclosures and contain limited amounts of combustible material. Cable trays and conduits are metallic and the cable in cable trays are flame retardant and tested in accordance with industry guidance.

Each building of the facility is equipped with a listed modular, multi-zone fire alarm control panel installed in accordance with NFPA 72 (NFPA, 2007d). Each panel has a dual power supply consisting of normal building power and backup power, by either a 24-hour battery or the facility uninterruptible power supply (UPS). The sprinkler system and hose-station water-flow detection are connected to separate control-panel zone modules. Fire detector and manual-pull station-alarm circuits are also on separate modules. Each zone module has separate alarm and trouble contacts for connection to the central alarm panel in the Control Room. Activation of a fire detector, manual-pull station, or water-flow detector results in an audible and visual alarm at the building control panel and the central alarm panel.

The central alarm panel—located in the control room—is a listed, microprocessor-based, addressable console. The central alarm panel has dual power supplies consisting of normal building power and backup power by either a 24-hour battery or the facility UPS. The central alarm panel monitors all functions associated with the individual alarm panels and fire pump controllers. All alarm and trouble functions are audibly and visually annunciated by the central alarm panel and automatically recorded via printout. Failure of the central alarm panel will not result in failure of any building-alarm control-panel functions.

The following conditions are monitored by the central alarm console through the fire pump controllers:

- Pump running,
- Pump failure to start,
- Pump controller in off or manual position,
- Battery failure,
- Diesel overspeed,
- Diesel high engine-jacket coolant temperature,
- Diesel low oil pressure, and
- Battery charger failure.

Both diesel and electric fire pumps are maintained in the automatic-start position at all times, except during periods of maintenance and testing. Remote manual-start switches are provided in the Control Room adjacent to the alarm console. Fire pumps can only be shut off at the controllers.

Portable Fire Extinguishers

Portable fire extinguishers are installed throughout all buildings in accordance with NFPA 10 (NFPA, 2007a). Multipurpose fire extinguishers are provided generally for Class A (ordinary combustibles), Class B (flammable and combustible liquids), and Class C (electrical equipment) fires. Specialized extinguishers are located in areas requiring protection of particular hazards.

Wheeled extinguishers are provided for use in water-exclusion areas. In areas with moderator control issues, the extinguishers are filled with carbon dioxide or dry chemical.

Fire Water Supply

The facility fire water supply consists of two 757,082-L (200,000-gal) water storage tanks designed and constructed in accordance with NFPA 22 (NFPA, 2008a). Within each tank, 681,374-liter (L) (180,000-gal) is reserved for fire protection. Fill and makeup to the tanks are from the well water supply to the site, and the water supply is capable of filling fire protection water inventory in a single tank within an 8-hour period. The fire pumps consist of one electric-driven pump and one diesel-driven pump, both rated for 5678-liters per minute (lpm) (1500-gallons per minute [gpm]), at 10.35-bar (150-pounds-force per square inch absolute [psia]), pumps. Both pumps are horizontal centrifugal pumps designed and installed in accordance with NFPA 20 (NFPA, 2007c). The maximum anticipated fire demand is 5678 lpm (1500 gpm), based on 3785 Lpm (1000 gpm) from a building-sprinkler system, plus 1892 lpm (500 gpm) for hose streams for a duration of 2 hours. The combination of two water tanks and two fire pumps provides 100 percent redundancy for fire protection. The tanks are arranged such that one will be available for suction at all times. In addition to fixed standpipes and fire hose stations, the facility will be provided with fire hose on mobile apparatus or at strategic locations throughout the facility. The amount of hose provided will be sufficient to ensure that all points within the facility will be able to be reached by at least two 38-mm (1½-inch [in]) diameter hoses and one 64-millimeters (mm) (2 ½-in)-diameter backup hose, consistent with NFPA 1410 (NFPA, 2005b). These lines will have a minimum nozzle pressure of 4.5 bar (65 pounds-force per square inch gauge [psig]) for the 38-mm (1½-in) hose, and 6.9 bar (100 psig) for the 64-mm (2 ½-in) hose.

Engineered Automatic Fire Suppression Systems

Automatic pre-action sprinkler systems are provided in the following buildings, subject to moderator control restrictions:

- Process Service Corridor in the Separations Building Module;
- UF_6 Handling Area,
- Technical Support Building, and
- Blending, Sampling, and Preparation Building.

Automatic wet pipe sprinkler systems are provided in the following buildings:

- Administration Building,
- Security and Secure Administration Building,
- Long-Term Warehouse,
- Fire Pump House,
- Centrifuge Assembly Building,
- Operations Support Building, and
- Short-Term Warehouse.

These systems are designed and tested in accordance with NFPA 13 (NFPA, 2007b). Sprinkler-system control valves are monitored under a periodic inspection program, and their proper positioning is supervised in accordance with NFPA 801 (NFPA, 2008c).

The NRC staff concludes that the applicant has reasonably determined the required fire protection features for preventing or mitigating fire accident scenarios that could lead to unacceptable performance, in accordance with the requirements in 10 CFR 70.61.

The applicant's ISA Summary has adequately addressed active-engineered fire protection systems in accordance with the guidance established in NUREG-1520 (NRC, 2002). In addition, the applicant's active-engineered fire protection systems meet the requirements of 10 CFR 30.33, 10 CFR 40.32, 10 CFR 70.22, 10 CFR 70.64, and 10 CFR 70.65 as they pertain to the fire protection aspects of the facility.

7.3.3.3 Facility Design Conclusions

The applicant has addressed building construction, fire area determination, electrical installation, life safety, drainage, and lightning protection adequately in the application. A description of ventilation characteristics, as they relate to fire protection and fire hazards, will be provided in the FHA. As stated in the introduction to Section 7.3 of this SER, the applicant meets the baseline design criteria through compliance with accepted consensus standards. The applicant meets defense-in-depth requirements as evaluated in Chapter 7 of this SER.

Consistent with the acceptance criteria from Section 7.4.3.3 of NUREG-1520 (NRC, 2002), the application documents the fire safety considerations used in the general design of the facilities containing licensed material, or facilities that impose an exposure threat to radiological facilities and is, therefore, acceptable. In addition, the applicant's facility's fire protection features meet the requirements of 10 CFR 30.33, 40.32, 70.22, 70.61, 70.64, and 70.65 as they pertain to the fire protection aspects of the facility.

7.3.4 Process Fire Safety

The applicant plans to use gas centrifuge machines to enrich uranium up to 5 weight (wt.) percent uranium-235 (^{235}U). The feed material will be UF_6, which will be enriched by using 96 enrichment cascades with hundreds of gas centrifuge machines per cascade. The UF_6 normal feed is 0.711 wt. percent ^{235}U; the expected product is a maximum of 5.0 wt. percent ^{235}U; and the depleted tails are typically 0.2 wt. percent ^{235}U. Enriched and depleted UF_6 streams are withdrawn from the cascade by pumps and returned to a solid phase in product and tails low-temperature take-off stations, respectively. The remainder of this section describes the NRC's staff review of key fire hazards and risks associated with the proposed facility and the gas centrifuge enrichment process.

7.3.4.1 Process Descriptions

UF_6: UF_6 is not flammable and does not disassociate to flammable constituents under conditions at which it will be handled at the facility. UF_6 does not react with oxygen, nitrogen, carbon dioxide, or dry air; but does react with water or water vapor. Hydrocarbons can be explosively oxidized if they are mixed with UF_6 in the liquid state or at elevated temperatures. For this reason, non-fluorinated hydrocarbon lubricants are not utilized in the UF_6 processes at the facility. UF_6 pumps are lubricated using a perfluoropolyether (PFPE) oil that is non-combustible. The used PFPE oil will be collected, packaged, and shipped offsite for disposal at a licensed low-level radioactive waste facility.

Hydrogen Fluoride (HF): HF is a byproduct of the chemical reaction of UF_6 with water vapor. HF is extremely reactive in both gaseous and aqueous form. HF alone is not flammable nor combustible. It can, however, react exothermically with water to generate sufficient heat to ignite nearby combustibles.

Uranyl Fluoride (UO_2F_2): UO_2F_2 is also a byproduct of the chemical reaction of UF_6 with water vapor. UO_2F_2 is stable in air to 300° C (572° F). It is neither flammable nor combustible and will not decompose to combustible constituents under conditions that will exist at the facility.

Centrifuge Machines and Components: The Model TC-12 centrifuge contains a rotor assembly—under a vacuum—inside an aluminum outer casing. The casing also provides a vacuum enclosure outside the rotor to reduce drag. The rotor is driven by an electromagnetic motor. The only combustibles of any significance are the electrical cabling going to the drive motors. Therefore, any fire originating in one of the cascades will most likely result in limited damage to the centrifuge and its components, resulting in a small release.

Control Room: The Control Room will be provided with automatic smoke detection throughout. Additionally, the Control Room will house the fire alarm control panel and will be continuously staffed. Hand-portable fire extinguishers will be provided in accordance with the National Fire Protection Association (NFPA) Standard 10 (NFPA, 2007a). IROFS boundaries will include appropriate electrical separation from normal instrument and control functions to ensure that fire induced spurious actuations do not occur. Based on the current design, all active engineered components that are IROFS will fail in the safe configuration.

Storage and Handling of UF_6: UF_6 cylinders are stored or handled in the cylinder storage pads, the CRSB, the UF_6 handling areas, and the blending and liquid-sampling areas. On the storage pads, fire concerns include the cylinder transport vehicle, a fire exposure from nearby vegetation, and fire exposure from a nearby vehicle accident. The applicant performed evaluations of these various fire scenarios and either concluded that they did not pose a threat to the stored cylinders or that, with adequate controls, the threats could be adequately mitigated by IROFS discussed in Section 7.3.2 of this chapter. In the CRSB, the primary fire concern was from a truck fire at the loading dock. The applicant also analyzed this and determined that the cylinders could be adequately protected by storing them at least 1 m (3.3 feet [ft]) from the edge of the loading dock. Combustible loadings in the UF_6 handling areas and the blending and liquid-sampling areas are limited, and transient combustibles will be controlled via IROFS43. Therefore, any fire originating in these areas will be limited.

Hydrogen Control: Hydrogen is used within the laboratories and may be generated at battery-charging stations in the facility. The laboratory will be protected by one or more of the following features:

- Hydrogen piping will be provided with excess flow control;
- Hydrogen supply will be isolated by emergency shutoff valves interlocked with hydrogen detection in the areas served by the hydrogen piping; and
- Natural or mechanical ventilation will be provided to ensure that hydrogen concentrations do not exceed 25 percent of the lower explosive limit. If mechanical ventilation is provided, it will be continuous or will be interlocked to start on detection of hydrogen in the area. Mechanical ventilation will also be provided with airflow sensors to sound an alarm if the fan becomes inoperative.
- Hydrogen control in battery-charging stations will be provided by measures identified in NFPA 70E (NFPA, 2004) and the American National Standards Institute (ANSI) Standard C2 (ANSI, 1981).
- Combustible Material Hazards: Materials of construction for the centrifuge process building, the supporting buildings, and centrifuge machines and components are predominantly non-combustible (e.g., steel, aluminum, and concrete floors). A minimum of fixed combustibles is expected to be present in the operations areas, and the applicant plans to control transient combustibles to minimize potential fire hazards by utilizing IROFS43 and general housekeeping procedures. Other quantities of combustible materials are as follows:
- Silicone oil in the UF_6 handling area and the blending and liquid sampling area is contained within the heater and chiller units associated with the cold traps, with each unit containing approximately 50 L (13.2 gal) of oil. The oil for each heater/chiller system will remain at a maximum temperature below the oil flash point temperature (with sufficient margin). High temperature switch cutout controls for individual units are set below the flash point.
- Oxygen gas (oxidizer), acetylene gas, and propane gas in the TSB workshops and laboratories;

- Acetone, toluene, petroleum ether, and petroleum ether in the TSB laboratories; and

- Primus gas, degreaser solvent, penetrating oil, and cutting oil in the TSB workshops.

7.3.4.2 Process Fire Safety Conclusions

Consistent with the acceptance criteria from Section 7.4.3.4 of NUREG-1520 (NRC, 2002), the application identifies the hazardous chemicals, processes, and design standards used to ensure safety in areas that have fire hazards that may threaten licensed material, and is, therefore, acceptable. There were no fire safety IROFS directly inherent to the process design. The fire safety IROFS used to protect the licensed material within the processes are analyzed in the individual discussions in Section 7.3.2 of this chapter. In its review, the staff has taken into account the potential presence of identified combustibles in the various accident scenarios. The identification of fire hazards and related analyses are documented in the applicant's ISA Summary (AES, 2009b and AES, 2010) that provides the supporting safety basis for the SAR (AES, 2009a).

7.3.5 Fire and Emergency Response

The facility will maintain a fire brigade consisting of employees trained in fire fighting techniques, first aid procedures, and emergency response. The fire brigade is organized, operated, trained, and equipped in accordance with NFPA 600 (NFPA, 2005a) for incipient fire fighting capability. The intent of the facility fire brigade is to be able to handle all minor fires and to provide a first-response effort designed to supplement the local fire department for major fires at the plant. The plant fire brigade, working with the plant's Emergency Operations Center, will coordinate offsite fire department activities to ensure moderator control and criticality safety. The fire brigade is staffed so that there is a minimum of five brigade members available per shift.

Periodic training is provided to offsite assistance organization personnel in the facility emergency training procedures. Facility emergency response personnel meet at least annually with each offsite assistance group to accomplish training and review items of mutual interest, including relevant changes to the program. The primary agency that will be available for this response is the City of Idaho Falls, Idaho Fire Department. This agency is a signatory to the Bonneville County Mutual Aid agreement and can request assistance from other signatory agencies including adjacent municipal fire departments and the fire/emergency response services of the US Department of Energy's Idaho National Laboratory as warranted. The applicant has received a letter from the Bonneville County Fire Protection District 1 (which includes response by the Idaho Falls Fire Department) including a commitment to fire protection and emergency response to the Eagle Rock Enrichment Facility (EREF) (AES, 2009a). A copy of this letter is included in Attachment 2 to the EREF Emergency Plan (AES, 2009a). The training and conduct of emergency drills is discussed in the EREF Emergency Plan and defines the fire protection and emergency-response commitments between the organizations. The Idaho Falls Fire Department is comprised of a roster of approximately 100 paid personnel—24 response personnel per shift, staffing five fire stations in a three-shift rotation. The department has five front-line engine companies (pumpers) and five in reserve, one 30 m (100 ft) telescoping platform, one heavy rescue truck and four light duty rescue/wildland trucks, three water tenders (6813 L [1800 gal], 11,356 L [3000 gal], and 12,114 L [3200 gallon]) tankers, one hazmat response vehicle, several command vehicles and ten ambulances equipped to provide advanced level life support. Six ambulances are staffed per shift with four in reserve.

The estimated response time to the EREF for a basic life support ambulance is 26 minutes with a second ambulance available within an additional 1-3 minutes. The EREF personnel will be trained and equipped to provide first aid and circulatory/respiratory support in the interim (e.g., provide cardiopulmonary resuscitation (CPR), apply automatic external defibrillation, and administer oxygen). The estimated response time to EREF for a structural fire engine and a full structural crew is 28 minutes with a second engine company within an additional 1-3 minutes. The initial response for a structural first alarm would be three engines, a rescue truck, an ambulance, and a staff officer. In the event of a fire, the EREF fire brigade will respond and the Idaho Falls Fire Department will be notified to respond.

The NRC staff concludes that the onsite fire brigade, onsite water supply, onsite hose lines, and mutual aid from adequately equipped fire departments can: adequately provide defense-in-depth protection from releases from all identified and credible fire scenarios, satisfy the requirements of 10 CFR 70.64(b), and are in accordance with the guidance in NUREG-1520 (NRC, 2002).

Consistent with the acceptance criteria from Section 7.4.3.5 of NUREG-1520 (NRC, 2002), the SAR (AES, 2009a) documents the fire protection systems and fire emergency response organizations that are provided and is, therefore, acceptable. In addition, the applicant's emergency response capability meets the requirements of 10 CFR 30.33, 40.32, 70.22, 70.61, 70.64, and 70.65, as they pertain to the fire protection aspects of the facility.

7.4 Evaluation Findings

The dominant fire risk to safety and health of workers and the public for the proposed process is a fire that could lead to loss of confinement of UF_6. This includes a fire damaging the centrifuge machines and piping that provide UF_6 confinement, or UF_6 cylinders inside or on the outdoor storage pad. The applicant's submittals provide sufficient information in accordance with requirements of 10 CFR 30.33, 10 CFR 40.32, 10 CFR 70.22, and 10 CFR 70.65 regarding potential fire hazards, consequences, and required controls for the proposed processes. The NRC staff determined that the applicant demonstrated compliance with the performance requirements of 10 CFR 70.61 for fire protection related to postulated accident scenarios. The applicant has identified a reasonable set of IROFS and defense-in-depth protection to ensure acceptable risks within the performance requirements of 10 CFR 70.61.

Based on the design of the facility, relative to fire protection and the designation of IROFS and measures that provide defense-in-depth, the staff concludes that the facility also meets the requirements of 10 CFR 30.33, 10 CFR 40.32, 10 CFR 70.22, 10 CFR 70.64 (a)(3) regarding baseline design criteria for protection against fires and explosions, and 10 CFR 70.64 (b) defense-in-depth.

7.5 References

(AES, 2009a) AREVA Enrichment Services, "Revision 1 to the License Application for the Eagle Rock Enrichment Facility," April 23, 2009.

(AES, 2009b) AREVA Enrichment Services, "Eagle Rock Enrichment Facility ISA Summary," April 23, 2009.

(AES, 2010) AREVA Enrichment Services, "Eagle Rock Enrichment Facility ISA Summary," April 23, 2009.

(NFPA, 2004) National Fire Protection Association, NFPA 70E, "Standard for Electrical Safety in the Workplace," 2004.

(NFPA, 2005a) National Fire Protection Association, NFPA 600, "Standard on Industrial Fire Brigades," 2005.

(NFPA, 2005b) National Fire Protection Association, NFPA 1410, "Standard on Training for Initial Emergency Scene Operations," 2005.

(NFPA, 2006) National Fire Protection Association, NFPA 220, "Standard on Types of Building Construction," 2006.

(NFPA, 2007a) National Fire Protection Association, NFPA 10, "Standard for Portable Fire Extinguishers," 2007.

(NFPA, 2007b) National Fire Protection Association, NFPA 13, "Standard for the Installation of Sprinkler Systems," 2007.

(NFPA, 2007c) National Fire Protection Association, NFPA 20, "Standard for Installation of Stationary Pumps for Fire Protection," 2007.

(NFPA, 2007d) National Fire Protection Association, NFPA 72, "National Fire Alarm Code," 2007.

(NFPA, 2008a) National Fire Protection Association, NFPA 22, "Standard for Water Tanks for Private Fire Protection," 2008.

(NFPA, 2008b) National Fire Protection Association, NFPA 70, "National Electric Code," 2008.

(NFPA, 2008c) National Fire Protection Association, NFPA 801, "Standard for Fire Protection for Facilities Handling Radioactive Materials," 2008.

(NFPA, 2008d) National Fire Protection Association, NFPA 2001, "Standard on Clean Agent Fire Extinguishing Systems," 2008.

(NRC, 2002) U.S. Nuclear Regulatory Commission, NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," March 2002.

(NRC, 2004) U.S. Nuclear Regulatory Commission, NUREG-1805, "Fire Dynamics Tools (FDT^s) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program," December 2004.

(NRC, 2005) U.S. Nuclear Regulatory Commission, "Confirmatory Calculations for Fire Protection Review of National Enrichment Facility Integrated Safety Analysis (ISA) Summary," March 2005.

CHAPTER 8.0 EMERGENCY MANAGEMENT

The purpose of reviewing the AREVA Enrichment Services's (AES's) Emergency Plan (EP) (AES, 2009) is to determine if AES has established adequate emergency management facilities and procedures to protect workers, the public, and the environment.

8.1 Regulatory Requirements

The regulations in Title 10 of the *Code of Federal Regulations* (CFR) 70.22(i)(1)(ii) require applicants to submit an EP if they are requesting authorization to possess (1) enriched uranium or plutonium for which a criticality accident alarm system is required, (2) uranium hexafluoride in excess of 50 kilograms (kg) (110 pounds[lb]) in a single container or 1,000 kg (2,200 lb) total or (3) plutonium in excess of 2 curies (Ci) in unsealed form or on foils or plated sources and their evaluation shows that the maximum dose to a member of the public offsite from a release of radioactive materials would exceed 0.01 Sv (1 rem) effective dose equivalent or an intake of 2 milligrams of soluble uranium. The regulatory requirements for the information that must be included in an EP are outlined in 10 CFR 70.22(i)(3). In addition, 10 CFR 70.64(a)(6) requires applicants to address baseline design criteria, including emergency capability to maintain control of licensed material and hazardous material produced from licensed material, evacuation of on-site personnel, and onsite emergency facilities and services that facilitate the use of available offsite services.

8.2 Regulatory Acceptance Criteria

The acceptance criteria for the NRC's review of the emergency management plan are outlined in Section 8.4.3 of NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility" (NRC, 2002).

8.3 Staff Review and Analysis

8.3.1 Facility Description

Section 1.0 of the EP (AES, 2009) contains descriptions of the licensed activity, the facility and site, and the area near the site. The information provided includes:

1. A description of the enrichment process;

2. A discussion of chemicals of concern, including form, physical state, location, and quantity (uranium hexafluoride (UF_6) has been identified as the only direct chemical of concern to be used at the facility);

3. A detailed description of the site location and layout;

4. A description of the major structures to be located at the site;

5. A description of the ventilation systems, including stack heights, maximum flow rates and filter efficiencies;

6. A description of the area near the site, including area land-use information;

7. Demography of the area, water use, and climate; and

8. Detailed maps of the facility and surrounding area.

As described in Section 10 of the EP (AES, 2009), the applicant will comply with the *Emergency Planning and Community Right-to-Know Act of 1986*, in accordance with 10 CFR 70.22(i)(3)(xiii). This is accomplished by conducting annual inventories; compiling the inventory information and providing it to the appropriate agencies; advising or training construction and operating personnel regarding the marking, storage, and location of all hazardous chemicals; and reporting accidental releases of these substances. This meets the requirements of 10 CFR 70.22(i)(3)(i).

8.3.2 Onsite and Offsite Emergency Facilities

Section 6.0 of the EP (AES, 2009) contains descriptive information regarding the emergency response equipment and facilities. The primary Emergency Operations Center (EOC) is in the Control Room, which is located in the Operational Support Building. The Eagle Rock Enrichment Facility (EREF) EOC controls communications to all principal points within and outside the facility. The EREF EOC contains current, as-built drawings; procedures; and operational engineering information to assist in routine operations and in emergency responses. The EREF EOC supports the following functions:

1. Assessment of abnormal conditions;

2. Determination of offsite protective action recommendations (PARs);

3. Emergency Response Organization (ERO) notification;

4. Notifications to county, State, and Federal emergency response authorities;

5. Direction and performance of accident mitigation;

6. Direction of facility operations; and

7. Implementation of onsite protective actions.

The backup EREF EOC is in the Security and Secure Administration Building, which is the principal entry area for the facility. Depending on the nature and location of the emergency situation, or if the Control Room becomes uninhabitable, the Emergency Director may move the emergency response personnel to the backup EREF EOC. The same documentation that is available in the primary EREF EOC is also available in the backup EREF EOC. The applicant describes these two EREF EOCs as being located in different and physically separated buildings, and it is very unlikely that both areas would be unavailable simultaneously.

Offsite emergency support and equipment are described in Section 4.3 of the EP (AES, 2009). This section provides information regarding fire, emergency medical services, and local law enforcement. A Memorandum of Understanding (MOU) has been established between the

EREF and the Bonneville County Fire Protection District 1 (including the Idaho Falls Fire Department) for fire and medical emergency services. MOUs have also been established with the Bonneville County Sheriff's Office for law enforcement services and Eastern Idaho Regional Medical Center for medical treatment facility services.

Section 6.4 of the EP (AES, 2009) describes the emergency monitoring equipment that is available including: (1) personnel monitoring equipment; (2) liquid effluent monitors; (3) gaseous effluent monitors; (4) hydrogen fluoride monitors; and (5) a meteorological measurement system for wind speed, direction, temperature and humidity.

Emergency equipment will be inventoried and tested on a quarterly basis, as discussed in Section 7.6 of the EP (AES, 2009). Deficiencies identified will be reported to the Safety, Security and Emergency Preparedness Manager, who will ensure that timely corrective action is taken.

Section 6.2 of the EP (AES, 2009) describes the communications systems, which include facility telephones that have facsimile, call tracing or call recording capabilities; the public address system; radios; and site alarms. The communications systems are designed with redundant devices for emergency communications so that a failure in one system does not leave the facility without communications capability. Cellular and satellite telephones are used to supplement and back up the primary telephone systems.

The radio systems that support the in-facility response have multichannels and high/low band capabilities to support offsite communications systems. These bandwidths provide the capability to communicate with the hospital; ambulance services; fire department; State, county, and local law enforcement agencies; and the Idaho Bureau of Homeland Security (BHS). The radio base stations are powered by diesel-backed electrical sources and remain operative following loss of offsite power. This meets the requirements of 10 CFR 70.64(a)(6)(iii).

8.3.3 Types of Accidents

Section 2.0 of the EP (AES, 2009) identifies postulated events that have high and intermediate consequences. Accident sequences, as well as mitigating and preventive measures, are also described. Since the EREF operates with only natural and low enriched (i.e., no reprocessed) uranium in the form of UF_6, there are no radiological hazards associated with the operation that could likely result in any significant offsite radiation doses. The only significant impact to the public safety is that associated with the potential release of UF_6 to the atmosphere. The applicant states that the possibility of a nuclear criticality incident occurring at the EREF has been determined to be highly unlikely. To prevent or limit the impact of any chemical or radiological release, the facility has been designed with operational safeguards appropriate to a modern chemical plant. The licensee performed a consequence analysis for a nuclear criticality scenario and for the Blending Donor Station Heater Controller Failure/Heater Run-Away Scenario. The results of these analyses are provided in Section 3.7.3 of the Integrated Safety Analysis Summary, which is contained in AES's license application. This meets the requirements of 10 CFR 70.22(i)(3)(ii).

8.3.4 Classification of Accidents

AES has established Emergency Action Levels consistent with Appendix A, "Examples of Initiating Conditions," to NRC Regulatory Guide 3.67, "Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities" (NRC. 1992). Section 3.1 of the EP (AES, 2009) explains the process used to classify an emergency as either an Alert or a Site Area Emergency declaration, and defines both types of incidents. The applicant has established that the threshold for escalating an event from an Alert to a Site Area Emergency declaration is based on indications of a release (e.g., indications such as the presence of a white vapor cloud issuing from buildings that house UF_6) that could require a response by an offsite response organization and to protect persons offsite from reaching the exposure limits set forth in 10 CFR 70.22(i)(1)(i). This process for classifying events is acceptable to the staff and meets the requirements of 10 CFR 70.22(i)(3)(iii). The processes for making the appropriate classification will be provided in the Emergency Plan Implementing Procedures.

Table 3.1-1 in the EP (AES, 2009) provides examples of site-specific incidents and the emergency classification that will be declared for each event. The Production Supervisor is responsible for accident classification and assumes the responsibility of the Emergency Director until relieved by the Plant Manager or his designee. This meets the requirements of 10 CFR 70.22(i)(3)(iii).

8.3.5 Detection of Accidents

Sections 2.2 and 6.4 of the EP explain the methods and systems available to detect accidents at the facility, including:

1. Visual observation of fire or UF_6 release,

2. Liquid and gaseous effluent monitors,

3. Hydrogen Fluoride monitors,

4. Various pressure and temperature monitors,

5. Criticality event alarm systems, and

6. Automatic fire and smoke detectors.

Action taken, in response to accidents, will be outlined in detailed procedures; and implementation will be directed by the Emergency Director. This meets the requirements of 10 CFR 70.22(i)(3)(iv).

8.3.6 Mitigation of Consequences

Section 5.3 of the EP (AES, 2009) describes actions and equipment that will be used to mitigate the consequences of accidents at the facility. The major hazard would be the chemical hazard caused by a release of UF_6. The main features used at the facility to mitigate the consequences of accidents due to a UF_6 release or a criticality event, include automatic interruption or termination of specific operations, fire detection and suppression systems, operator response to abnormal conditions/alarms, and shutdown of the ventilation system. This meets the requirements of 10 CFR 70.22(i)(3)(v).

8.3.7 Assessment of Releases

Section 5.2 of the EP (AES, 2009) describes the actions that will be taken to assess the extent of an accident at the facility. In case of an Alert declaration, dose projections of offsite radiation and hazardous material exposures will be made and provided to offsite emergency response agencies. Environmental monitoring and sampling of areas off of the EREF site will be coordinated between Bonneville County and BHS through Annex C, "Radiological Incident Response," of the Bonneville County Emergency Operations Plan—and the Idaho Hazardous Materials Incident Command and Response Support Plan. Projections of offsite radiation exposures will be based on the estimated amount released, the point of release; and the meteorological conditions at the time of the release, and will be performed using the Radiological Assessment System for Consequence Analysis software. This meets the requirements of 10 CFR 70.22(i)(3)(vi).

8.3.8 Responsibilities

The responsibilities of facility personnel during normal operations and during emergency situations are described in Sections 4.1 and 4.2, respectively, of the EP (AES, 2009). In case of an emergency, the Production Supervisor assumes the duties of the Emergency Director until the Plant Manager or a designee arrives. In 4.2.1 of the EP (AES, 2009), the applicant states that the position of Production Supervisor is filled at all times during normal operations and backshifts. As stated in Section 4.2.1 of the EP, the Emergency Director coordinates the response effort. Specific responsibilities of the Emergency Director include:

Non-Delegable Responsibilities

1. Declaring or terminating emergency events;

2. Notifying State, county and Federal emergency response authorities (the Emergency Director is responsible for notification accuracy and timeliness; however a communicator may actually make the notifications); and

3. Providing PARs to authorities responsible for implementing offsite emergency measures.

Delegable Responsibilities

1. Determining onsite protective actions;

2. Directing the assessment of actual or potential consequences, both onsite and offsite throughout the event;

3. Requesting support from offsite agencies; and

4. Approving press statements prior to their release.

Section 4.2.2.2 of the EP summarizes the responsibilities of the remaining on-call ERO. Section 4.3 provides the local offsite assistance to the facility for fire, emergency medical services, and local law enforcement. MOUs have been established with local agencies, and will be reviewed annually and renewed, if necessary. This meets the requirements of 10 CFR 70.22(i)(3)(vii).

8.3.9 Notification and Coordination

As discussed in Section 8.3.4 of this Safety Evaluation Report, the classification of emergencies is outlined in Section 3.1 of the EP (AES, 2009) and is the responsibility of the Emergency Director. Section 3.2 of the EP (AES, 2009) provides a clear commitment to promptly notify offsite EROs of an emergency, including the notification to the NRC Operations Center within 60 minutes. Sections 4.3 and 4.4 of the EP (AES, 2009) provide an adequate description of provisions for assistance from offsite emergency response organizations. These sections adequately describe the agreements held between AES and local offsite EROs and agencies, procedures for the access to the site for these response organizations, and equipment and services available from these organizations.

The Offsite EOC Liaison is dispatched to the Bonneville County EOC when activated and provides event and facility response details to county authorities. The Offsite Incident Command Liaison is dispatched to the near-site Command Post when one is established and will provide facility layout, event status, and response details to the Incident Commander. Also, mutual aid agreements are in place to provide additional support if other services or equipment is not available.

As discussed previously, it is the responsibility of the Emergency Director to: (1) declare an Alert or Site Area Emergency, (2) activate the onsite ERO, (3) notify offsite emergency response authorities of an emergency, (4) notify the NRC Operations Center, (5) decide which onsite protective actions to initiate, (6) decide which offsite protective actions to recommend and provide that information to offsite emergency response authorities, (7) decide to request support from offsite organizations, and (8) decide to terminate the emergency or enter recovery mode. This meets the requirements of 10 CFR 70.22(i)(3)(viii).

8.3.10 Information to be Communicated

Section 3.3 of the EP (AES, 2009) provides an adequate description of the type of information to be given to offsite EROs during an emergency. AES will use the EREF Event Notification Form as a script for initial notification of an emergency at the facility to appropriate offsite emergency response authorities. The emergency notification form or an NRC Form 361A, "Fuel Cycle and Materials Event Notification Worksheet," may serve as the basis for follow-up messages to the offsite emergency response authorities. If a Site Area Emergency is declared, PARs for the public are provided to the State and county agencies. Specific recommendations would depend on the event in progress (e.g., amount of UF_6 released, concentration of UF_6 expected offsite, and meteorological conditions). In most cases, recommendations would involve avoiding the area of the facility until the event concludes. This meets the requirements of 10 CFR 70.22(i)(3)(ix).

8.3.11 Training

Section 7.2 of the EP (AES, 2009) describes the training AES will provide to workers on how to respond to an emergency. All workers receive general employee training, which includes quality assurance; radiation protection (including the use of dosimetry and protective clothing); and safety, emergency, and administrative procedures. Training in criticality safety, radiation protection, and emergency procedures, specific to each type of job function, is also provided under the nuclear safety training program. Emergency response personnel receive additional training to provide specific information about how the ERO responds during emergency conditions; including staffing, determining and estimating potential offsite releases of radiation and chemicals, and interfacing with offsite assistance organizations. This training is required before an individual is assigned to the emergency organization, and refresher training is provided at least once every year.

The nuclear safety training program includes: (1) instructions to workers, (2) practices designed to keep radiation exposures As Low As Reasonably Achievable, (3) contamination control methods, (4) use of monitoring equipment, (5) emergency procedures and actions, (6) nature and sources of radiation, (7) safe use of chemicals, (8) biological effects of radiation, (9) use of personnel monitoring devices, (10) principles of nuclear criticality safety, (11) risk to pregnant females, (12) radiation protection practices, (13) protective clothing, (14) respiratory protection, and (15) personnel surveys.

Specific topics, performance objectives, content, training schedules, and the number of training hours required for each position are contained in the administrative procedures. The contents of the formal nuclear safety training programs are reviewed and updated as required at least every 2 years by the Environmental Health, Safety, and Licensing Manager—or his or her designee—to ensure that the programs are current and adequate. Individuals requiring unescorted access to the Controlled Area receive annual retraining.

Facility tours and classroom training are also provided to offsite response organizations. The training includes: (1) information concerning facility access control, (2) potential accident scenarios, (3) emergency action levels, (4) notification procedures, (5) exposure guidelines, (6) personnel monitoring devices, (7) communications, (8) contamination control, and (9) the role of the offsite assistance organizations in responding to an emergency at the facility. Physicians associated with the Eastern Idaho Regional Medical Center are offered annual emergency training involving the transportation and treatment of radiologically or chemically contaminated patients. This meets the requirements of 10 CFR 70.22(i)(3)(x).

8.3.12 Safe Shutdown (Recovery and Facility Restoration)

Section 9.0 of the EP (AES, 2009) states that during an emergency, immediate action is directed towards limiting the consequences of the incident so as to afford maximum protection to facility personnel and the general public. Once corrective measures have been taken and effective control of the facility has been reestablished, a systematic and planned approach to normal operations is taken. Depending on the type and extent of the event, the Emergency Director may elect to transition to a Recovery Organization, rather than reverting to the normal day-to-day organizational structure. The purpose of establishing a Recovery Organization is to provide dedicated personnel and structure to the handling of repair, restoration, and recovery planning and activities.

Section 9.1 of the EP (AES, 2009) states that the following criteria are to be considered prior to terminating from an emergency event:

1. Conditions requiring emergency classification no longer exist;

2. Any fire, flood, earthquake or similar initiating event no longer exists;

3. The extent of damage and condition of facility systems and equipment is understood;

4. Radioactive and/or hazardous chemical releases have been controlled, such that further incidents will be prevented;

5. Environmental assessment activities in progress are minor and only necessary to determine the extent of impact from the event, not the active tracking of material still being released;

6. Facility radiation, contamination, and hazardous chemical levels are stable or decreasing and acceptable, given current conditions;

7. All required notifications have been made;

8. The EREF EOC was staffed and activated; and

9. Offsite conditions do not unreasonably limit access of outside support to the facility.

Section 9.1 further discusses that it is not necessary that all criteria listed above be met to terminate from an emergency event; however, they all must be considered. For example, it is possible that some conditions may remain that exceed an emergency action level, but the state of emergency no longer exists. This meets the requirements of 10 CFR 70.22(i)(3)(xi).

8.3.13 Exercises and Drills

Section 7.3 of the EP (AES, 2009) provides adequate provisions for drills and biennial exercises that are used to test the adequacy of procedures, emergency equipment and instrumentation, and to ensure that all emergency response personnel are familiar with and proficient in their duties. Section 7.4 of the EP provides that post-drill and post-exercise evaluations will be conducted by those involved, and appropriate improvements will be implemented as deemed necessary by the licensee. Areas evaluated include the adequacy of the EP, procedures, equipment, facilities, personnel training, and overall response effectiveness. The Safety, Security and Emergency Preparedness Manager is responsible for the planning, scheduling, and conducting of emergency response drills and exercises for the facility.

Offsite EROs are invited to participate in the biennial exercise, and the NRC is invited to participate or observe. Exercise objectives and scenarios will be submitted to the NRC for review and comment at least 60 days before the exercise. Section 7.3 of the EP (AES, 2009) includes an adequate provision for quarterly communications checks to verify the operability of equipment used by the ERO to communicate with offsite agencies and response organizations. This meets the requirements of 10 CFR 70.22(i)(3)(xii).

8.4 Evaluation Findings

The NRC staff has evaluated AES's EP (AES, 2009) for the facility. The licensee has established an EP for responding to the radiological hazards resulting from a release of radioactive material or hazardous chemicals relating to the processing of licensed material in accordance with 10 CFR 70.22(i)(1)(ii). The NRC staff reviewed AES's EP with respect to 10 CFR 70.22(i)(3), 70.64(a)(6), and the acceptance criteria in Section 8.4.3 of NUREG-1520. The NRC staff concluded that AES's EP is adequate to demonstrate compliance with the regulatory requirements, in that: (1) the facility is properly configured to limit releases of radioactive materials in case of an accident; (2) a capability exists for measuring and assessing the significance of accidental releases of radioactive materials; (3) appropriate emergency equipment and procedures are provided onsite to protect workers against radiation and other chemical hazards that might be encountered after an accident; (4) a system has been established to notify Federal, State and county government agencies and to recommend appropriate protective actions to protect members of the public; and (5) necessary recovery actions have been established to return the facility to a safe condition after an accident.

The requirements of the EP are implemented through approved written procedures. Changes that decrease the effectiveness of the EP may not be made without NRC approval. The NRC will be notified of other changes that do not decrease the effectiveness of the EP within 6 months of making the changes.

8.5 References

(AES, 2009) AREVA Enrichment Services LLC, "Emergency Plan for the Eagle Rock Enrichment Facility," 2009.

(NRC, 1992) U.S. Nuclear Regulatory Commission, Regulatory Guide 3.67, "Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities," 1992.

(NRC, 2002) U.S. Nuclear Regulatory Commission, NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," 2002.

CHAPTER 9.0 ENVIRONMENTAL PROTECTION

The purpose of the U.S. Nuclear Regulatory Commission's (NRC's) review of the applicant's environmental protection plan for the Eagle Rock Enrichment Facility (EREF) is to determine whether the applicant's proposed environmental protection measures are adequate to protect the environment and the health and safety of the public, as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Parts 20, 30, 40, 51, and 70.

9.1 Regulatory Requirements

For its application to be considered acceptable, the applicant, AREVA Enrichment Services LLC (AES), must satisfy the following regulatory requirements regarding environmental protection:

1. Part 20 specifies the effluent control and treatment measures necessary to meet the dose limits and dose constraints for members of the public specified in Subparts B, D, and F; the survey requirements of Subpart F; the waste disposal requirements of Subpart K; the records requirements of Subpart L; and the reporting requirements of Subpart M.

2. CFR 30.33 specifies in part that an application for the possession and use of byproduct material will be granted provided that, among other things, the applicant's proposed equipment and facilities are adequate to protect health and minimize danger to life or property, and that the applicant is qualified by training and experience to use the byproduct material for the purpose requested in such a manner as to protect health and minimize danger to life and property.

3. 10 CFR 40.31(k) states that, "A license application for a uranium enrichment facility must be accompanied by an Environmental Report required under Subpart A of Part 51 of this chapter."

4. 10 CFR 40.32(e) states that, "In the case of an application for a license for a uranium enrichment facility, or for a license to possess and use source and byproduct material for uranium milling, production of uranium hexafluoride, or for the conduct of any other activity which the Commission determines will significantly affect the quality of the environment, the Director of Nuclear Material Safety and Safeguards or his designee, before commencement of construction of the plant or facility in which the activity will be conducted, on the basis of information filed and evaluations made pursuant to Subpart A of Part 51 of this chapter, has concluded, after weighing the environmental, economic, technical and other benefits against environmental costs and considering available alternatives, that the action called for is the issuance of the proposed license, with any appropriate conditions to protect environmental values. Commencement of construction prior to this conclusion is grounds for denial of a license to possess and use source and byproduct material in the plant or facility. As used in this paragraph, the term 'commencement of construction' means any clearing of land, excavation, or other substantial action that would adversely affect the environment of a site.[7]

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[7]On March 17, 2010, the NRC granted an exemption to 10 CFR 30.33(a)(5), 10 CFR 10.43(e), and 10 CFR 70.23(a)(7) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML093220446) authorizing AES to conduct certain preconstruction activities, including clearing land, grading, and excavation, on the proposed EREF site.

5. The term does not mean site exploration, roads necessary for site exploration, borings to determine foundation conditions, or other preconstruction monitoring or testing to establish background information related to the suitability of the site or the protection of environmental values."

6. Part 51 specifies that the applicant must submit an Environmental Report (ER) for construction and operation of a uranium enrichment facility, as required by 10 CFR 51.60(b)(1)(vii).

7. 10 CFR 70.22(a)(7) specifies that the applicant must provide a description of the equipment and facilities which will be used by the applicant to protect health and minimize danger to life or property.

8. 10 CFR 70.59 outlines the radiological effluent monitoring reporting requirements for a Part 70 licensee.

9. 10 CFR 70.65(b) specifies that an applicant for a facility must provide an "Integrated Safety Analysis (ISA) Summary" that includes a list of the items relied on for safety (IROFS) established by the applicant.

9.2 Regulatory Acceptance Criteria

The acceptance criteria guidance for NRC's review of the applicant's environmental protection program is outlined in Section 9.4.3.2 of NUREG-1520 (NRC, 2002).

9.3 Staff Review and Analysis

9.3.1 Radiation Safety

9.3.1.1 As Low As Is Reasonably Achievable (ALARA) Goals for Effluent Control

Air Effluent ALARA Goal

The applicant estimated the maximum individual committed effective dose equivalent (CEDE) for air effluents during normal operations at the proposed facility. The applicant estimated that the total air effluent releases would be less than 19.5 megabecquerel (MBq) (528 microcuries [μCi]) of uranium-234 (U-234), uranium-235 (U-235), and uranium-238 (U-238) isotopes—the principal constituents of natural, depleted, and enriched uranium that would be processed at the facility. Uranium-236 (U-236) is not considered to be a principal constituent in that it contributes significantly less than 1 percent to the total releases. Experience with similarly designed European facilities has typically shown that quantities of radioactivity in air effluent were much less than the estimated quantities, such that annual emissions are actually expected to be less than 20 grams (0.71 ounces) which corresponds to 0.506 MBq (13.7 μCi) of uranium isotopes. Therefore, the staff finds that a value of 19.5 MBq (528 μCi) uranium represents a reasonably conservative upper bound on the annual quantity of air effluent from the facility.

As noted in Table 4.12-22 of the applicant's ER, the CEDE to the maximally exposed hypothetical member of the public (teen) located at the north-northeast side of the controlled area boundary, resulting from the release to the atmosphere of 19.5 MBq (528 μCi) of uranium from gaseous release points, would be less than 0.9 microsieverts (μSv) (0.09 [mrem]) (AES, 2009b) or 0.09 percent of the 1 mSv (100 mrem) limit on dose to the public in Part 20. This estimated maximum public dose is also well below the 0.1 mSv (10 mrem) ALARA constraint on air emissions described in 10 CFR 20.1101.

A reasonable initial ALARA goal for air effluents, described in NRC Regulatory Guides 8.34 and 8.37 (NRC, 1992; NRC, 1993) is 10-20 percent of the regulatory limit. In Section 9.2 of the license application, the applicant committed to meeting the 0.1 mSV (10 mrem) ALARA constraint in 10 CFR 20.1101(d). In addition, the estimated dose to the maximally exposed member of the public is a small fraction of the ALARA goals identified in the above-referenced Regulatory Guides. The applicant's approach meets the guidance found in Section 9.4.3.2.1(1) of NUREG-1520 (NRC, 2002) that ALARA goals are set at a modest fraction of the values in 10 CFR Part 20 and, therefore, is acceptable to the staff.

Liquid Effluent ALARA Goal

The applicant estimated the maximum releases of uranium material from liquid effluents during normal operations at the proposed facility. The applicant estimated that the maximum annual quantity of radiological material in liquid effluent would be 900 becquerels (0.024 μCi) of uranium. This effluent would be dispersed as an atmospheric release of distillate from the evaporator in the Liquid Effluent Collection and Treatment System (AES, 2009b). The liquid distillate is evaporated after various precipitating and filtering steps, and the final airborne release is estimated as being less than 0.01 percent of the estimated release from gaseous effluents. Because this adds negligible source term to the atmospheric release estimates, there are negligible increases to the environmental or public radiological exposures resulting from liquid effluents. The estimated maximum public dose from the liquid effluent source term (about 1/10,000 of the estimated exposure from gaseous effluents) is well below the 0.1 mSv (10 mrem) ALARA constraint on liquid emissions described in 10 CFR 20.1101.

As noted in Section 8.7 of the applicant's ER, radioactive material may be released from the EREF as the result of gaseous and liquid effluent discharges, including controlled releases from the uranium enrichment process lines during decontamination and maintenance of equipment. The CEDE to the maximally exposed member of the public (teen) located at the north side of the controlled area boundary, resulting from the combined annual release to the atmosphere of 19.5 MBq (528 μCi) of uranium from the EREF, would be less than 0.9 μSv (0.09 mrem) (AES, 2009b).

A reasonable initial ALARA goal for liquid effluent described in NRC Regulatory Guide 8.34 (NRC, 1992) is 10-20 percent of the regulatory limit. As stated in the previous section, the applicant has committed to meeting the 0.1 mSv (10 mrem) ALARA constraint in 10 CFR 20.1101(d). In addition, the estimated dose to the maximally exposed member of the public is a small fraction of the ALARA goals identified in the above-referenced Regulatory Guides. The applicant's approach meets the guidance found in Section 9.4.3.2.1(1) of NUREG-1520 (NRC, 2002) that ALARA goals are set at a modest fraction of the values in 10 CFR Part 20 and is, therefore, acceptable to the staff.

9.3.1.2 *Air Effluent Controls To Maintain Public Doses ALARA*

In Section 4.12.2.1.5 of the applicant's ER (AES, 2009b), the applicant identified plant design features developed to assure that radiological impacts to the environment and public are well below regulatory limits. These include:

- Process systems that handle uranium hexafluoride (UF_6) operate at sub-atmospheric pressure to minimize outward leakage of UF_6;

- UF_6 cylinders are moved only when cool and when UF_6 is in solid form to minimize the risk of inadvertent release due to mishandling;

- Process off-gas from UF_6 purification and other operations passes through desublimers to solidify and reclaim as much UF_6 as possible. Remaining gases pass through high-efficiency filters and chemical absorbers to remove hydrogen fluoride (HF) and uranium compounds;

- Gaseous effluent passes through pre-filters, high-efficiency particulate air (HEPA) filters, and activated carbon filters, all of which greatly reduce the radioactive material in the final discharged effluent to very low concentrations;

- Liquid waste is routed to collection tanks and treated through a combination of precipitation, filtration, and evaporation to remove radioactive material prior to release of the distillate vapors to the atmosphere; and

- Effluent paths are monitored and sampled to ensure compliance with regulatory discharge limits.

Of primary importance in ensuring that air effluents are ALARA are the treatment systems for the plant's ten Gaseous Effluent Ventilation Systems (GEVS) and the heating, ventilating, and air conditioning (HVAC) systems for contaminated building exhaust. The GEVSs are identified as: (1) the Separations Building Modules (SBMs) Safe-by-Design GEVS (one in each of the four modules), (2) the Separations Building Modules Local Extraction GEVS (one in each of the four modules), (3) the Technical Support Building (TSB) GEVS, and (4) the Centrifuge Test and Post Mortem Facilities GEVS within the Centrifuge Assembly Building (CAB). The TSB; the Blending, Sampling & Preparation Building (BSPB); and the Centrifuge Test and Post Mortem Facilities have HVAC systems that function to maintain negative pressure and exhaust filtration for rooms served by these systems.

<u>GEVS</u>

The function of each GEVS is to remove particulate matter containing uranium and HF from potentially contaminated gas streams. Each GEVS includes ducts; prefilters; HEPA filters; potassium carbonate impregnated activated carbon filters; fans; monitors and controls; inlet and outlet isolation dampers; and a discharge stack or vent.

The SBM Safe-by-Design GEVS sub-atmospheric duct system transports potentially contaminated gases to a set of redundant filters (pre-filter, HEPA filter, potassium carbonate impregnated activated carbon filter, a final HEPA filter) and fans. The SBM Local Extraction

GEVS collects potentially contaminated gaseous effluent from local flexible hose connections that are used during cylinder connection and disconnection and maintenance activities. The cleaned gases are discharged via SBM rooftop exhaust vents to the atmosphere.

The TSB GEVS transports potentially contaminated gases to a set of redundant filters (pre-filter, HEPA filter, potassium carbonate impregnated activated carbon filter, a final HEPA filter) and fans. The cleaned gases discharge via exhaust vents on the TSB roof. The Centrifuge Test and Post Mortem Facilities GEVS has one set of filters (pre-filter, HEPA filter, potassium carbonate impregnated activated carbon filter, a final HEPA filter) and a single fan. It discharges cleaned gases via exhaust vents on the roof of the CAB.

HVAC - Potentially Contaminated Areas

The TSB Contaminated Area HVAC system has two active sets of filters (roughing filter, HEPA filter, potassium carbonate impregnated activated carbon filter, a final HEPA filter) and fans. The Ventilated Room HVAC System in the BSPB and Centrifuge Test and Post Mortem Facilities Exhaust Filtration HVAC System each have one set of filters (roughing filter, HEPA filter, potassium carbonate impregnated activated carbon filter, a final HEPA filter) and one fan.

The TSB Contaminated Area HVAC System discharges cleaned gases via exhaust vents on the roof of the TSB. The Ventilated Room HVAC System discharges cleaned gases via an exhaust vent on the roof of the BSPB. The Centrifuge Test and Post Mortem Facilities Exhaust Filtration System discharges cleaned gases via exhaust vents on the roof of the CAB.

Staff Evaluation of Air Effluent Controls

The staff evaluated the air effluent controls, including the GEVS and HVAC systems. The applicant's proposed use of HEPA filters is standard industry practice for control of air effluents that potentially contain airborne particle matter. Also, the use of activated charcoal for capture and control of HF is well established. The applicant will also prepare and maintain operational procedures that limit activities with dispersible forms of uranium to areas with appropriate air effluent controls. In some cases, these designated areas will also be fitted with local control devices (flexible hoses attached to a GEVS, hoods, and glove boxes). Therefore, based on the above, the staff finds that the applicant's controls will ensure that radiation levels to the public will remain well below regulatory limits in 10 CFR 20.1301 and ALARA air effluent goals, that the applicant's approach to effluent controls are meets the guidance found in Section 9.4.3.2.1(2) of NUREG-1520 (NRC, 2002) that the applicant describe and commit to use effluent controls to maintain public doses ALARA, and, therefore, the controls are acceptable to staff.

9.3.1.3 Liquid Effluent Controls to Maintain Public Doses ALARA

The Liquid Effluent Collection and Treatment System for the EREF includes two stages of precipitation and filtration to remove uranic material contained in liquid effluents collected from plant processes. The final process stage of evaporation releases the resulting distillate steam directly to the atmosphere without condensing vapor out of the air stream.

The proposed liquid effluent control systems will significantly reduce the quantity of uranium from 114 kilograms per year (kg/yr) entering the system to less than 0.036 grams per year (g/yr) in evaporated distillate. The uranium removed by the liquid effluent control will be disposed offsite at a licensed disposal facility. As described earlier in Section 9.3.1.1 of this report, the dose to the maximally exposed offsite member of the public from the evaporation of liquid effluent distillate is about 1/10,000 of the estimate for all gaseous effluents. This dose is much less than 1 percent of the regulatory limit found in 10 CFR 20.1301. A reasonable initial ALARA goal for liquid effluent described in Regulatory Guide 8.34 (NRC, 1992) is 10-20 percent of the regulatory limit.

Staff Evaluation of Liquid Effluent Controls

As stated above, the applicant does not anticipate any liquid discharges of licensed radioactive materials from the proposed facility, and the processing of liquid waste will result in a release to the atmosphere that is only a fraction of the gaseous effluent release. Therefore, based on the above, the staff finds that the applicant's controls will ensure that radiation levels to the public will remain well below the regulatory limits in 10 CFR 20.1301 and ALARA liquid effluent goals. The applicant's approach to effluent controls meets the guidance found in Section 9.4.3.2.1(2) of NUREG-1520 (NRC, 2002) that the applicant describe and commit to use effluent controls to maintain public doses ALARA, and, therefore, are acceptable to the staff.

9.3.1.4 ALARA Reviews and Reports to Management

In Section 4.2 of its Safety Analysis Report (SAR) (AES, 2009a), the applicant describes an ALARA program for the proposed facility. The ALARA program would include annual reviews of the content and implementation of the radiation protection program, including the effluent control program. The Radiation Protection/Chemistry Manager is responsible for implementing the ALARA program and ensuring that adequate resources are committed to make the program effective. The Radiation Protection/Chemistry Manager prepares an annual ALARA program evaluation report. As described in Section 4.2 of the SAR, the report reviews: (1) radiological exposure and effluent release data for trends; (2) audits and inspections; (3) use, maintenance, and surveillance of equipment used for exposure and effluent control; and (4) other issues, as appropriate, that may influence the effectiveness of the radiation protection and ALARA programs (AES, 2009a). Copies of the report are submitted to the AES President, Plant Manager, Radiation Safety Committee, and the Safety Review Committee. This approach adequately addresses the regulatory requirements in 10 CFR 20.1101(c) and is consistent with the guidance in Section 9.4.3.2.1(3) of NUREG-1520 (NRC, 2002) and, therefore, it is acceptable to the staff.

9.3.1.5 Waste Minimization

In Section 4.13.5 of the ER (AES 2009b), the applicant described facility features and systems that will minimize the generation of radioactive waste. These features and systems are based on principles of control, conservation, reprocessing, and recovery. Specific examples include: (a) a decontamination workshop designed to remove radioactive contamination from equipment and allow some equipment to be reused rather than treated as waste, (b) closed-loop cooling systems have been incorporated in the design to reduce water usage, (c) outer packaging associated with consumables will be removed prior to use in a contaminated area, (d) collected waste will be volume reduced at a centralized waste processing facility, and (e) use of glove boxes to minimize the spread of contamination.

A contributing program to the waste minimization effort is the facility contamination control program which is addressed in Chapter 4. Contamination levels will be reviewed by the Safety Review Committee (SRC), which fulfills the functions of the ALARA Committee, during its routine meetings. The SRC will also evaluate assessments made by the radiation protection organization and make recommendations which are tracked to completion through the facility's corrective action program.

These features and systems are consistent with the regulatory requirements in 10 CFR 20.1406; the guidance provided in NRC's Information Notice 94-23, "Guidance to Hazardous, Radioactive, and Mixed-Waste Generators on the Elements of a Waste Minimization Program" (NRC, 1994), and the guidance in Section 9.4.3.2.1(4) of NUREG-1520 (NRC, 2002), and are, therefore, acceptable to the staff.

9.3.1.6 Safe Handling of Radioactive Wastes

The applicant has identified design features and procedures for safe handling of air and liquid effluent and solid wastes from both construction and operation of the proposed facility. The staff's evaluation of air and liquid effluent is described in Sections 9.3.1.2 and 9.3.1.3 of this document.

In Section 3.12.2 of the applicant's ER (AES 2009b), the applicant has described a solid waste management program at the proposed facility for industrial (non-hazardous), radioactive mixed, and hazardous wastes. Solid waste will be grouped into one of these waste categories and, in addition, radioactive and mixed waste will be further segregated according to the quantity of liquid that is not readily separable from the solid material. The EREF may send wastes that are candidates for volume reduction, recycling, or treatment to licensed treatment facilities that have the ability to volume reduce most Class A low-level wastes (LLWs); and to process contaminated oils and some mixed wastes. The applicant will also operate several systems for reprocessing and recovery of uranic materials when practicable. The applicant does not propose to process or treat onsite solid waste that does not contain economically recoverable uranic materials.

The applicant proposes to dispose of all solid radioactive wastes as Class A LLW. Industrial waste, including miscellaneous trash; vehicle air filters; empty cutting oil cans; miscellaneous scrap metal; and paper will be shipped offsite for minimization and then sent to a permitted waste landfill.

Radioactive waste will be collected in labeled containers in each Restricted Area and transferred to the Solid Waste Collection Room for inspection. Suitable waste will be volume reduced and all radioactive waste will be disposed of at a licensed LLW disposal facility.

Hazardous wastes (e.g., spent blasting sand, empty spray-paint cans, empty propane-gas cylinders, solvents such as acetone and toluene, degreaser solvents, diatomaceous earth, hydrocarbon sludge, and chemicals such as methylene chloride and petroleum ether) and some mixed wastes will be generated at the facility. These wastes will be collected at the point of generation, transferred to the Solid Waste Collection Room, inspected, and classified. Any mixed waste that may be processed to meet land disposal requirements may be treated in its original collection container and shipped as LLW for disposal.

As noted in Section 3.12.2.1.2.9 of the applicant's ER (AES, 2009b), the operation of the facility would yield an annual production of 1,222 cylinders of depleted UF_6 per year, or approximately 15,270 metric tonnes (16,832 tons). The Full Tails Cylinder (FTC) Storage Pad would have a capacity of 33,638 cylinders. The NRC has evaluated the environmental impacts of alternatives for disposition of depleted UF_6 in its "Environmental Impact Statement for the Proposed Eagle Rock Enrichment Facility in Bonneville County, Idaho" (NRC, 2010). During temporary storage of this material on the FTC Storage Pad, the applicant will have a number of mitigation measures to minimize public and occupational health impacts. Among these measures is a cylinder management program to monitor storage conditions on the FTC Storage Pad; to monitor cylinder integrity by conducting routine inspections for breaches; and to perform cylinder maintenance, as needed. The cylinders will be stored on saddles made of materials that will not cause corrosion of the cylinders. The storage array of the cylinders will permit easy visual inspection. The cylinders will be surveyed for external contamination before being placed on the FTC Storage Pad and before being transported offsite. In addition, the cylinders will be re-inspected annually for damage or surface coating defects. These inspections are to insure that the cylinders are free from bulges, dents, gouges, cracks, or significant corrosion, and that the cylinder plugs are undamaged and not leaking. If significant deterioration is detected, then the contents of the cylinder will be transferred to another cylinder and a root cause assessment of the deterioration will be conducted.

On the basis of this analysis, the staff finds that the applicant's implementation of its program for management of solid radiological and non-radiological wastes related to facility operation reduces unnecessary exposures as per ALARA requirements in 10 CFR 20.1101(b) is consistent with waste minimization guidance in Section 9.4.3.2.1(4) of NUREG-1520 (NRC, 2002) and, therefore, is acceptable to the staff.

9.3.2 Effluent and Environmental Monitoring

9.3.2.1 Air Effluent Monitoring

As described earlier in Section 9.3.1.1, the staff finds that the radioactive materials in airborne effluents would result in exposures well below the ALARA constraint specified in 10 CFR 20.1302(c). The applicant has proposed to demonstrate compliance with air effluent limits by calculation of the total effective dose equivalent to the individual who is likely to receive the highest dose. Such a demonstration of regulatory compliance is in accordance with 10 CFR 20.1302(b)(1).

The staff reviewed the applicant's assumptions and conclusions used in its calculations in Sections 4.12 and 8.7 of the ER (AES, 2009b) and determined that they are reasonable as emissions were estimated from emissions from similarly designed plants. The staff's evaluation of radiation exposures from normal operations is found in Chapter 4 (Section 4.2.10.2, "Facility Operations") of the Draft Environmental Impact Statement (EIS) (NRC, 2010).

The applicant has identified all airborne effluent discharge locations, and will monitor discharges from potentially contaminated processes and areas in accordance with 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" (NRC, 1985) and NUREG-1302, "Offsite Dose Calculation

Manual Guidance: Standard Radiological Effluent Controls for Boiling Water Reactors" (NRC 1991). The applicant will perform continuous monitoring at the exhaust vents for alpha/beta emitting radioactivity and HF. The applicant will also perform monitoring of work areas that process dispersible uranium, but are not expected to have airborne contamination. Sampling of each contributing source will not be part of the effluent monitoring design. However, each potentially contaminated area will be monitored by health physics surveys for airborne contamination should such surveys be required for effective process and effluent control.

A list of HVAC systems, GEVSs, and exhaust filtration systems that discharge building ventilation exhaust is provided in Table 9.3-1 of this report.

Weekly samples of stack air effluent will be measured for gross alpha radioactivity. Weekly samples will be composited quarterly for isotope-specific analyses for U-234, U-235, U-236, and U-238. The applicant has proposed measurement sensitivities of 5×10^{-14} µCi/mL for weekly gross alpha measurements and 5×10^{-14} µCi/mL for uranium isotopes. This value is less than 2 percent of the effluent concentration limit for class "D" solubility uranium materials in 10 CFR 20, Appendix B. Because the uranium materials in gaseous effluent from the facility are expected to be class "D," this detection limit is acceptable to staff.

As noted in the applicant's ER (AES, 2009b), the applicant will develop a program of corrective actions to be taken when established action levels of radiation are exceeded for any of the measured parameters. As proposed by the applicant, action levels are divided into three priorities:

1. The sample parameter's lowest alarm point will be selected based on conditions of service, e.g., three times the normal background level.

2. The sample parameter exceeds any of the existing administrative limits.

3. The sample parameter exceeds any regulatory limit.

The program of corrective actions will be implemented to ensure that: (a) the cause for the action level exceedance can be identified and immediately corrected; (b) applicable regulatory agencies are notified, if required; (c) lessons learned are communicated to appropriate personnel; and (d) applicable procedures are revised accordingly, if needed.

As described in Section 1.3 of the applicant's ER (AES, 2009b), in addition to meeting NRC requirements, the applicant will also obtain required Federal and State permits for hazardous air pollutants.

As described in Section 6.1.2 of the applicant's ER (AES, 2009b), the applicant's reporting procedures comply with the requirements of 10 CFR 70.59 and the specific guidance in Regulatory Guide 4.16 (NRC, 1985).

On the basis of this analysis, including a review of the information provided by the applicant as noted above, the staff finds that the applicant's air effluent monitoring during operation of the facility will be adequate to ensure that radioactive materials in air effluent meet the regulatory limits in 10 CFR.1301; is consistent with the guidance in Section 9.4.3.2.2(1) of NUREG-1520 (NRC, 2002); and, therefore, is acceptable to the staff.

Table 9.3-1 Overview of Gaseous Effluent Monitoring Program

Sample Location	Sample Type	Analysis / Frequency
Separations Building GEVS exhaust vents TSB GEVS exhaust vent TSB Contaminated Area HVAC System exhaust vent Centrifuge Test and Post Mortem Facilities GEVS exhaust vent[a] Centrifuge Test and Post Mortem Facilities Exhaust Filtration System exhaust vent[a] Ventilated Room HVAC System exhaust vent	Continuous air particulate filter	Gross alpha/beta- Weekly Isotopic analysis[d]- Quarterly composite
Evaporator	Continuous liquid condensate sample from exhaust vent	Gross alpha/beta – Weekly Isotopic analysis[d] – Quarterly composite
Process Areas[b]	Local area continuous air particulate filter[c]	Gross alpha/beta – Weekly Isotopic analysis[d] – Quarterly composite
Non-Process Areas[b]	Local area continuous air particulate filter[c]	Gross alpha/beta – Quarterly composite

Notes:

a The continuous sampling system is operated only when the Centrifuge Test Facility or Post Mortem Facility is in operation.

b A "Process Area" is any area of the facility where UF_6 process flow between feed, product, or tails cylinders occurs—including areas where cylinders containing UF_6 are opened for testing, inspection, or sampling. A "Non-Process Area" is any other area where uranic material is present in an open form.

c These will generally be collected with mobile continuous air monitors, as required to complement the effluent monitoring program.

d Isotopic analysis for uranium if gross alpha and gross beta activities indicate that an individual radionuclide could be present in a concentration greater than 10 percent of the concentrations specified in Table 2 of Appendix B to 10 CFR Part 20.

9.3.2.2 Liquid Effluent Monitoring

As described in Section 9.3.1.2 of this SER, there are no expected liquid effluents to surface water bodies. Instead, evaporated distillate will be released to the atmosphere while residual evaporator liquids and precipitants are treated and disposed of offsite at a licensed facility. The applicant will demonstrate compliance with gaseous effluent limits as discussed in Section 9.3.2.1. This section also contains a discussion of the proposed action levels.

The staff reviewed the applicant's assumptions and conclusions used in its calculations (ER Sections 4.12 and 8.7 [AES 2009b]) and determined that they are reasonable as the emission estimates are based on similarly designed facilities. The staff's evaluation of radiation exposures from normal operations is found in Section 4.2.10.2 of the Draft EIS (NRC, 2010).

General site stormwater runoff is routed to the Site Stormwater Detention Basin. The two Cylinder Storage Pads Stormwater Retention Basins collects stormwater runoff from the Cylinder Storage Pads (i.e., Full Feed Cylinder Storage Pads, Full Tails Cylinder Storage Pads, Full Product Cylinder Storage Pad, and Empty Cylinder Storage Pads), as well as treated water from the Domestic Sanitary Sewage Treatment Plant. The applicant proposes to include sampling and analysis of water and sediment from each of the retention-detention basins in the Radiological Effluent Monitoring Program.

As described in Section 1.3 of the applicant's ER (AES, 2009b), in addition to meeting the NRC's regulatory requirements, the applicant will obtain required Federal and State permits relating to liquid discharges (primarily related to groundwater protection). In addition to having direct permitting authority, the State of Idaho also has oversight responsibility for the U.S. Environmental Protection Agency's (EPA's) water permits granted through the Idaho Environmental Protection and Health Act (Idaho Code Chapter 1, Title 39).

As described in Section 6.1.2 of the applicant's ER, the applicant has developed reporting procedures that when implemented will assure compliance with the regulatory requirements of 10 CFR 70.59 and the guidance provided in Regulatory Guide 4.16 (NRC, 1985).

On the basis of this analysis, the staff finds that the applicant's liquid effluent monitoring during operation of the facility ensures that radioactive materials in liquid effluent will meet the limits in 10 CFR 20.1301; is consistent with guidance in Section 9.4.3.2.2(1) of NUREG-1520 (NRC, 2002); and the monitoring is, therefore, acceptable to the staff.

9.3.2.3 Laboratory Quality Control

The applicant has a Radiological Environmental Monitoring Program (REMP) with associated quality assurance. The REMP provides data to confirm the effectiveness of effluent controls and the effluent monitoring program. All environmental samples will be analyzed onsite for facility-related radiological constituents. Samples may also be shipped to a qualified independent laboratory for analyses.

The Quality Control (QC) procedures used by the laboratories performing the plant's REMP will be used to validate the analytical results and will conform to guidance in Regulatory

Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination)—Effluent Streams and the Environment" (NRC, 1979). These QC procedures include the use of established standards such as those provided by the National Institute of Standards and Technology (NIST), as well as standard analytical procedures such as those established by the National Environmental Laboratory Accreditation Conference (NELAC).

Monitoring procedures will employ established analytical methods and instrumentation. The instrument maintenance and calibration program will be in accordance with manufacturers' recommendations. The onsite laboratory and any contractor laboratory used to analyze samples will participate in third-party laboratory intercomparison programs appropriate to the media and analytes being measured. All radiological and non-radiological laboratory vendors will be certified by the National Environmental Laboratory Accreditation Program (NELAP) or an equivalent state laboratory accreditation agency for the analytes being tested.

For the physiochemical monitoring program (described in the applicant's ER, Section 6.2.8), the applicant has a quality assurance program that will use a set of formalized and controlled procedures for sample collection, laboratory analysis, chain of custody, reporting of results, and corrective actions. Samples sent to laboratories will include blanks and duplicates at specified frequencies to provide data for identifying routine reporting or analytical errors as part of quality assurance checks on the data. Analyses will only be performed at laboratories with appropriate EPA and State of Idaho certifications. The laboratory analyses will be conducted using the best available standard techniques at State or EPA-certified laboratories.

The staff finds that these procedures are adequate to validate the analytical results produced by the REMP, which is described in the applicant's ER Section 6.1.2 (AES, 2009b). These procedures are consistent with regulatory the guidance in Section 9.4.3.2.2(1)(h) of NUREG-1520 (NRC, 2002); adequate to validate the analytical results from the laboratory; and therefore, acceptable to the staff.

9.3.2.4 Environmental Monitoring

The applicant has established its REMP for the facility. The REMP is a major part of the applicant's effluent compliance program. The effectiveness of the applicant's effluent controls will be confirmed through implementation of the REMP. The purpose of the REMP is to verify confinement integrity at the facility and to support the primary means of demonstrating compliance with applicable radiation protection standards for the environment and the public. Compliance is demonstrated primarily through effluent monitoring (see applicant's ER Section 6.1.2 [AES, 2009b]).

As part of the REMP, the applicant will establish background and baseline concentrations of radionuclides in environmental media through sampling and analysis. The REMP will be initiated at least 2 years before plant operations to develop a sufficient environmental baseline. The types of samples to be collected under the REMP are listed in Table 9.3-3 of this report. The staff's evaluation of the proposed corrective action levels, which apply to both effluent monitoring and the REMP, is described in Section 9.3.2.1 of this report. The scope of the applicant's REMP meets the environmental monitoring criteria found in NUREG-1520 (NRC, 2002), Section 9.4.3.3.2(2). As noted in the applicant's ER (Section 6.1.2 [AES, 2009b]), the REMP sampling locations are based on NRC guidance found in NUREG-1302, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Boiling Water Reactors," (NRC, 1991).

The applicant has an adequate and timely program to collect information to determine baseline concentrations of radionuclides, which information will be used to demonstrate compliance with applicable radiation protection standards. The staff finds that the applicant's environmental monitoring program adequately addresses the regulatory requirements in 10 CFR 20.1406, the guidance found in Section 9.4.3.3.2(2) of NUREG-1520 (NRC, 2002) and the program is, therefore, acceptable to the staff.

Table 9.3-2 Types of Samples Collected under the REMP

Sample Type/Location	Minimum Number of Sample Locations	Sampling and Collection Frequency	Type of Analysis
Continuous Airborne Particulate	5	Continuous operation of air sampler with sample collection as required by dust loading but at least biweekly. Quarterly composite samples by location.	Gross beta/gross alpha analysis each filter change. Quarterly isotopic analysis on composite sample.
Vegetation	9	1 to 2-kg (2.2 to 4.4-lb) samples collected semiannually	Isotopic analysis[a]
Groundwater	10	4-L (1.06-gal) samples collected semiannually	Isotopic analysis[a]
Basins	1 from each of 3 basins[b]	4-L (1.06-gal) water sample/1 to 2-kg (2.2 to 4.4-lb) sediment sample collected quarterly	Isotopic analysis[a]
Soil	9	1 to 2-kg (2.2 to 4.4-lb) samples collected semiannually	Isotopic analysis[a]
Domestic Sanitary Sewage Treatment Plant	1	1 to 2-kg (2.2 to 4.4 lb) solid fraction sample semiannually	Isotopic analysis[a]
TLD	18	Quarterly	Gamma and neutron dose equivalent

Notes:

[a] Isotopic analysis for Uranium.

[b] Site Stormwater Detention Basin and Cylinder Storage Pads Stormwater Retention Basins.

9.3.3 ISA Summary

In Chapter 3 of this report, the staff provides its evaluation of the ISA Summary (AES, 2009c) and documents its conclusion that the ISA Summary is complete, provides reasonable estimates of the likelihood and consequences of each accident sequence, and contains sufficient information to determine whether adequate engineering or administrative controls are identified for each accident sequence. Chapter 11 of this report contains the staff's evaluation of management measures used to ensure that IROFS will satisfactorily perform their intended safety functions. In this Section, the staff evaluates whether the ISA uses acceptable methods to estimate environmental effects that may result from accident sequences.

Under 10 CFR Part 70, Subpart H, an applicant is to assure, among other things, compliance with various performance requirements. 10 CFR 70.61(c)(3) identifies the environmental performance requirement that the applicant apply controls such that a credible intermediate consequence event is unlikely to occur or that the consequence of such an event will not exceed a 24-hour averaged release of radioactive material outside the restricted area in concentrations 5,000 times the values in Table 2 of Appendix B to 10 CFR Part 20. The restricted areas are defined by the applicant in Section 4.7.1.2 of the SAR (AES, 2009a). The restricted areas include the storage areas for UF_6 in the Cylinder Receipt and Shipping Building and the potentially contaminated areas in the TSB. Restricted areas will also include "radiation areas," "airborne radioactivity areas," "high-radiation areas," "contaminated areas," and areas where inhalation of 1 milligram (mg) soluble uranium in a week is possible without respiratory protection. These areas will be posted accordingly.

The applicant derived enhanced definitions of consequence severity levels based on the following rationale. For releases to the atmosphere of soluble uranium compounds (i.e., UF_6), the applicable value of Table 2 of Appendix B to 10 CFR Part 20 is 3×10^{-12} μCi/mL. This is the value for uranium isotopes U-234, -235, and -238. The performance requirement is, therefore, a 24-hour averaged concentration of any mixture of these isotopes that will not exceed $3 \times 10^{-12} \times 5000$, or 1.5×10^{-8} μCi/mL total radioactivity. AES assumed a bounding enrichment of 6 percent. Using footnote 3 of Part 20, Appendix B, the specific activity is approximately 2.8 μCi per gm U. Therefore, the environmental performance requirement, after conversion to a mass concentration, is a 24-hour averaged concentration of 5.4 mg U/m^3. The environmental performance requirement, expressed as the enhanced severity level of 5.47 mg U/m^3, was evaluated at the restricted area boundary. In Sections 3.7 and 3.8 of the ISA Summary (AES, 2009c), the applicant showed that it has adequately reduced the risks to the environment from accidents for which the consequences could otherwise exceed the environmental consequence severity level.

In its ISA Summary, the applicant identified various sequences for radiological and non-radiological accidents which were evaluated to ensure adequate protection of worker health and safety. By assuring that all credible high-consequence events are rendered highly unlikely and that all intermediate-consequence events are unlikely, the applicant also ensured that the environmental performance requirements of 10 CFR 70.61(c)(3) will be met. The staff determined that environmental consequences may occur only if uncontrolled, intermediate or high consequences to workers are also present. The staff did not identify any accident sequence that would fail to meet the environmental performance requirements of 10 CFR 70.61(c)(3).

The applicant's approach to risk reduction will be accomplished through a combination of preventive and mitigative measures, with an emphasis on preventive measures. A more complete discussion is found in Chapter 3 of this report, which addresses accident sequences for high and intermediate consequences. It also addresses preventive and mitigative measures. The staff finds that the applicant's ISA adequately addresses radiological and non-radiological risks to the environment consistent with 10 CFR 70.61, is consistent with guidance in Section 9.4.3.2.3 of NUREG-1520 (NRC, 2002), and the ISA is, therefore, acceptable to the staff.

9.4 Evaluation Findings

The applicant has developed a program to implement adequate environmental protection measures during operation, which measures include: (1) environmental and effluent monitoring, and (2) effluent controls to maintain public doses ALARA as part of the radiation protection program. The NRC staff concludes that the applicant's program, as described in its application and ER, is adequate to protect the environment and the health and safety of the public and complies with regulatory requirements in 10 CFR Parts 20, 30, 40, 51, and 70.

The NRC staff expects to issue a Final EIS, as required by 10 CFR 51.20, in February 2011 for this license application. The Final EIS will consider the environmental impacts of the proposed construction, operation, and decommissioning of the proposed facility and compare alternatives, to inform the NRC staff recommendation concerning the proposed license application.

9.5 References

(AES, 2009a) AREVA Enrichment Services LLC, "Eagle Rock Enrichment Facility Safety Analysis Report," Revision 1, 2009.

(AES, 2009b) AREVA Enrichment Services LLC, "Eagle Rock Enrichment Facility Environmental Report," Revision 1, 2009.

(AES, 2009c) AREVA Enrichment Services LLC, "Eagle Rock Enrichment Facility ISA Summary," Revision 1, 2009.

(NRC, 2010) U.S. Nuclear Regulatory Commission, NUREG-1945, "Draft Environmental Impact Statement for the Proposed Eagle Rock Enrichment Facility in Bonneville County, Idaho," 2010.

(NRC, 1979) U.S. Nuclear Regulatory Commission, Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operations) – Effluent Streams and the Environment," 1979.

(NRC, 1985) U.S. Nuclear Regulatory Commission, 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," 1985.

(NRC, 1991) U.S. Nuclear Regulatory Commission, NUREG-1302, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Boiling Water Reactors," 1991.

(NRC, 1992) U.S. Nuclear Regulatory Commission, Regulatory Guide 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses," 1992.

(NRC, 1993) U.S. Nuclear Regulatory Commission, Regulatory Guide 8.37, ALARA Levels for Effluents From Materials Facilities," 1993.
(NRC, 1994) U.S. Nuclear Regulatory Commission, Information Notice 94-23, "Guidance to Hazardous, Radioactive, and Mixed-Waste Generators on the Elements of a Waste Minimization Program," 1994.

(NRC, 2002) U.S. Nuclear Regulatory Commission, NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," 2002.

(PNNL, 2007) Pacific Northwest National Laboratory, GENII 2.06, 2007.

CHAPTER 10.0 DECOMMISSIONING

The purpose of the U.S. Nuclear Regulatory Commission's (NRC's) review of AREVA Enrichment Services's (AES's) decommissioning plans is to evaluate whether the applicant will be able to decommission the facility safely and in accordance with NRC requirements.

At the time of the initial License Application for a uranium enrichment facility, an applicant is required to submit a decommissioning funding plan (DFP). The purpose of NRC's review of the DFP is to determine whether the applicant has considered decommissioning activities that may be needed in the future, has performed a credible site-specific cost estimate for those activities, and has presented NRC with financial assurance (FA) to cover the cost of those activities in the future. The DFP, therefore, should contain an overview of the applicant's proposed decommissioning activities, the methods used to determine the cost estimate, and the FA mechanism. This overview should contain sufficient detail to enable the reviewer to determine whether the decommissioning cost estimate is reasonably accurate.

10.1 Regulatory Requirements

The following NRC regulations require planning, FA, and recordkeeping for decommissioning, as well as procedures and activities to minimize waste and contamination:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 20.1401-1406 "Radiological Criteria for License Termination"
- 10 CFR 30.35 "Financial Assurance and Recordkeeping for Decommissioning"
- 10 CFR 30.36 "Expiration and Termination of Licenses and Decommissioning of Sites and Separate Buildings or Outdoor Areas"
- 10 CFR 40.36 "Financial Assurance and Recordkeeping for Decommissioning"
- 10 CFR 40.42 "Expiration and Termination of Licenses and Decommissioning of Sites and Separate Buildings or Outdoor Areas"
- 10 CFR 70.22(a)(9) "Contents of Applications" (DFP)
- 10 CFR 70.25 "Financial Assurance and Recordkeeping for Decommissioning"
- 10 CFR 70.38 "Expiration and Termination of Licenses and Decommissioning of Sites and Separate Buildings or Outdoor Areas"

10.2 Regulatory Guidance and Acceptance Criteria

The guidance applicable to NRC's review of the decommissioning section of the Safety Analysis Report (SAR) is contained in Chapter 10 of "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520 (NRC, 2002a) and in Volume 3 of "Consolidated NMSS Decommissioning Guidance," NUREG-1757 (NRC, 2003). Chapter 10 of NUREG-1520 (NRC, 2002a) is applicable to the Eagle Rock Enrichment Facility (EREF), except that the review used NUREG-1757 (NRC, 2003), which is the updated version of NUREG-1727, "NMSS Decommissioning Standard Review Plan" (NRC, 2000), which is referenced in NUREG-1520 (NRC, 2002a).

NUREG-1757 (NRC, 2003) provides guidance for developing final decommissioning plans required under 10 CFR 30.36(g), 10 CFR 40.42(g), and 10 CFR 70.38(g). A final decommissioning plan will be provided at the time of decommissioning. At the time of initial licensing and for license renewals, Section 10.1 of NUREG-1520 (NRC, 2002a) describes an overview of the proposed decommissioning activities needed to develop the DFP. This overview is a more generalized discussion of the detailed information that would be needed for the final decommissioning plan described in NUREG-1757 (NRC, 2003).

With regard to the acceptance criteria, because depleted uranium (DU) deconversion services are not currently available in the U.S., DU generated in the operation of the EREF is considered as a potential decommissioning obligation in the DFP. Thus, in addition to providing FA for the decommissioning of the facility and site, AES intends to provide financial assurance for the expected costs associated with the disposition of depleted uranium tails.

The requirements in 10 CFR 40.36(d) and 10 CFR 70.25(e) specify that the DFP must contain a certification by the licensee that FA for decommissioning has been provided in the amount of the cost estimate for decommissioning. AES has requested an exemption, under 10 CFR 40.14 and 70.17, to the decommissioning funding requirements in order to incrementally fund its FA obligation as depleted uranium is generated and as each Separations Building Module (SBM) becomes operational. Section 1.2.4.2 of this SER describes the exemption request and the NRC staff's evaluation, which concluded that the exemption could be granted.

10.3 Staff Review and Analysis

The NRC's staff review of the planned decommissioning activities as described in Chapter 10 of the SAR (AES, 2010) focused on the applicant's conceptual decommissioning activities for the EREF, the decommissioning cost estimates, and the FA for decommissioning activities. The applicant identified the decommissioning activities that may be needed in the future for decommissioning and presented site-specific estimates of decommissioning costs for those activities. The applicant intends to decommission the site such that any residual radioactivity that is distinguishable from background radiation results in a total effective dose equivalent to an average member of the critical group that does not exceed 25 mrem (0.25 mSv) per year. Using the cost data as a basis, the applicant stated that it has presented FA to cover the costs required to release the EREF for unrestricted use.

The following subsections contain the NRC staff's assessment of the applicant's proposed decommissioning schedule and activities, cost estimate, and funding plan. Before license termination, the applicant will provide a detailed decommissioning plan that will include specific activities which will be used to protect workers, the public, and the environment.

10.3.1 Request for Alternate Schedule for Decommissioning

The requirements in 10 CFR 70.38(h) state that the decommissioning of the site or a separate building or outdoor area must be completed within 24 months of initiating decommissioning activities. However, 10 CFR 70.38(i) states the Commission may approve requests for alternate decommissioning schedules. In its September 28, 2009, response to NRC's August 26, 2009, request for additional information (RAI) (NRC, 2009b), the applicant requested approval of an alternate schedule for decommissioning and provided justification for the longer schedule, stating that decommissioning of each SBM would take approximately 4.5 years and completion of site decommissioning would take approximately 8 years (AES, 2009b).

The NRC staff reviewed the applicant's request for an alternate schedule for decommissioning in light of the five considerations in 10 CFR 70.38(i). These are: (1) whether it is technically feasible to complete decommissioning within the allotted 24-month period; (2) whether sufficient waste disposal capacity is available to allow completion of decommissioning within the allotted 24-month period; (3) whether a significant volume reduction in wastes requiring disposal will be achieved by allowing short-lived radionuclides to decay; (4) whether a significant reduction in radiation exposure to workers can be achieved by allowing short-lived radionuclides to decay; and (5) other site-specific factors which the Commission may consider appropriate on a case-by-case basis. The applicant addressed all five considerations in the RAI response (AES, 2009b). In regards to technical feasibility, the applicant asserted in the RAI response dated September 28, 2009, that the decommissioning of a single SBM is assumed to take 4.5 years, 3 years for decommissioning of the centrifuges and associated equipment and 1.5 years for decontamination of the structure (AES, 2009b). Dismantling and decontamination in the four SBMs will be performed in a phased approach such that the decommissioning of all four SBMs is completed within an 8-year timeframe. The primary reason for the 3-year timeframe for decommissioning the centrifuges in each SBM is the quantity of the centrifuges and associated equipment that must be decommissioned and the fact that this equipment is classified. An additional consideration is the size of the SBM itself.

Based on the applicant's justification that it is not technically feasible to complete decommissioning within a 24-month period, the staff determined that the alternate schedule is warranted and, therefore, the applicant's request is approved.

10.3.2 Conceptual Decontamination and Decommissioning Activities

Section 10.1 of NUREG-1520 (NRC, 2002a) states that the DFP is required to contain an overview of the proposed decommissioning activities. This section of the Safety Evaluation Report (SER) describes the staff's review of the overview of the proposed decommissioning activities. A detailed decommissioning plan will be provided at the time of decommissioning in accordance with 10 CFR 30.36(g), 10 CFR 40.42(g), and 10 CFR 70.38(g).

Section 10.1.4, "Decommissioning Strategy" of the SAR (AES, 2009a) notes that the plan for decommissioning the EREF is to promptly decontaminate or remove all materials from the site which prevent release of the facility and site for unrestricted use. At the end of useful plant life, the EREF will be decommissioned such that the site and remaining facilities may be released for unrestricted use as defined in 10 CFR 20.1402 (AES, 2009a). The applicant performed a site characterization survey that included collection and analysis of soil and groundwater samples to determine background levels of uranium and thorium, both of which exist in nature and will be used at the facility. The initial site characterization data is provided in sections 3.3 and 3.11 of AES's Environmental Report (AES, 2009b). In the RAI (NRC, 2009b), the staff indicated that the number of samples in the initial site characterization were not sufficient to use for demonstration of compliance with the 10 CFR Part 20 Subpart E decommissioning criteria. In an RAI response (AES, 2009b), the applicant provided the following commitment to perform sampling based on the guidance provided by Section A.3.4 of Appendix A to NUREG-1757 (NRC, 2000):

> Prior to the commencement of construction, AES will collect additional surface soil samples and analyze them for radiological constituents. The site property will be divided

into four survey units, and 15 surface soil samples will be taken per survey unit (i.e., 60 additional soil samples). The sample collections will be taken from areas that include (1) the detention and retention basins, (2) Full Tails, Full Feed, and Empty Cylinder Storage Pads north of the main facilities, (3) the Technical Services Building (TSB), Blending, Sampling and Preparation Building, SBMs, Uranium Hexafluoride (UF_6) Handling Areas, and Full Product Cylinder Storage Pad, and (4) areas on-site, but outside those that are scheduled to be disturbed during plant construction. During construction of the main plant facilities, additional soil samples from disturbed areas next to facility foundations will be taken to characterize foundation soils prior to UF_6 cylinders arriving on-site.

NRC staff reviewed the site characterization data and commitments to collect and analyze additional samples and determined that the data and the commitment to conduct additional sampling and analysis will be adequate to determine a background value for the site, which can be used at decommissioning time to determine whether the site meets the unrestricted release criteria.

To formalize AES's commitment, the staff will impose the following license condition:

> *Prior to the commencement of construction, AES shall collect additional surface soil samples and analyze them for radiological constituents. The site property will be divided into four survey units, and 15 surface soil samples shall be taken per survey unit (i.e., 60 additional soil samples). The sample collections shall be taken from areas that include (1) the detention and retention basins, (2) Full Tails, Full Feed, and Empty Cylinder Storage Pads north of the main facilities, (3) the Technical Services Building, Blending, Sampling and Preparation Building, SBMs, UF_6 Handling Areas, and Full Product Cylinder Storage Pad, and (4) areas on-site, but outside those that are scheduled to be disturbed during plant construction.*

Section 10.1.5, "Decommissioning Design Features," of the SAR states that decommissioning planning begins with ensuring design features are incorporated into the plant's initial design to simplify eventual dismantling and decontamination. The plans are implemented through proper management and health and safety programs. Decommissioning policies address radioactive waste management, physical security, and material control and accounting. Major features incorporated into the facility design that facilitate decontamination and decommissioning include radioactive contamination control, worker exposure and waste volume control, management organization, health and safety, waste management, security and material control, and recordkeeping.

The NRC staff reviewed the applicant's decommissioning design features and design feature implementation plans and determined that they provide reasonable assurance that the plant will be designed to facilitate eventual decommissioning in accordance with NRC regulations and are therefore adequate.

Section 10.1.6, "Decommissioning Process," of the SAR describes the decommissioning methodology to be employed at the EREF. The four SBMs will be shutdown in sequence starting with SBM 1. Termination of SBM 4 operations will mark the end of uranium enrichment operations at the facility. Decommissioning of the remaining plant systems and buildings will begin after SBM 4 operations have been permanently terminated. A conceptual decommissioning schedule is provided in Figure 10.1-1 of the SAR, "Eagle Rock Enrichment Facility – Conceptual Decommissioning Schedule."

The decommissioning process will include performing a radiological survey of the facility immediately prior to the start of decommissioning and preparation of a detailed Decommissioning Plan in accordance with 10 CFR 70.38 and the applicable guidance provided in NUREG-1757. Decommissioning activities will generally include: (1) installation of decontamination facilities, (2) purging of process systems, (3) dismantling and removal of equipment, (4) decontamination and destruction of Confidential and Secret Restricted Data material, (5) sales of salvaged materials, (6) disposal of wastes, and (7) completion of a final radiation survey. Credit will not be taken for any salvage value that might be realized from the sale of potential assets (e.g., recovered materials or decontaminated equipment) during or after decommissioning.

Decommissioning, using the decontamination and dismantlement (DECON) approach, requires residual radioactivity to be reduced below specified levels so the facilities may be released for unrestricted use. The unrestricted release criteria in 10 CFR Part 20.1402 serve as the basis for decontamination costs estimated herein. The intent of decommissioning the facility is to remove all enrichment-related equipment from the buildings such that only the building shells and site infrastructure remain.

The standard decontamination methodology to be used during EREF decommissioning will employ conventional decontamination techniques as described in Section 10.1.6 of the SAR. Decontaminated components may be reused or sold as scrap. All equipment that is to be reused or sold as scrap will be decontaminated to a level at which further use is unrestricted. Materials that cannot be decontaminated will be disposed of in a licensed radioactive waste disposal facility. Contaminated portions of the buildings will be decontaminated as required. When decontamination is complete, all areas and facilities on the site will be surveyed to verify that further decontamination is not required. Decontamination activities will continue until the entire site is demonstrated to be suitable for unrestricted use.

Section 10.1.6.9 of the SAR describes the final radiation survey that will be used to verify proper decontamination to allow the site to be released for unrestricted use. The evaluation of the final radiation survey will be based in part on an initial radiation survey performed prior to initial operation, discussed above, which provides a datum for measurements which determine any increase in levels of radioactivity. The survey procedures and results will be analyzed and shown to be below allowable residual radioactivity limits; otherwise, further decontamination will be performed.

Section 10.1.7, "Decontamination Facilities," of the SAR describes a specialized facility that will be required to accommodate decommissioning and describes a general procedure for decontaminating contaminated plant components, including centrifuges. This specialized facility is needed for optimal handling of the thousands of centrifuges to be decontaminated, along with the UF_6 vacuum pumps and valves. Additionally, a general purpose facility is required for handling the remainder of the various plant components. These facilities are assumed to be installed in existing plant buildings [such as the Centrifuge Assembly Building (CAB)]. The decontamination facility will have four functional areas that include: (1) a disassembly area, (2) a buffer stock area, (3) a decontamination area, and (4) a scrap storage area for cleaned stock. Contaminated plant components will be cut up or dismantled, then processed through the decontamination facilities. Contamination of site structures will be limited to areas in the SBMs, the Blending, Sampling and Preparation Building, the CAB, and TSB, and will be maintained at low levels throughout plant operation by regular cleaning. Although decommissioning operations are planned to be underway while all the activities considered in the integrated safety analysis (ISA) continue to occur in the other portions of the plant, the current ISA has not considered these decommissioning risks. An updated ISA will be performed at a later date, but prior to decommissioning, to incorporate the risks from decommissioning operations on concurrent enrichment operations.

The NRC staff has reviewed the applicant's conceptual decommissioning process and determined that the applicant has considered site-specific decommissioning activities that may be needed in the future, and, therefore, that the applicant's proposed process provides reasonable assurance that decommissioning can be performed in accordance with NRC regulations.

10.3.3 Financial Assurance for Decommissioning

Applicants for licenses under 10 CFR Part 30, 40 and 70 are subject to FA requirements for decommissioning, decontamination and reclamation pursuant to 10 CFR 30.35, 40.36 and 70.25. As part of the application, AES submitted various documents to demonstrate that adequate FA will be provided to decontaminate and decommission the facility and site to unrestricted release criteria. AES intends to provide financial assurance for site and facility decommissioning as well as for the disposition of DU. AES is responsible for the disposal of DU. Section 3113(a) of the USEC Privatization Act, 42 USC § 2297h-11(a), requires the Department of Energy (DOE), upon request of the operator of a uranium enrichment facility licensed by the NRC, to accept DU for disposal for a fee if it is determined that DU is a low level radioactive waste. In 2005, the Commission made that determination. *Louisiana Energy Services, L.P.* (National Enrichment Facility), CLI-05-5, 61 NRC 22, 36 (2005). Therefore, the applicant has assumed that DOE will take title to and possession of DU for disposal. The applicant intends to provide FA for both site and facility decommissioning and the disposition of DU incrementally over time, and thereby accrete the total amount FA provided proportionally to the then current decommissioning cost estimate.

As described in section 10.2.1 of the SAR, the applicant intends to rely on a Letter of Credit (LOC) to provide such FA as set forth in 10 CFR 30.35(f)(2), 10 CFR 40.36(e)(2), and 10 CFR 70.25(f)(2). The applicant provided draft copies of the LOC and standby trust agreement and supporting documentation. The applicant will finalize the financial instruments to be relied on and, prior to receiving any licensed material at the EREF, will provide executed originals of the reviewed financial instruments to the NRC for final review and confirmation. As required by 10 CFR 70.38(e), AES will provide adequate FA continuously until license termination.

Unless otherwise noted, all updates to the DFP and cost estimate will be based on the costs of a third party contractor, will not take credit for any salvage value that might be realized from the sale of potential assets during or after decommissioning, shall be updated to current year dollars -- taking into account applicable changes in costs such as, but not limited to, changes in labor, disposal, and shipping costs; inflation; currency exchange rates; and other site and facility factors -- and shall contain a 25 percent contingency factor. These requirements will be included in a license condition that will be imposed to ensure compliance with AES's incremental funding mechanism, discussed in Section 10.3.3.1.1.

10.3.3.1 Description of Incremental Funding Approach and Approach to Updating the DFP, Cost Estimate, and Financial Instruments

In Sections 1.2.5, 10.2.1 and 10.2.2 of the SAR (AES, 2009a), AES describes its proposed approach to providing FA incrementally over time by updating the decommissioning cost estimate and FA instruments. Specifically, in Section 1.2.5 of the SAR (AES, 2009a), the applicant requested an exemption to 10 CFR 40.36(d) and 10 CFR 70.25(e), which require a certification of FA to be provided in the amount of the cost estimate. As described in Section 10.2.1 of the SAR (AES, 2009a), the applicant intends to build, install and operate the SBMs sequentially over time and thus requests to provide FA on an incremental and forward looking basis, proportional to the then current decontamination and decommissioning liability.

These updates will be made annually on a forward-looking, incremental basis until the facility is at full capacity operation, which would occur in approximately March 2022, according to the AES's schedule. Until such time, AES will provide annual updates to the DFP, cost estimate and FA mechanism for site and facility decommissioning, and the DU disposition cost estimate and its associated FA mechanism. As described in AES's supplemental response to RAI SE-1, after the facility has reached full capacity operation, the cost estimate and FA mechanisms for site and facility decommissioning will be updated at least triennially, and the cost estimate and FA instrument for DU disposal will continue to be updated annually (AES, 2010a).

10.3.3.1.1 Annual, Forward-Looking Updates until Full Capacity Operation of EREF

As stated in AES's response to RAI SE-1 (AES, 2009b) and as further clarified in AES's supplemental response to RAI SE-1 (AES, 2010a), the first facility decommissioning cost estimate will be provided six months prior to the delivery of natural UF_6, [less than 20 kilograms (kg) Uranium (U)] as testing material for the CAB. The facility decommissioning cost estimate will then be updated six months in advance of the first delivery of natural UF_6 (more than 50 kgU) as feed material for the first SBM. Thereafter, it will be updated annually (i.e., six months in advance of each anniversary of the first delivery of natural UF_6 (more than 50 kgU) as feed material for the first SBM).

According to AES's milestones described in Chapter 10 of the SAR (AES, 2009a) and in the response to RAI SE-1 (AES, 2009b), AES intends to install four SBMs and introduce initial feed material in four separate Phases.

- Phase One - The receipt of Natural UF_6 as feed material for the first SBM
- Phase Two - The receipt of Natural UF_6 as feed material for the second SBM
- Phase Three - The receipt of Natural UF_6 as feed material for the third SBM
- Phase Four - The receipt of Natural UF_6 as feed material for the fourth SBM

AES expects Phase One to commence shortly after completing testing at the CAB. Thereafter, AES expects two years to elapse between subsequent Phases. Until the facility is at full capacity operations, as stated in its response to RAI SE-1 (AES, 2009a) and supplemental response to RAI SE-1 (AES, 2010b), AES commits to providing NRC with an updated, forward looking cost estimate for both site and facility decommissioning, and DU disposition on an annual basis and in accordance with the phasing of the first delivery of natural UF_6 as feed material for each SBM, at least six months prior to the described above milestones. The NRC will review each cost estimate. After NRC review and approval, at least 21 days prior to any of the above phases, AES will provide an executed version of the reviewed financial mechanism(s) whose aggregate amount will be equal to at least the approved, updated cost estimate, in accordance with 10 CFR 70.25(e). The specific scope of each update is described later in this section.

At each of the above phases, AES will not receive any additional material as initial feed material for each subsequent SBM not previously in operation until the decommissioning cost estimate is approved and full funding for decommissioning the site and facility and disposition of DU, is provided to and confirmed by NRC. Providing an executed FA mechanism will not permit AES to bring additional material prior to the approval of the DFP and cost estimate. This requirement will be included in a license condition that will be imposed to ensure compliance with AES's proposed incremental funding.

Initial Financial Assurance

AES's initial FA is intended to cover the testing phase of facility. During the testing phase, the facility is not in operations, with respect to enriching uranium. At least six months prior to the receipt of natural UF_6 (less than 20 kgU) as test material, AES will provide an updated decommissioning cost estimate and final copies of the proposed financial instrument(s) for NRC review. The updated cost estimate and associated FA instrument(s) shall be sufficient to cover the decommissioning costs of the: (1) CAB; and (2) any part of the facility not fully decontaminated as approved by NRC (i.e., NRC has determined can be released for unrestricted use). According to Section 1.2.3 of the SAR (AES, 2009a), AES expects to utilize "...[o]ther source materials and by-product materials...for instrument calibration purposes....," and "[t]hese materials will be identified during the design phase, and AES will submit a request to amend the Materials License to incorporate the proposed quantities and types for the sealed and unsealed instrument calibration sources to its possession limits." As stated by AES in its responses to NRC's RAI SE-1 and GI-1 (AES, 2009b), the above referenced update to the decommissioning cost estimate and funding instrument will also provide adequate funding for the decommissioning activities related to these additional materials. After NRC review, and at least 21 days prior to the receipt of test material, AES will provide an executed version of the reviewed financial mechanism(s) whose aggregate amount will be equal to at least the approved, updated decommissioning cost estimate.

Financial Assurance Updates

Following the initial FA submittal and until the facility is at full capacity, AES will provide an updated DFP, cost estimate and financial assurance instrument(s) at least six months in advance of the planned date for receipt of natural UF_6 (greater than 50 kgU) as initial feed material for SBMs One through Four. In addition, after SBM One begins operation, AES will provide annual updates to the DFP, cost estimate, and FA instruments. Each update will be forward-looking and will cover: (1) any part of the facility that would currently be in operation; (2) any part of the facility that has been in operation, or any part of the facility reasonably believed to be contaminated and not fully decontaminated and decommissioned as approved by NRC; (3) all plant areas where licensed material is stored or used; (4) any part of the facility that would be in operation within the next 12 months; and (5) an update to the DU disposal cost estimate and financial assurance mechanism (described in the next paragraph). After NRC review and approval, at least 21 days prior to the receipt of initial feed material to each SBM, AES will provide an executed version of the reviewed financial mechanism(s) -- whose aggregate amount will be equal to at least the approved, updated decommissioning cost estimate – for NRC review and confirmation. According to AES's supplemental response to RAI SE-1 (AES, 2010a), the applicant will also provide an executed version of the reviewed financial mechanism(s) with each annual update to the DFP and cost estimate.

The DU disposal cost estimate and financial mechanism, submitted in connection with the update that encompasses the receipt of initial feed material for SBM One, will cover the disposition of the accumulated DU expected onsite stemming from the first three years of operations. According to AES's supplemental response to RAI SE-1 (AES, 2010a), annually for the next two years, AES will provide an update to its DU disposition cost estimate and financial mechanism in conjunction with the annual updates described in the above paragraph, sufficient to cover the disposition of the accumulated DU expected onsite stemming from the first three years of operations. According to AES's supplemental response to RAI SE-1 (AES, 2010a), after the first two years of operations, AES will provide annual updates to the DU disposition cost estimate, sufficient to cover the accumulated DU onsite and a projection of the amount of DU that would be onsite within the subsequent 12 months. All updates for DU disposal will be based on an updated DOE cost estimate and the total amount provided for DU disposition will not be less than the updated DOE cost estimate. In its supplemental response to RAI SE-1 (AES, 2010a), AES states the accumulated DU onsite quantity will not include any quantity of DU that would have been already delivered offsite for disposition. NRC finds this acceptable, provided that DOE has taken title to and possession of such DU and assumed responsibility for its decommissioning and disposal.

This method of providing incremental and annual updates automatically captures changes in construction schedules. Since each update is forward looking for the next 12 months, if SBM construction schedules are accelerated, the timing and/or scope of the updates would automatically address such changes. Since AES committed to providing annual updates to the cost estimate and financial mechanism(s), should construction schedules be delayed, the cost estimate for site and facility decommissioning and DU disposal will continue to be updated annually.

To ensure compliance with AES's proposed methodology of providing FA on an incremental, forward-looking basis, the staff will impose the following license condition:

The licensee shall provide FA on the following schedule:

a. *The licensee shall provide an updated DFP, updated facility decommissioning cost estimate, and final copies of proposed financial assurance instruments to the NRC for review at least six months prior to the following dates:*

   (1) *planned date for obtaining test material (≤ 20 kg U) for the CAB*

(2) planned date for obtaining feed material (> 50 kgU) for initial production in the first SBM

(3) planned date for obtaining feed material (> 50 kgU) for initial production in the second SBM

(4) planned date for obtaining feed material (> 50 kgU) for initial production in the third SBM

(5) planned date for obtaining feed material (> 50 kgU) for initial production in the fourth SBM

The updates shall be forward-looking through the 12-month period beginning on the applicable date listed above. For each update, the licensee shall provide final executed copies of the NRC-reviewed financial assurance instruments to NRC at least 21 days prior to receipt of test material or receipt of feed material for initial production in an SBM.

b. After the first SBM begins operations, and until the plant reaches full capacity, the licensee shall, on an annual basis, provide an updated DFP, an updated facility decommissioning cost estimate, and final copies of proposed financial assurance instruments to NRC for review. These annual updates shall be provided six months prior to the anniversary date of obtaining feed material for initial production in the first SBM, and shall be forward-looking through the 12-month period beginning on the anniversary date. For each annual update, the licensee shall provide final executed copies of the NRC-reviewed financial assurance instruments to NRC at least 21 days prior to the anniversary date.

If the licensee provides an annual update at least six months prior to the planned date for obtaining feed material for initial production in the second, third, or fourth SBM, that annual update may also serve as the update required in paragraph (a) for that date.

c. The updated DFPs, updated cost estimates, and financial assurance instruments described in paragraphs (a) and (b) shall include full funding for decontamination and decommissioning of: (1) any part of the facility currently in operation; (2) any part of the facility that has been in operation, or any other part of the site or facility reasonably believed to be contaminated, that has not been fully decontaminated and decommissioned as approved by NRC (including the CAB); (3) all plant areas where licensed material is stored or used; and (4) any part of the facility (including SBMs) expected to be in operation by the end of the applicable forward-looking 12-month period in paragraph (a) or (b).

d. The licensee shall provide an initial depleted uranium (DU) disposition cost estimate and final copies of proposed financial assurance instruments for DU disposition in conjunction with the updated DFP, updated facility decommissioning cost estimate, and financial assurance instruments that will be submitted at least six months prior to obtaining feed material for initial production in the first SBM. The DU disposition cost estimate and proposed financial *assurance instruments shall include full funding to cover disposition of the first three years of DU tails generation. The DU disposition cost estimate shall include an update to the U.S. Department of Energy (DOE) DU disposition cost estimate. The total amount funded for DU disposition shall not be less than the updated DOE cost estimate.*

For the initial DU disposition cost estimate, the licensee shall provide final executed copies of the NRC-reviewed financial assurance instruments for DU disposition to NRC at least 21 days prior to the receipt of feed material for the first SBM.

e. *The licensee shall provide updates to the DU disposition cost estimate and financial assurance instruments for DU disposition as described below:*

   (1) *During the first two years of operation, the licensee shall provide updated DU disposition cost estimates and final copies of proposed financial assurance instruments for DU disposition in conjunction with the updates required in paragraphs (a) and (b). The updated cost estimates shall provide full funding to cover disposition of the first three years of DU tails generation.*

   (2) *After the first two years of operation and until the facility reaches full capacity, the licensee shall provide updated DU disposition cost estimates and final copies of proposed financial assurance instruments for DU disposition in conjunction with the updates required in paragraphs (a) and (b). The updated DU disposition cost estimates shall provide full funding to cover disposition of all DU stored onsite and all DU expected to be generated by the end of the applicable forward-looking 12-month period in paragraph (a) or (b).*

   (3) *After the plant reaches full capacity, the licensee shall continue to provide annual updates to the DU disposition cost estimate, along with revised financial assurance instruments. These annual updates shall include full funding to cover disposition of all DU stored onsite and all DU expected to be generated by the end of the 12-month period beginning on the anniversary date of obtaining feed material for initial production in the first SBM. The annual updates to the DU disposition cost estimate and final copies of proposed financial assurance instruments shall be provided to NRC for review six months prior to the anniversary date.*

The licensee may exclude from the updated DU disposition cost estimates any DU that the DOE has taken title to and possession of pursuant to Section 3113 of the USEC Privatization Act. All updates to the DU disposition cost estimates shall include an update to the DOE cost estimate for DU disposition. The total amount funded for DU disposition shall not be less than the updated DOE cost estimate.

For DU disposition cost estimate updates, the licensee shall provide final executed copies of the NRC-reviewed financial assurance instruments for DU *disposition to NRC at least 21 days prior to the receipt of feed material for an SBM, or the anniversary date of obtaining feed material for initial production in the first SBM, as applicable.*

f. *If the construction and/or operation of any SBM is delayed or cancelled, the licensee is not relieved of its commitment to provide updated DFP, facility decommissioning cost estimates, DU disposition cost estimates, and final copies of proposed financial assurance instruments to NRC as described in paragraphs (a)-(e).*

g. *When an update to the DFP, cost estimates for facility decommissioning and DU disposition, and financial assurance instruments encompasses the first delivery of natural uranium hexafluoride (> 50 kgU) as feed material to an SBM not previously in operation, the licensee shall not receive such initial feed material until the NRC reviews the updated DFP and cost estimates and confirms the executed financial assurance instrument(s).*

h. *All updates to the DFP, cost estimates for facility decommissioning and DU disposition, and financial assurance instruments, shall be updated to current year United States dollars and shall encompass all current cost data, taking into account changes in inflation, foreign currency exchange rates, possession limits, licensed material, labor rates, disposal and shipping rates, and site and facility factors. All costs shall be based on the costs of a third party contractor and shall not take credit for any salvage value that might be realized from the sale of potential assets during or after decommissioning. All costs (including those for DU disposition) shall include a contingency factor of at least 25 percent.*

10.3.3.1.2 Updates for Full Capacity Operation of EREF

After the facility has reached full production output, updates to the cost estimate for DU disposal follow a different schedule than updates to the cost estimate for site and facility decommissioning. Subsequent updates to the facility decommissioning cost estimate and revised financial instruments for facility decommissioning will be provided at least every three years pursuant to 10 CFR 30.35(e), 40.36(d), and 70.25(e). The updated facility decommissioning cost estimate updates will be accompanied with final copies of proposed financial instruments whose aggregate amount is sufficient to cover the updated decommissioning cost estimate and cost estimate for DU disposal.

Updates to the cost estimate and revised financial instruments for the disposition of DU will continue to be provided on an annual, forward-looking basis. This annual update to the DU disposition cost estimate will be based on an updated DOE cost estimate and will be sufficient to cover the disposition of accumulated DU onsite and the projected amount of DU that would be generated onsite within the subsequent 12 months. The updated cost estimate for the disposition of DU will be accompanied with final copies of proposed financial instruments whose aggregate amount is sufficient to cover the approved decommissioning cost estimate and updated cost estimate for DU disposal.

Under AES's requested approach to provide financial assurance on an incremental and forward looking basis, the staff finds that AES will maintain adequate financial assurance at least equal to its most recently approved cost estimate for decommissioning. The staff finds that the described schedule and methodology for updating the DFP, cost estimate and FA mechanisms is satisfactory , will provide updates to the cost estimate at least triennially consistent with 10 CFR 30.35(e), 40.36(d), and 70.25(e), and therefore the staff finds that AES's approach will not endanger life or property.

10.3.3.2 Discussion on the Adequacy of the Cost Estimate

A detailed review was conducted of AES's decommissioning cost estimate. The decommissioning cost estimate was developed by the centrifuge supplier, Enrichment Technology Company Limited (ETC). In its supplemental response to RAI D-4 (AES, 2010a), AES states that ETC has experience in Europe with regard to dismantling, decontaminating, and disposing of parts of earlier generations of centrifuges. Additionally, AES stated that ETC "undertook studies to look at how the modern centrifuges as using in the EREF would be decommissioned and declassified at the end of their lifetime" (AES, 2010a). AES further states that this "process flow...was fully tested [by ETC] using both intact and crashed centrifuges of the same type as used in EREF" (AES, 2010a). AES stated that "a layout and process flow for a declassification facility has been developed and costed...[which] included the required manning levels and throughputs; these have been built into the decommissioning model" (AES, 2010a). Furthermore, AES stated that ETC updates its "model...on a regular basis for each plant and [is] updated to account for any changes in disposal costs or additional experience gained" (AES, 2010a). Based on AES's description, the staff finds that ETC's experience provides an adequate basis for the cost estimate.

AES's decommissioning cost estimate does not include provisions for restoration of contaminated areas on the facility grounds. In its response to RAI D-8 (AES, 2009b) and supplemental response to RAI D-8 (AES, 2010b), AES stated that based on European experience, "enrichment facilities are expected to operate cleanly because of plant contamination controls and environmental monitoring and should result in no ground contamination." Additionally, if surface contamination levels exceed the levels described in section 4.8.5.1 of the SAR, clean-up of the contamination is initiated within 24 hours (AES, 2009a). Given the basis supplied by AES, the staff finds AES's basis acceptable. However, the staff is including a provision in the license condition described in Section 10.3.3.1.1 requiring AES to provide FA for contamination that is not cleaned-up.

Relying on its application, the initial financial assurance provided by AES is estimated to be $12,782,500. Since the aforementioned dollar amount may change due to changes in inflation, foreign currency exchange rates, labor rates, disposal rates, possession limits, and other cost factors, and because AES may decide to utilize other source and/or byproduct materials for testing purposes, the decommissioning cost estimate and its associated funding instrument will be updated as described in Section 10.3.3.1.1 of the SAR (AES, 2009a), Initial Financial Assurance. At the time of CAB operation, no SBM will be in operation.

After providing initial FA, AES intends to commence Phase One of operations. According to section 10.2.2, and Table 10.1-14 of the SAR (AES, 2009a), as well as the responses to RAI SE-1 (AES, 2009b), the estimated cost to decommission all applicable site and facility areas covered by Phase One, excluding DU disposal, is $131,098,750, assuming a third party contractor completes the decommissioning and including a 25 percent contingency factor. AES estimates that the cost to decommission the entire site and facility (all four phases), excluding DU disposal, is $445,427,500. Therefore, the amount of FA provided in connection with Phase One is approximately 29% of the total cost of site and facility decommissioning. According to sections 10.2.2 and 10.3 and Table 10.3-1 of the SAR (AES, 2009a), AES estimates that it will generate 11,452 MT DU during the first three years of operations, and the estimated cost of DU disposal is $7.66 per kg. AES states that an amount of $109.6 million, based on a DOE cost estimate and including a 25 percent contingency, is sufficient to cover the cost of disposing the DU generation stemming from the first three years of operations. Therefore, the Phase One's total estimated cost of site and facility decommissioning, and DU disposition is $240,751,000. As stated previously, AES will be required to appropriately update the cost estimate to current year United States Dollars and include all changes to relevant cost factors.

The staff finds the cost estimate consistent with NUREG-1757, Volume 3, based on costs of a third party contractor, does not take credit for any salvage value that might be realized from the sale of potential assets during or after decommissioning, and includes a 25 percent contingency factor. Additional key assumptions of AES's cost estimate are that: (a) decommissioning activities will be performed in accordance with current day regulatory requirements; (b) decommissioning costs and DU disposition costs are presented in 2007 dollars; and (c) some costs were provided in Euros (€), and a rate of 0.714€ to $1 was used to convert 2007 € to 2007 $. As indicated previously, AES's estimated cost to decommission the entire facility, excluding DU disposition, is $445,427,500 (2007 U.S. dollars). The CAB, and subsequently the first SBM, are not expected to require FA until 2012, therefore these figure are not updated to current year United States Dollars, since AES will be required to update them appropriately prior to receipt of any licensed material.

According to Table 10.1-14 and Section 10.3 of the SAR (AES, 2009a), based on a DOE cost estimate, AES estimates total DU disposition costs, excluding a 25% contingency, to be $2.461 billion. Thus, AES estimates the total site and facility decommissioning and DU disposition cost, including a 25% contingency factor, to be $3,523,436,000. Relying on AES's proposed methodology of providing FA on an incremental basis, this total estimated cost of DU disposal, would not be incurred immediately, but over the life of the facility. Therefore, the cost to decommission the facility would not be $3,523,436,000 immediately at the commencement of operations.

Based on its review, the staff finds the cost estimate to decommission the site and facility, and the estimated cost for DU disposition, satisfies the requirements of 10 CFR 30.35(e) 40.36(d), 70.25(e), is consistent with NUREG-1757, Volume 3 and therefore is acceptable.

10.3.3.3 Review of Financial Instruments

AES provided NRC with a draft LOC and Standby Trust Agreement for review. The purpose of these financial instruments is to provide reasonable assurance that funds will be available when needed, to decommission and decontaminate the licensed site and facility to unrestricted release criteria, as set forth in 10 CFR 20.1402. The submitted draft instruments are not the proposed final versions. As stated previously, at least six months prior to the receipt of licensed material for the CAB, AES will provide an updated facility decommissioning cost estimate and final copies of the proposed financial instruments. At least 21 days prior to the receipt of licensed material, AES will provide executed copies of the NRC-reviewed financial instruments for final confirmation before the applicant receives licensed material at the site and facility. The specific regulatory requirements with regard to the LOC are set forth in 10 CFR 30.35(f)(2)(i) to (iii), 40.36(e)(2)(i) to (iii) and 70.25(f)(2)(i) to (iii), which state:

(i) The [LOC] must be open-ended or, if written for a specified term, such as five years, must be renewed automatically unless 90 days or more prior to the renewal date, the issurer notifies the [NRC], the beneficiary, and the licensee of its intention not to renew. The [LOC] must also provide that the full face amount be paid to the beneficiary automatically prior to the expiration without proof of forfeiture if the licensee fails to provide a replacement acceptable to the [NRC] within 30 days after receipt of notification of cancellation.

(ii) The LOC must be payable to a trust established for decommissioning costs. The trustee and trust must be acceptable to the [NRC]. An acceptable trustee includes an appropriate State or Federal government agency or an entity which has the authority to act as a trustee and whose trust operations are regulated and examined by a Federal or State agency.

(iii) The [LOC] must remain in effect until the [NRC] has terminated the license.

The staff has conducted a review of AES's draft LOC. The proposed LOC language is consistent with the guidance in NUREG-1757, Volume 3, Appendix A.10. However, at this time, the language does not satisfy the NRC regulations in 10 CFR 30.35(f)(2)(i), 40.36(e)(2)(i) and 70.25(f)(2)(i), because, among other items, no financial institution is specified at this time and the aggregate dollar amount of the instrument is not indicated. As stated above, AES will finalize this instrument and financial instruments are not required at this time, consistent with 10 CFR 70.25(b)(2). Once AES finalizes the specific instrument and submits it for review with an updated cost estimate, the staff will review it for compliance with regulatory requirements at that time.

The staff has conducted a review of AES's draft Standby Trust Agreement. The specific regulatory requirements for standby trust agreements are set forth in 10 CFR 30.35(f)(2)(ii), 40.36(e)(2)(ii) and 70.25(f)(2)(ii). The language of the Standby Trust Agreement is consistent with guidance contained in NUREG-1757 Volume 3, Appendix A.17. However, at this time, the language does not satisfy the regulations because, among other items, no trustee is specified. As stated above, AES will finalize this instrument and financial instruments are not required at this time, consistent with 10 CFR 70.25(b)(2). Once AES finalizes the specific instrument and submits it for review with an updated cost estimate, the staff will review it for compliance with te regulatory requirements at that time.

On the basis of the NRC staff's above review, the applicant has demonstrated that (1) the cost estimate for both the decommissioning of the site and facility, and disposal of DU is adequate; (2) the language of the draft FA instruments is acceptable; (3) and the proposed incremental funding of decommissioning FA is adequate to maintain appropriate and reasonable levels of FA for the operational phases of the project. Under the above incremental funding approach, AES will maintain an appropriate amount of decommissioning financial assurance proportional to its decommissioning and disposal obligations at any point in time. Therefore, the proposed approach will not endanger human health and safety or property.

10.4 Evaluation Findings

The NRC staff has evaluated the applicant's plans and FA for decommissioning in accordance with NUREG-1757, "Consolidated NMSS Decommissioning Guidance," September 2003. On the basis of this evaluation, the NRC staff finds the applicant's plans and FA for decommissioning comply with the NRC's regulations set forth in 10 CFR 30.35, 40.36, and 70.25, and provide reasonable assurance of protection for workers, the public, and the environment.

To ensure compliance with the site characterization commitments, the staff will impose the following license condition:

> *Prior to the commencement of construction, AES shall collect additional surface soil samples and analyze them for radiological constituents. The site property will be divided into four survey units, and 15 surface soil samples shall be taken per survey unit (i.e., 60 additional soil samples). The sample collections shall be taken from areas that include (1) the detention and retention basins, (2) Full Tails, Full Feed, and Empty Cylinder Storage Pads north of the main facilities, (3) the Technical Services Building, Blending, Sampling and Preparation Building, SBMs, UF_6 Handling Areas, and Full Product Cylinder Storage Pad, and (4) areas on-site, but outside those that are scheduled to be disturbed during plant construction. During construction of the main plant facilities, additional soil samples from disturbed areas next to facility foundations shall be taken to characterize foundation soils prior to UF_6 cylinders arriving on-site.*

To ensure compliance with the proposed methodology of providing FA on an incremental, forward-looking basis, the staff will also impose the license condition described in Section 10.3.3.1.1 of this SER.

10.5 References

(NRC, 2002a) U.S. Nuclear Regulatory Commission, NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," March 2002.

(NRC, 2003) U.S. Nuclear Regulatory Commission, NUREG-1757, "Consolidated NMSS Decommissioning Guidance," September 2003.

(NRC, 2009) U.S. Nuclear Regulatory Commission, "Request for Additional Information," August 26, 2009.

(AES, 2009a) AREVA Enrichment Services LLC, Eagle Rock Enrichment Facility, "Revision 1 to License Application for the Eagle Rock Enrichment Facility"; April 23, 2009.

(AES, 2009b) AREVA Enrichment Services LLC, Eagle Rock Enrichment Facility; Response to Requests for Additional Information - AREVA Enrichment Services LLC License Application for the Eagle Rock Enrichment Facility; September 28, 2009.

(AES, 2010a) AREVA Enrichment Services LLC, Enclosure to email from Jim Kay, AES, to Breeda Reilly, NRC, "Decommissioning Funding Plan and Financial Assurance RAI SE-1, D-4, and D-8," May 25, 2010.

(AES, 2010a) AREVA Enrichment Services LLC, Enclosure to email from Jim Kay, AES, to Breeda Reilly, NRC, "Response to RAI D-8," June 18, 2010.

CHAPTER 11.0 MANAGEMENT MEASURES

Management measures are functions performed by the licensee, generally on a continuing basis, that are applied to items relied on for safety, to ensure the items are available and reliable to perform their functions when needed. Management measures are required to assure compliance with performance requirements of 10 CFR 70.61; 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 Title 10 of the *Code of Federal Regulations* (CFR) Part 70 definition of management measures, 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 AREVA Enrichment Services, LLC (AES) Eagle Rock Enrichment Facility's (EREF's) application provided conclusive information to ensure that the management measures applied to IROFS, as documented in the ISA Summary, ensure that the IROFS will be available and reliable to perform their functions when needed, to comply with the performance requirements of 10 CFR 70.61. This review also determines whether the measures are applied to the IROFS in a graded manner commensurate with the IROFS importance to safety.

In implementing the QA Program Description (QAPD), AES will apply management measures commensurate to the reduction of the risk attributable to the items. The applicant uses QA Level designations -- QA Level 1, 2 and 3 -- to describe their graded approach. As described in the license application, management measures will be applied to QA Level 1 and QA Level 2 items (AES, 2009a). All IROFS will be designated as QA Level 1 or QA Level 2 items. As described in the QAPD, QA Level 1 items and activities include those items and activities whose failure or malfunction could directly result in a condition that adversely affects public, worker and the environment as described in 10 CFR 70.61. In addition, the failure of a single QA Level 1 item could result in a high or intermediate consequence. QA Level 2 items and activities are described in the QAPD as those items and activities whose failure or malfunction could indirectly result in a condition that adversely affects public, worker and the environment as described in 10 CFR 70.61 (CFR, 2008a). The failure of a QA Level 2 item, in conjunction with the failure of an additional item, could result in a high or intermediate consequence. All building and structure IROFS associated with credible external events are QA Level 2. QA Level 2 items and activities also include those attributes of items and activities that could interact with IROFS due to a seismic event, and result in high or intermediate consequences as described in 10 CFR 70.61.

The QAPD was reviewed by the staff and accepted on April 8, 2010 (NRC, 2010). Based on that review, documented in the accompanying Staff Evaluation Report, the staff found the program acceptable for application to the design, construction, operation, including maintenance and modification, and decommissioning of the proposed EREF (NRC, 2010). The Staff Evaluation Report discusses the staff review of the QAPD which was based on NUREG-1520 (NRC, 2002) and documents the staff's conclusion that the QAPD adequately describes the application of other QA elements and has adequately established other QA elements as part of Management Measures as required by 10 CFR Part 70.62(d) (NRC, 2010).

11.1 Regulatory Requirements

The requirements for fuel cycle facility management measures are specified in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material."

1. 10 CFR 70.4 states that management measures include: (1) 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.

2. 10 CFR 70.62(a)(3) states that records must be kept for all IROFS failures, describes required data to be reported, and sets time requirements for updating the records.

3. 10 CFR 70.62(d) requires an applicant or licensee to establish management measures to ensure that engineered or administrative controls and control systems that are designated as IROFS are designed, implemented, and maintained, as necessary, to ensure they are available and reliable to perform their function when needed, to comply with the performance requirements of 10 CFR 70.61.

4. 10 CFR 19.12 states requirements for instructions to workers that are applicable to personnel training and qualifications.

5. 10 CFR 70.64(a)(1) requires the design of new applications or new processes at existing facilities to be developed and implemented in accordance with management measures, to provide adequate assurance that items relied on for safety will be available and reliable to perform their function when needed. Appropriate records of these items must be maintained by or under the control of the licensee throughout the life of the facility.

6. 10 CFR 70.22(a)(8) states requirements for license applications to address proposed procedures to protect health and minimize danger to life and property.

7. 10 CFR 70.72 requires a licensee to establish a CM program, documented in written procedures, to evaluate, implement, and track each change to the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel.

8. 10 CFR 70.74 states requirements for incident investigation and reporting.

11.2 Regulatory Acceptance Criteria

The acceptance criteria for the U.S. Nuclear Regulatory Commission's (NRC's) review of the AES EREF management measures program are contained in Section 11.4.3 of NUREG-1520 (NRC, 2002).

11.3 <u>Staff Review And Analysis</u>

11.3.1 Configuration Management (CM)

NRC staff has reviewed AES's CM function in accordance with the regulatory acceptance criteria of Section 11.4.3.1 of NUREG-1520 (NRC, 2002).

AES's CM program is implemented through the QA program requirements as described in EREF QAPD (see Section 11.4.8 of this SER for a discussion of the QAPD) and associated procedures. The AES President has the overall responsibility for the establishment and implementation of the QA program. The CM program is described in Section 11.1 of the Safety Analysis Report (SAR) (AES, 2009a).

11.3.1.1 CM Policy

As described in Section 11.1.1of the SAR (AES, 2009a), through the CM system, AES will control any documentation that can create changes to onsite, structures, processes, systems, equipment, components, computer programs, and activities of personnel—including the ISA. The Engineering Organization will be responsible to maintain the CM program as the project goes from design and construction to the operational phase. The Engineering Organization will have lead engineers in charge of each discipline that will also be responsible for the interdisciplinary reviews for design changes. After issuance of the license, the Engineering Manager will be responsible for design of and modification to facility SSCs.

Section 11.1.1 of the SAR (AES, 2009a) also explains that design changes will be reviewed in accordance with formal procedures including, as a minimum, a review for ISA impacts. Changes to the facility or to activities of personnel will be evaluated in accordance with the requirements of 10 CFR 70.72, as applicable. In addition, for changes that involve or could affect the uranium onsite, a Nuclear Criticality Safety (NCS) evaluation or analysis, as applicable, will be prepared and approved. Each change or modification will also be evaluated and documented for radiation exposure to minimize worker exposures and to maintain the facility within the as low as reasonably achievable (ALARA) program. After completion of any modification, the Engineering Manager—or designee— ensures that the applicable testing has been completed to the affected system, that the system operates correctly, and that the modification and documentation are complete.

The scope of the CM program will include all the QA Level 1 and QA Level 2 items and activities. The scope of QA Level 1 and QA Level 2, as described in Section 11.1.1.1 of the SAR (AES, 2009a), includes documents that expand through design, construction, initial startup and operations, including the appropriate documentation for the processes. During design, documents such as calculations, safety analyses, design criteria, engineering drawings, system descriptions, technical documents, and specifications will be maintained in CM when initially approved. During construction, documents such as vendor data, test data, inspection data, and applicable procedures will be included. Drawings and specifications prepared and issued for procurement, fabrication or construction of QA Level 1 and QL Level 2 items and activities will be included in CM. The implementing documents that support the CM will be controlled within the document control system to ensure that only reviewed and approved documentation is used.

The applicant describes the functional interfaces of the CM program with other management measures in Section 11.1.1.2 of the SAR (AES, 2009a); those management measures include QA, records management, maintenance, training and qualifications, incident investigations, audits and assessments, and procedures. Three areas of special note are QA, maintenance, and records management. The QA program specifically establishes the framework for the CM and other management measures. Maintenance will be established as part of the design basis and controlled under the CM program. Records management will provide evidence on the conduct of the activities associated with CM and other management measures.

In Section 11.1.1.3 of the SAR (AES, 2009a), the application states that the objectives of CM will be to ensure design and operation within the design basis of QA Level 1 and QA Level 2 items and activities. Furthermore, the application states that the objectives will be met by identifying and controlling the preparation and review of documentation and changes of QA Level 1 and QA Level 2 items. The physical configuration of the facility will be maintained consistent with the approved design.

The applicant provides a description of the CM activities in Section 11.1.1.4 of the SAR (AES, 2009a). Design activities included in the CM program will be conducted through a systematic process of preparation, review and approval. The process will ensure consistency between the design and design bases of the QA Level 1 and QA Level 2 items and activities during the design and construction phases. In addition, it will include the activities for QA Level 1 and QA Level 2 items during operations to ensure the activities are within the limits and constraints established in the ISA—and that changes to the facility are controlled in accordance with 10 CFR 70.72. Finally, the CM program will include training records of personnel conducting activities associated with QA Level 1 and QA Level 2 items and activities to ensure that only qualified personnel are performing the work.

As described in Section 11.1.1.5 of the SAR (AES, 2009a), the interfaces between AES and contractors or among contractors performing quality-related activities will be documented. AES and contractors will have the responsibility to identify quality problems and elevate their concerns to the appropriate management if there is disagreement among the parties.

11.3.1.2 Design Requirements

The design requirements will be established and maintained by the Engineering Organization during design, construction and operations. Section 11.1.2 of the SAR (AES, 2009a) states that design requirements will be documented in a design requirements document that provides a hierarchical distribution of the requirements through design basis documents. Design documents associated with QA Level 1 and QA Level 2 will be based on design control provisions and associated procedural controls over design to establish and maintain the technical baseline (AES, 2009a). The design documents will be subject to interdisciplinary review and design verification, as applicable.

Qualified individuals will prepare design documents ensuring that the appropriate codes, standards, and licensing commitments are included (AES, 2009a). Any deviations or changes will be identified in the design documentation package. The design documentation package will be reviewed by an additional qualified individual for concept and conformity to design inputs emphasizing conformance with the applicable codes, standards and licensing commitments (AES, 2009a). The individual performing the review will be independent and will have the authority to delay the approval process until resolution of any issues. The Engineering Manager will document the review, and the manager having the responsibility for the design function will approve the document. The QA Manager will audit the design control process through augmented audit teams.

The design verification will be performed by qualified individuals other that those who performed the design (AES, 2009a). Any verification performed by supervisors of the individuals doing the design will be documented and approved by the supervisor's management. Any changes to QA Level 1 and QA Level 2 design documents will be reviewed, checked, and approved commensurate with the original approval requirements to ensure consistency with the design bases (AES, 2009a).

The responsible engineer will send approved design documents to the document control center for distribution (AES, 2009a). When required, the document control center will maintain the required receipt verification in the file.

The design interface will be maintained by communication among the appropriate personnel (AES, 2009a). During the operational phase, any design changes will be provided to pertinent personnel to ensure the correct performance of their duties.

As described in Section 11.1.2.1 of the SAR (AES, 2009a), EREF QAPD requires procedures to give specifications for the work performed. The procedures, instructions and drawings will incorporate the acceptance criteria established by design. Procedures will also be reviewed to ensure that they are maintained up-to-date with facility configuration and regulatory requirements (AES, 2009a).

11.3.1.3 Document Control

As described in Section 11.1.3 of the SAR (AES, 2009a), procedures will be established to control the preparation and issuance of documents. Measures will ensure that documents, including revisions, are adequately reviewed, approved and released for use by authorized personnel (AES, 2009a). The document control procedures will establish the distribution requirements and controls to ensure that documents are transmitted and received in a timely manner at the appropriate locations. The controlled copies will be distributed to the persons performing the activity for their use.

The applicant will use an electronic document management system to file project records and to make available the official (controlled) copy of the current documents. The system will index the controlled documents through unique numbering of the documents, including revision numbers. Personnel will be trained to use the system to retrieve controlled documents. Superseded or cancelled documents will be appropriately labeled and maintained as records.

Documents that are relevant and relied on for safety through the document control and records management procedures will be captured as described in Section 11.1.3 of the application (AES, 2009a). Some examples of these documents include design requirements, design bases, ISA of IROFS, NCS analyses, NCS evaluations, as-built drawings, specifications, procedures that are IROFS, procedures involving training, QA/Quality Control (QC) documentation, maintenance, audits and assessments reports, emergency operating procedures, emergency response plans, assessment reports and engineering documents.

11.3.1.4 Change Control

Section 11.1.4 of the SAR (AES, 2009a) describes the change control process and states that procedures will be used to control changes to the technical baseline including an appropriate level of technical, management, and safety review and approval prior to implementation. Changes will be controlled throughout the design, construction, and operations phases to maintain consistency among the design requirements, physical configuration and facility documentation. The ISA and other documents affected by design bases of QA Level 1 and QA Level 2 items and activities including the design requirement documents and basis of design documents, will be systematically reviewed and modified to reflect design and operational changes from an established safety basis prior to implementation.

As described in Section 11.1.4.1 of the SAR (AES, 2009a) during the design phase, a systematic interdisciplinary review will ensure that: (1) design changes do not impact the ISA; (2) if any changes impact the ISA, they are accounted for in subsequent ISA changes; or (3) changes are not approved or implemented before review. Potential changes that reduce the level of commitment or margin of safety in the design bases of QA Level 1 and QA Level 2 items and activities will be submitted to the NRC for review and approval prior to implementation of the change.

As described in Section 11.1.4.2, of the SAR (AES, 2009a) during the construction phase, any changes to documents issued for construction, fabrication and procurement will be documented, reviewed, approved, and posted against each affected design document (AES, 2009a). The changes will continue to be evaluated against the approved design bases of QA Level 1 and QA Level 2 items and the ISA. Upon issuance of the license, the applicant is mandated by the 10 CFR 70.72 to implement a change control process that includes the reporting of changes made without prior NRC approval and submit a license amendment for changes that require the NRC's approval before implementation (AES, 2009a).

Section 11.1.4.3 of the SAR (AES, 2009a) states that during the operations phase, changes will be documented reviewed and approved prior to implementation. For changes made in accordance with 10 CFR 70.72, the applicant will have measures in place to ensure that changes to the onsite documentation are made promptly to avoid inadvertent access by responsible facility personnel to outdated information that may affect the performance of their duties. The applicant will evaluate each change or modification to the facility for required changes to procedures, personnel training, testing program, or regulatory documents such as the ISA, ISA Summary, SAR, QA Program, Environmental Report, Physical Security Plan, Emergency Plan, Fundamental Nuclear Material Control Plan, and Standard Practice and Procedure Plan for the Protection of Classified Matter. The applicant states that measures will be established to ensure that the quality of the facility's SSCs is not compromised by planned changes. The modification process will be described in administrative procedures that are approved by the Engineering and QA Managers. The procedures will include the requirements that must be met to implement the modification and the requirements for initiating, approving, monitoring, designing, verifying and documenting the modification. The modifications will be evaluated to ensure consistency among the facility's procedure, personnel training, testing program, and regulatory documents. Other areas that the applicant will consider during the evaluation of changes or modifications are: radiation exposure, modification cost, lessons learned, QA aspects, potential operability or maintainability concerns, constructability concerns, post-modification testing requirements, environmental considerations, and human factors. After the completion of the modification or change, the system will be tested and personnel are trained to ensure correct operations. When the system becomes operational, all the required documentation will be distributed to operations and maintenance including formal notice to all appropriate managers.

11.3.1.5 Assessments

As described in Section 11.1.5, of the SAR (AES, 2009a) periodic assessments of the CM program will be conducted to determine the program effectiveness and to correct deficiencies. Document assessments and system walkdowns will be planned, conducted, and documented in accordance with the facility's audit and assessment program as described in Section 18 of EREF QAPD.

11.3.1.6 Evaluation Findings

As described in the preceding sections of this SER, the staff reviewed the applicant's CM function following the guidance in Section 11.4.3.1 of NUREG-1520 (NRC, 2002). Based on this review, the staff found that the applicant described in Section 11.1 of the SAR (AES, 2009a) the following elements of the CM function: the CM program (Section 11.1.1 of the SAR), design requirements (Section 11.1.2 of the SAR), document control (Section 11.1.3 of the SAR), change control (Section 11.1.4 of the SAR), and assessments (Section 11.1.5 of the SAR). Also, the CM program is described in Section 11.1.1 of the SAR in terms of its policy, objectives, activities, scope of structures systems and components, organizational structure and its interfaces. The application described how design requirements and associated design bases are established and are maintained through control of the design process. Technical management review is also described including approval functions. The applicant provided a description of the functional interfaces of CM with other management measures that addresses each management measure individually.

In Section 11.1.2 of the SAR (AES, 2009a), the application described a design process leading to drawings and other statements of requirements that proceeds logically from the design basis. The applicant also described how design requirements and associated design bases are established and are maintained through design control provisions. The technical management review and approval functions are adequately described.

In Section 11.1.3 of the SAR (AES, 2009a), the applicant described an acceptable method to create and control documents that are relied on for safety. These documents will include design requirements, design basis, ISAs, NCS analyses, NCS evaluations, as-built drawings, specifications, all procedures that are IROFS, procedures involving training, QA/QC documentation, maintenance, audits and assessments, emergency operating procedures, emergency response plans, system modification documents, assessment reports, and others that the applicant deems part of CM.

In Section 11.1.4 of the SAR (AES, 2009a), the applicant described how the CM function will maintain strict consistency among the design requirements, the physical configuration, and the facility documentation during design, construction and operations. This section described how changes to the technical baseline are controlled through procedures to ensure that each change is evaluated, implemented, and tracked. Sections 11.1.4.2 and 11.1.4.3 of the SAR (AES, 2009a) described an acceptable process for providing reasonable assurance that the ISA is systematically reviewed and modified to reflect design or operational changes from an established safety basis and that all documents outside the ISA that are affected by safety-basis changes are properly modified, authoritatively approved, and made available to personnel. The applicant also described the documentation process following changes made to ensure that the change process fully implements the provisions of 10 CFR 70.72.

The applicant described in Section 11.1.5 of the SAR (AES, 2009a) the assessments of the CM program. The section confirms that periodic assessments of the CM function will be conducted to determine the program's effectiveness and to correct deficiencies. The applicant indicated in Section 11.1.5 of the SAR that such assessments will be planned and conducted in accordance with Section 18 of the EREF QAPD.

Thus, the CM program described by the applicant in Section 11.1 and its subsections is consistent with the guidance provided in Section 11.4.3.1 of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

11.3.2 Maintenance

The NRC staff has reviewed the maintenance program following the regulatory acceptance criteria in Section 11.4.3.2 of NUREG-1520 (NRC, 2002).

The applicant describes the maintenance and functional testing programs for the operations phase of the facility. The maintenance organization will be responsible for planning, scheduling, tracking, and maintaining the records for maintenance activities.

As described in Section 11.2 of the SAR (AES, 2009a), the methods and practices that will be applied, to the corrective, preventive, and functional tests maintenance elements will include, as applicable: (a) part lists; (b) as-built or redlined drawings; (c) a notification to the operations function before conducting repairs and removing an IROFS form service; (d) radiation work permits; (e) replacement with like-kind parts and the control of new or replacement parts to ensure compliance with 10 CFR Part 21, (f) compensatory measures while performing work on IROFS; (g) procedural control of removal of components from service for maintenance and for return to service; (h) ensuring safe operations during the removal of IROFS from service; and (i) notification to operations personnel that repairs have been completed (AES, 2009a).

Written procedures for the performance of maintenance will be prepared and will include steps (a) through (i) as listed above. Maintenance procedures will include reviews by the various safety disciplines, including criticality, fire, radiation, industrial, and chemical process safety. AES states that the procedures describe, as a minimum, the following: (a) qualifications of personnel authorized to perform the maintenance, functional testing, or surveillance/monitoring; (b) controls on and specification on any replacement components or materials to be used; (c) post-maintenance testing to verify operability of the equipment; (d) tracking and records management of maintenance activities; and (e) safe work practices.

As applicable, contractors that work on or near IROFS identified in the ISA Summary will be required by the applicant to follow the same maintenance guidelines described for the AES maintenance function.

Section 11.2 of the SAR (AES, 2009a) also states that maintenance procedures for QA Level 1 and QA Level 2 items and activities will commit, as applicable to: (a) reviews of the work to be performed for pre-maintenance activities; (b) NCS evaluation and analysis, if required, for new procedures or work activities that involve or could affect uranium onsite; (c) steps that require notification of affected parties before performing work and completion of maintenance work; and (d) control of work by comprehensive procedures to be followed by maintenance technicians.

11.3.2.1 Surveillance/Monitoring

In Section 11.2.1 of the SAR (AES, 2009a) the surveillance/monitoring is described as the process that will be used to detect degradation and adverse trends of IROFS so that action may be taken prior to component failure. The parameters to be monitored will be selected based upon their ability to detect the predominant failure modes of the critical components. Surveillance of QA Level 1 and QA Level 2 items will be performed at specified intervals. The surveillance activity supports the determination of performance trends for IROFS indicating when potential performance degradation exists, adjusting the preventive maintenance frequencies or taking any other corrective action. Moreover, surveillance/monitoring results will be evaluated to determine any impact on the ISA or any updates needed. For surveillance tests that can only be done while the equipment is out of service, proper compensatory measures will be prescribed in maintenance procedures.

Section 11.2.1 also describes how incident investigations may identify root causes of failures that are related to the type or frequency and maintenance. These incident investigation results will be used as lessons learned in the surveillance/monitoring and preventive maintenance program as appropriate. Records showing the current surveillance schedule, performance criteria, and test results for IROFS will be maintained in accordance with the Records Management System.

11.3.2.2 Corrective Maintenance

As described in Section 11.2.2 of the SAR (AES, 2009a), the corrective maintenance will involve repairs or replacement of equipment that has unexpectedly degraded or failed. The program will provide for restoration of a QA Level 1 or QA Level 2 item to acceptable performance through a planned, systematic, controlled, and documented approach for the repair and replacement activities. After conducting a corrective maintenance and before returning QA Level 1 or QA Level 2 items to operational status, the applicant will conduct a functional test, if necessary, to ensure that the QA Level 1 or QA Level 2 items will perform their intended safety functions. Corrective maintenance results will be evaluated to determine any impact on the ISA or any updates needed.

11.3.2.3 Preventive Maintenance

Section 11.2.3 of the SAR (AES, 2009a) states that preventive maintenance (PM) will include pre-planned and scheduled periodic refurbishment, partial or complete overhaul, or replacement of QA Level 1 and QA Level 2 items, if necessary, to ensure their continued safety function. The applicant will use the results of surveillance and monitoring, including failure history, to plan PM activities.

Calibration and testing are also described in Section 11.2.3 of the SAR (AES, 2009a) within the PM function. Calibration standards will be used by facility personnel to calibrate equipment and monitor devices to plant safety and safeguards. The applicant will provide compensatory measures during testing of non-redundant QA Level 1 and QA Level 2 items to ensure that their function is performed until the item is back in service.

The applicant will use industry experience, operating data, surveillance data, and plant equipment operating experience to determine initial PM frequency and procedures. The applicant states that in determining the frequency of PM, consideration will be given to appropriately balancing the objective of preventing failures through maintenance against the objective of minimizing the unavailability of IROFS because of PM. Feedback from PM, corrective maintenance, and incident investigation will be used, as appropriate, to modify the frequency or scope of PM activities. The applicant will document the rationale for any PM deviations from industry standards or vendor recommendations.

After conducting PM and before returning QA Level 1 or QA Level 2 item to operational status, the applicant will conduct a functional test, if necessary, to ensure that the QA Level 1 or QA Level 2 item will perform its intended safety function. Records pertaining to PM will be maintained in accordance with the Records Management System. PM activities results will be evaluated to determine if there are impacts on the ISA or if there are updates needed.

11.3.2.4 Functional Testing

Functional testing for QA Level 1 and QA Level 2 items as described in Section 11.2.4 of the SAR (AES, 2009a) will be performed, as appropriate, following initial installation as part of periodic surveillance testing, and after corrective or preventive maintenance or calibration to ensure that the item is capable of performing its safety function, when required.

The applicant's overall testing program is divided into two major testing programs: pre-operational testing and operational testing program. The objectives of these programs as described in Section 11.2.4.1 of the SAR (AES, 2009a) are to ensure that IROFS: (1) have been adequately designed and constructed; (2) meet contractual, regulatory, and licensing requirements; (3) do not adversely affect worker or the public's health and safety; and (4) can be operated in a dependable manner so as to perform their intended function. The programs also ensure that operating and emergency procedures are correct and that personnel have acquired the correct level of expertise.

As described in Section 11.2.4.2 of the SAR (AES, 2009a) test requirements will be specified in written procedures, except that in lieu of written procedures, appropriate sections of related documents (e.g., American Society for Testing and Materials methods, external manuals, and maintenance instructions) may be used. The applicant describes the content of test procedures. Procedures will be detailed enough that qualified personnel can perform the required functions without direct supervision.

The pre-operational testing program is described in Section 11.2.4.3 of the SAR (AES, 2009a) and will consist of three parts: constructor turnover, pre-operational functional testing, and initial start-up testing. During the constructor turnover, the constructor will be responsible for completion of as-built drawing verification, purging, cleaning, vacuum testing, system turnover, and initial calibration of instrumentation in accordance with the design and installation specifications that were provided by the vendors and architect engineers. The applicant will conduct pre-operational functional testing following constructor turnover to initially determine various facility parameters and to initially verify the capability of SSCs to meet performance requirements. The applicant will perform initial start-up testing beginning with the introduction of uranium hexafluoride (UF_6) and ending with the start of commercial operation.

The pre-operational testing program objective will be to perform all necessary tests to verify that the QA Level 1 and QA Level 2 items that are essential to the safe operation of the plant are capable of performing their intended safety function. The applicant will perform initial start-up testing to ensure safe and orderly UF_6 feeding, and to verify parameters assumed in the ISA. The overall program will be reviewed and approved by the Plant Manager prior to the initial UF_6 introduction. Results of each preoperational test will be reviewed and approved by the responsible department manager or designee. The results for start-up testing will be reviewed and approved by the Start-Up Manager.

The applicant describes the operational testing program in Section 11.2.4.4 of the SAR (AES, 2009a). The program will consist of periodic testing and special testing. The applicant will conduct periodic testing to monitor various facility parameters, and to verify continuing integrity and capability of QA Level 1 and QA Level 2 items. The applicant will conduct special testing which consists of testing that does not fall under any other testing program and is of a non-recurring nature. The procedures for periodic and special testing will be sufficiently detailed that qualified personnel can perform their functions without supervision.

The overall responsibility for the operational program will be shared among the Maintenance Manager; the Operations Manager; and the Environmental, Health, Safety and Licensing (EHS&L) Manager. The applicant also describes the responsibilities for Test Coordinators during operational testing in Section 11.2.4.4 of the SAR (AES, 2009a).

As described in Section 11.2.4.4.1 of the SAR (AES, 2009a) periodic testing will consist of testing conducted to verify the continuing capability of QA Level 1 and QA Level 2 items to meet performance requirements. The testing will begin during the pre-operational stage and continue to be scheduled through the facility's life. The testing schedule will be revised periodically to ensure that the changes in the testing requirements are reflected. The Maintenance Department will maintain a testing status index to assist groups in assuring that surveillances are being completed within the specified time interval. The applicant will process any occurrence of missed tests through their Corrective Action Program (CAP).

Special testing, as described in Section 11.2.4.4.2, of the SAR (AES, 2009a) will consist of testing conducted at the facility that is not a facility pre-operational test, periodic test, post-modification test or post-maintenance test. Some of the purposes for special testing will be: (1) acquisition of particular data for special analysis; (2) determination of information relating to facility incidents; (3) verification that required corrective actions reasonably produce expected results and do not adversely affect the safety of operations; and (4) confirmation that facility modifications reasonably produce expected results and do not adversely affect systems, equipment, and/or personnel by causing them to function outside established design conditions (applicable to testing performed outside of a post-modification test). The applicant will consider special tests as facility changes or changes in process safety information that will require to be reviewed by the ISA method and in accordance with 10 CFR 70.72(a) and (c). The applicant does not intend to include routine surveillances, normal operation evaluations, and activities where which there is previous experience in the conduct and performance of the activity within special testing.

11.3.2.5 Evaluation Findings

As described in the preceding sections of this SER, the staff reviewed the applicant's maintenance program following the guidance in Section 11.4.3.2 of NUREG-1520 (NRC, 2002).

In Section 11.2 of the SAR (AES, 2009a) the applicant described the maintenance program which includes surveillance and monitoring (Section 11.2.1 of the SAR), corrective maintenance (Section 11.2.2 of the SAR), PM (Section 11.2.3 of the SAR), and functional testing (Section 11.2.4 of the SAR). The applicant described how the maintenance function will be designed and will have measures in place to ensure that the objective of preventing failures through maintenance is appropriately balanced against the objective of minimizing unavailability of IROFS because of monitoring or PM.

In Section 11.2.1 of the SAR (AES, 2009a), the applicant described the surveillance/monitoring function. The description included the statement that surveillance activities will be conducted at specified intervals. The applicant also described how the surveillance activity supports the determination of performance trends for IROFS, thus providing data useful in determining PM frequencies. The applicant also provides an adequate description of the records retention for surveillance activities. Furthermore, for surveillance tests that can be done only while IROFS are out of service, the applicant stated that proper compensatory measures will be prescribed for the continued normal operation of a process. The applicant provided a description of how the results of incident investigations are used to modify the affected maintenance function and eliminate or minimize the root cause.

In Section 11.2.2 of the SAR (AES, 2009a), the applicant described the approach for corrective maintenance. The applicant described how the maintenance function provides a planned, systematic, integrated, and controlled approach for the repair and replacement activities associated with identified unacceptable performance deficiencies of IROFS.

In Section 11.2.3 of the SAR (AES, 2009a), the applicant described PM which includes preplanned and scheduled periodic refurbishing, or partial or complete overhaul, of IROFS to minimize occurrences of their unanticipated losses. The applicant also described how the retention of records showing the PM schedule and results for all IROFS subject to PM will be handled. In addition the applicant addressed calibration as part of the maintenance function. The applicant provided the methodology that will be used to determine the PM frequency and also provided how the PM function will be designed to assure that the objective of preventing failures through maintenance will be appropriately balanced against the objective of minimizing unavailability of IROFS because of monitoring or PM.

In Section 11.2.4 of the SAR (AES, 2009a), the applicant described the methods and approved procedures used to perform functional testing, as needed, of IROFS after PM or corrective maintenance. The applicant also described how functional tests are designed to include all operational aspects of the IROFS during startup of new processes. The applicant described the handling and maintenance of records showing the functional test schedule and results for all IROFS subject to functional testing.

Based on the above, the staff found that the maintenance program description provided by the applicant in Section 11.2 and its subsections is consistent with the guidance provided in Section 11.4.3.2 of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

11.3.3 Training and Qualifications

The NRC staff has reviewed the applicant's training and qualification program following the regulatory acceptance criteria in Section 11.4.3.3 of NUREG-1520 (NRC, 2002).

The applicant's QA program provides training and qualification requirements during design, construction, and operations phases for QA training of personnel performing QA Level 1 and QA Level 2 work activities; for non-destructive examination, inspection and test personnel; and for QA auditors. As described in Section 11.3 of the SAR (AES, 2009a), the principal objective of the training program system is to ensure job proficiency of all of all facility personnel involved in QA Level 1 and QA Level 2 activities through effective training and qualification.

11.3.3.1 Organization and Management of the Training Function

As described in Section 11.2.1 of the SAR (AES, 2009a), line management will be responsible for the content and effective conduct of training. The position description for line managers includes their training responsibilities; they are given the authority to implement training for their personnel supported by the training organization. The Training Manager will be responsible for the facility training programs. The applicant will use performance-based training as the primary management tool for analyzing, designing, developing, conducting, and evaluating training.

The applicant will have administrative procedures establishing requirements for indoctrination and training for personnel performing activities relied on for safety and to ensure that all phases of training are conducted reliably and consistently. The applicant will grant exception from training requirements when justified, documented, and approved by appropriate management. The applicant will include affected lesson plans in the change control process when any design changes or facility modifications are implemented. The applicant will maintain programmatic and individual training records to support management information needs associated with personnel training, job performance, and qualifications. Individual records will include general employee training, technical training, and employee development training conducted at the facility.

11.3.3.2 Analysis and Identification of Functional Areas Requiring Training

As described in Section 11.3.2 of the SAR (AES, 2009a), a needs/job analysis will be performed and will identify tasks to ensure that appropriate training is provided to personnel working on tasks related to QA Level 1 and QA Level 2 items and activities. On-the-job training (OJT) will include Job Hazards Analysis to provide the employees with the skills necessary to perform their jobs safely.

Relevant technical and management personnel will be consulted as necessary to identify activities where training is appropriate. The applicant will compare and review the activities selected for training with training materials as part of a training effectiveness evaluation. The applicant will update the list of activities selected for training as necessitated by changes in procedures, processes, plant systems, equipment, or job scope.

11.3.3.3 Position Training Requirements

As stated in Section 11.3.3 of the SAR (AES, 2009a), minimum training requirements will be developed for positions whose activities are relied on for safety. The initial identification of job-training requirements will be based on experience; and the level at which an employee will initially enter the training program will be determined by an evaluation of the employee's past experience, level of ability and qualifications. The applicant's position descriptions will describe the entry level criteria for positions whose activities are relied on for safety. The applicant also states that facility personnel may be trained through participation in general employee training, technical training, or employee development/management-supervisory training.

Training requirements will be applicable to personnel within the plant organization who have direct relationship to the operation, maintenance, testing, or other technical aspects of the facility's IROFS. To maintain personnel proficiency, the applicant will conduct continuing and periodic retraining.

General Employee Training

Section 11.3.3.1 of the SAR (AES, 2009a) describes the general employee training as those including QA, radiation protection, safety, emergency, and administrative procedures established by facility management and applicable regulations. It states that persons that are under the supervision of facility management, including contractors, must participate in general employee training; however, certain facility support personnel may not participate in all topics depending on their normal work assignment. The general employee training will include the following topics: general administrative controls and procedure use, QA policy and procedures, facility systems and equipment, nuclear safety, industrial safety, health and first aid, emergency plan and implementing procedures, facility security programs, chemical safety, fire protection and fire brigade, and new employee orientation.

Nuclear Safety Training

As described in Section 11.3.3.1.1 of the SAR (AES, 2009a), training programs will be established for various types of job functions commensurate with criticality and radiation safety responsibilities associated with each function. The nuclear safety training program will include topics such as notices, reports and instruction to workers; practices designed to keep radiation exposures ALARA; methods of controlling radiation exposures; contamination control methods; use of monitoring equipment; emergency procedures and actions; nature and sources of radiation; safe use of chemicals; biological effects of radiation; use of personnel monitoring devices; principles of nuclear criticality safety; risk to pregnant females; radiation protection practices; protective clothing; respiratory protection; and personnel surveys. Also Section 11.3.3.1.1 states that the criticality safety program will be in accordance with ANSI/ANS 8.19 - 1996 and ANSI/ANS 8.20 - 1991.

The training sessions will be conducted by instructors assigned by the EHS&L Manager. The trainings will have an initial examination that must be passed to demonstrate understanding and effectiveness of the training. Employees will be required to complete the nuclear safety training before having unescorted access into controlled areas. Unescorted access to controlled areas requires annual retraining.

The EHS&L Manager or designee will periodically revise and update the nuclear safety training programs to ensure that they are still current and adequate. The radiation protection sections will be reviewed at least annually or by the request of the Radiation Protection/Chemistry manager or designee. Also, supervisors will review radiation safety topics at least annually with their workers.

<u>Fire Brigade Training</u>

Section 11.3.3.1.2 of the SAR (AES, 2009a) describes the primary purpose of the fire brigade training program as to develop a group of employees that will be skilled in fire prevention, fire-fighting techniques, first-aid procedures, and emergency response. Its intention is not to replace the local fire fighters. They will be the first-response and will supplement the local fire fighters. The program will provide initial training, semi-annual classroom training and drills, annual practical training, and leadership training for the fire brigade leaders.

<u>Technical Training</u>

Technical Training, as described by the applicant in Section 11.3.3.2 of the SAR (AES, 2009a), will be designed, developed and implemented for assisting employees to understand applicable fundamentals, procedures, and practices common to a gas centrifuge uranium facility. In addition, the technical training will be used to develop the skill to perform the assigned work. The technical training will consist of initial training, OTJ training and qualifications, continuing training, and special training.

<u>Initial Training</u>

As described in Section 11.3.3.2.1 of the SAR (AES, 2009a), initial training will be used to provide understanding of the fundamentals, basic principles, and procedures related to the employees' work activities. Initial training will consist of—but not limited to—live lectures, taped and filmed lectures, self-guided study, demonstrations, laboratories, workshops, and OJT training.

The training requirements for certain new or transferred employees that are partially qualified from previous, applicable training or experience will be determined by applicable regulations; performance in review sessions; comprehensive examinations; or other techniques that can identify the employee's level of ability. Allowances will be made to suit specific situations, depending on regulatory requirements and individual needs. The trainee progress will be evaluated by written examinations and oral and written tests.

The training program will be arranged in logical blocks or modules and provides brief descriptions of the modules. The description of operations, mechanical maintenance, instrumentation and electrical maintenance, health physics and chemistry, and engineering/professional will be included as part of initial training modules.

Operations Initial Training

The applicant describes four modules for the operation's initial training: general systems, specific systems, nuclear preparatory, and plant familiarization. The general systems module will provide the trainees with basic concepts and fundamentals in mathematics, physics, chemistry, heat transfer, and electrical theory. The applicant may conduct OJT orientation. The specific training module will provide basic instruction in system and component identification, and basic system operating characteristics. The training module will provide a general overview of enrichment plant equipment and acquaints the trainees with enrichment plant terminology and nomenclature. It also provides instructions describing basic systems operations.

The applicant states that the nuclear preparatory training module will be presented to operations personnel following the specific systems module. The module develops the necessary concepts in basic nuclear physics, plant chemistry, basic thermodynamics, radiation protection, and enrichment theory.

The plant familiarization module will be provided by the applicant as orientation for employees to become familiar with plant layouts, plant systems, and practicable laboratory and equipment work at the facility.

Mechanical Maintenance Initial Training

The applicant describes three modules for the mechanical maintenance initial training: general systems, fundamental shop skills, and plant familiarization. The general systems module will be the same as the previously described course in the Operations Initial Training.

The fundamental shop skills module, as described by the applicant, will provide instruction in fundamentals of mechanical engineering performance. The applicant will combine academic instruction and hands-on training in this module to familiarize the trainees with design operational and physical characteristics of enrichment facility components and basic skills and procedures used to perform mechanical repairs and equipment replacement. The applicant will use task training lists to assure the trainees level of performance. Trainees will be able to use work procedures to guide them through an assigned task. All the phases of this module will stress the radiological and industrial safety.

The plant familiarization module will be the same course as noted previously.

Instrumentation and Electrical Maintenance Training

The applicant describes four modules for the instrumentation and electrical maintenance initial training: general systems, basic instrument and electrical, basic performance and plant familiarization. The general systems module will be the same as the previously described course in the Operations Initial Training.

The basic instrument and electrical module, as described by the applicant, will provide trainees with refresher training in electrical and electronic fundamentals, digital techniques and application, instrumentation and control theory and application, and an introduction to the types and proper use of measuring and test equipment commonly used in enrichment facilities. Also, the applicant will provide working knowledge of nuclear and non-nuclear instrumentation systems; overall integrated plant operation and control; and, in particular, the hazards of calibration during plant operation.

The applicant explained the fundamental performance modules as the performance familiarization with plant test procedures, test equipment, and testing—as well as plant records, reports, and data collection. As part of this module the applicant will provide training to trainees for basic understanding of thermodynamics used in testing plant heat transfer.

The plant familiarization module will be the same course as noted previously.

Health Physics and Chemistry Initial Training

The applicant describes four modules for the health physics and chemistry initial training: general systems, fundamental health physics, fundamental chemistry, and plant familiarization. The general systems module will be the same as the previously described course in the Operations Initial Training.

The applicant will use the fundamental health physics training to present trainees with a comprehensive and technical understanding of the nuclear processes. The module will utilize non-automated counting and spectrographic equipment, and portable survey instruments. The applicant describes the training as providing detailed overview of administrative material.

The fundamental chemistry module, as described by the applicant, will provide familiarization with chemistry theory, techniques, and procedures to ensure that chemistry technicians can work safely and competently.

The plant familiarization module will be the same course as noted previously.

Engineer/Professional Initial Training

The applicant's engineer/professional training program will be for technical staff and managers. It is described in four modules: facility orientation; basic engineer/professional training; enrichment/chemical engineer/professional training; and engineer/professional systems training.

The facility's orientation training module, as described by the applicant, will provide an orientation to each section of the facility—including an OJT task list with objectives that must be completed while working on the section (AES, 2009a).

The basic engineer/professional training module, as described by the applicant, will provide a basic understanding of uranium enrichment, systems and components required for final product, and the interrelationship that is required among facility organizations to achieve the overall objective (AES, 2009a).

The enrichment/chemical engineer/professional training, as described by the applicant, will provide specific theoretical information related to the operations of the enrichment plant (AES, 2009a). Some of the topics, such as thermal science and nuclear physics, will address the applications in the enrichment facility (AES, 2009a).

The engineer/professional systems training module, as described by the applicant, will provide an overview of plant systems, components, and procedures that are required to operate the facility safely and efficiently (AES, 2009a).

OJT Training and Qualifications

The applicant describes the OJT as a systematic method of providing the required job-related skills and knowledge for a position. OJT will be performed in the work area, and it will supplement the formal classroom training. As described in Section 11.3.3.2.2 of the SAR (AES, 2009a), the objective of the program is to assure the trainee's ability to perform job tasks as described in task descriptions and the Training and Qualification Guides.

Continuing Training

Continuing training is defined by the applicant in Section 11.3.3.2.3 of the SAR (AES, 2009a) as any training not provided as initial qualification and basic training which improves job-related knowledge and skills. The categorization of continuing training will be as follows:

- Facility systems and component changes;
- OJT/Qualifications program retraining;
- Policy and procedure changes;
- Operating experience program documents review to include industry and in-house operating experiences;
- Continuing training required by regulation (e.g., emergency plan training);
- General employee, special, administrative, vendor, and/or advanced training topics supporting tasks that are elective in nature;
- Training identified to resolve deficiencies (task-based) or to reinforce seldom-used knowledge skills;
- Refresher training on initial training topics;
- Structured pre-job instruction, mock-up training, and walkthroughs; and
- Quality awareness

The applicant's continuing training will consist of formal and informal components performed at a certain frequency, ensuring proficiency on the job. A systematic approach will be used to determine the content of continuing training that can be offered, as needed, in any of the topics previously mentioned.

Special Training

Special training, as described in Section 11.3.3.2.4 of the SAR (AES, 2009a), involves those subjects of a unique nature required for a particular area of work. The applicant will provide special training to selected personnel based on specific needs. Special training will not be directly related to disciplinary lines.

11.3.3.4 Basis and Objectives for Training

The applicant's learning objectives are described in Section 11.3.4 of the SAR (AES, 2009a) and will identify the training content based on needs/jobs analyses and position-specific requirements. The objectives will include the knowledge, skills and abilities to be demonstrated; the conditions under which these actions are going to take place; and the standards of performance that is expected by the applicant.

11.3.3.5 Organization of Instruction, Using Lesson Plans and Other Training Guides

As described in Section 11.3.5 of the SAR (AES, 2009a), lessons plans will be developed under the guidance of the training function from the learning objectives. The lesson plans will be reviewed and approved by the training function and the cognizant organization in the subject matter prior to use.

11.3.3.6 Evaluation of Trainee Learning

Section 11.3.6 of the SAR (AES, 2009a) states that observation, demonstration, and oral or written tests will be used by the applicant to evaluate the trainee's understanding and command of learning activities. The evaluations are going to be performed by individuals qualified in the subject matter.

11.3.3.7 Conduct of OJT Training

As described in Section 11.3.7 of the SAR (AES, 2009a), OJT training will be used in combination with classroom activities that are QA Level 1 and QA Level 2. OJT training will be conducted by competent, designated personnel using the current performance-based training materials. The applicant states that the completion of OJT training will be demonstrated by actual task performance or performance of a simulation of the task with the trainee explaining task actions using conditions encountered during the performance of the task—including references, tools, and equipment reflecting the actual task to the extent practical.

11.3.3.8 Evaluation of Training Effectiveness

As described in Section 11.3.8 of the SAR (AES, 2009a), the applicant will evaluate the training program periodically to measure its effectiveness. The evaluation will consider feedback provided from trainees after completion of the classroom training sessions and identifies the program's strengths and weaknesses. The evaluation, as described by the applicant, will also determine whether the program content matches current job needs and if corrective actions are needed to improve the program's effectiveness.

The training function will be leading any training program evaluations and implementing the corrective actions. Some elements that are addressed by the applicant in the training objectives are:

- 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 configuration management 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 strength and weaknesses. The applicant will review identified weaknesses; recommend improvements; and make any changes to the affected procedures, practices, or training materials.

The facility system and training program will be audited by the QA Department. Trainees and vendors may provide inputs related to the facility's training program.

11.3.3.9 Personnel Qualification

Section 11.3.9 of the SAR (AES, 2009a) describes the personnel qualification as qualification and training requirements that will be established and implemented for process operator candidates in plant procedures. Qualification requirements for key management positions are described by the applicant in Chapter 2, "Organization and Administration" of the SAR. The training and qualification requirements for QA personnel are provided in the QAPD for design, construction, operation, and decommissioning of the EREF.

11.3.3.10 Periodic Personnel Evaluations

As described in Section 11.3.10 of the SAR (AES, 2009a), personnel who perform activities relied on for safety will be evaluated at least biennially to determine whether they are capable of continuing to perform these activities. Personnel will be evaluated by written test, oral test, or on-the-job performance evaluation. The applicant will provide retraining or other appropriate action when results of the evaluations dictate the need. Retraining will be provided to personnel if any new or revised information results in plant modifications, procedure changes, and QA program changes.

11.3.3.11 Evaluation Findings

As described in the preceding sections of this SER, the staff reviewed the applicant's training and qualifications program following the guidance in Section 11.4.3.3 of NUREG-1520 (NRC, 2002).

The applicant described in Section 11.3 of the SAR (AES, 2009a) the training programs for the operational phase of the facility, including the preoperational functional testing and initial startup testing.

In Section 11.3.1 of the SAR (AES, 2009a), the applicant described the organization and management of the training function. The applicant also described: (1) the line management responsibilities for the content and effective conduct of the training, (2) the job function, responsibility, authority, and accountability of personnel involved in managing, supervising, and implementing training; (3) the performance-based training as the primary management tool for analyzing, designing, developing, conducting, and evaluating training and (4) maintenance of programmatic and individual training records that support management information needs and provide required data on each individual's training and qualification. In addition the applicant provided an adequate method to ensure that: (1) documentation and implementation of procedures to provide reasonable assurance that all phases of training are conducted reliably and consistently; (2) training documents are linked to the CM system to provide reasonable assurance that the training reflects design changes and modifications; and (3) exemptions are from training to trainees and incumbents are will be granted only when justified, documented, and approved by management.

The applicant described in Section 11.3.2 of the SAR (AES, 2009a) the analysis and identification of functional areas required training to ensure that formal training is provided for each position or activity that is relied on for safety either in a classroom or OTJ or both. Section 11.3.3 of the SAR (AES, 2009a) described the different parts of the training program –General Employee Training, Technical Training and Employee Development/Management Supervisory Training – that will be conducted and which personnel will be required to complete them. The applicant also described how each activity selected for training (initial or continuing) from the facility-specific activities is correlated with supporting procedures and training materials.

In Section 11.3.3 of the SAR (AES, 2009a), the applicant described the position training requirements as appropriate for facility personnel and other staff who perform regulated activities commensurate with the assigned functional responsibility and authority of the respective personnel.

Section 11.3.4 of the SAR (AES, 2009a) described the training basis and objectives which includes knowledge, skills, and abilities that the trainee should acquire; the conditions under which required actions will take place; and the standards of performance the trainee should achieve upon completion of the training activity.

The applicant described in Section 11.3.5 of the SAR (AES, 2009a) the lesson plans and other training guides that will ensure that guidance is provided consistent to the conduct of training activities and based on required learning objectives derived from specific job performance requirements. The applicant will have requirements in place for the review and approval of all lesson plans or guides and other training materials before their issue and use.

The applicant described in Section 11.3.6 of the SAR (AES, 2009a) the evaluation of trainee accomplishment which will evaluate the trainee understanding through observation/demonstration or oral or written test as appropriate to measure the training skills and job performance.

The applicant described in Section 11.3.7 of the SAR (AES, 2009a) the OTJ training element that will be used for some training activities of activities relied on for safety. The applicant described OTJ training as a well-organized training activity that uses current training materials and is conducted by designated personnel who are competent in the program standards and training methods. Completion of OTJ training will be by actual task performance or "walked down," the conditions of task performance, references, tools, and equipment reflecting the actual task to the extent possible, when the actual task cannot be performed.

The applicant described in Section 11.3.8 of the SAR (AES, 2009a) the process by which the training effectiveness is evaluated and its relation to job performance. The process described by the applicant will ensure that its activities convey all required skills and knowledge and are used to revise the training, where necessary, based on the performance of trained personnel in the job setting. In addition, the applicant will (1) periodically conduct a comprehensive evaluation of individual training to identify strengths and weaknesses, (2) use feedback from trainee performance during training and from former trainees and their supervisors to evaluate and refine the training, and (3) initiate, evaluate, track, and incorporate improvements and changes to initial and continuing training to correct training deficiencies and performance problems.
The applicant described in Section 11.3.10 of the SAR (AES, 2009a) the periodic personnel evaluations that provide for continuing assurance of personnel training and qualification. The applicant provided a method that addresses periodic requalification of personnel by training or testing or both, as necessary, at least biennially, to provide reasonable assurance that personnel continue to understand, recognize the importance of, and be qualified to perform activities that are relied on for safety.

Based on the above, the staff found that the training and qualifications program description provided by the applicant in Section 11.3 and its subsections is consistent with the guidance provided in Section 11.4.3.3 of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

11.3.4 Procedures Development and Implementation

The NRC staff has reviewed the applicant's description of its procedures development and implementation following the regulatory acceptance criteria in Section 11.4.3.4 of NUREG-1520 (NRC, 2002).

As described in Section 11.4 of the SAR (AES, 2009a), the applicant will conduct activities involving licensed materials or QA Level 1 and QA Level 2 items and activities in accordance with approved procedures. Procedures will be made available to the NRC for inspection prior to the initial enrichment activities at the facility.

The applicant describes the four types of plant procedures that will be used to control activities: (1) operating procedures, (2) administrative procedures, (3) maintenance procedures, and (4) emergency procedures.

Operating procedures, as described by the applicant, will be developed for workstation and Control Room operators. The applicant will implement a methodology for identifying, developing, approving, implementing, and controlling operating procedures. The operating procedure will be used to directly control process operations. The operating procedures will include, among other things, the: (a) purpose of the activity; (b) governing regulations, polices, and guidelines; (c) type of procedure; (d) steps for each operating process phase; (e) hazards and safety considerations; (f) operating limits; (g) measures to prevent exposure; (h) associated IROFS and their functions, and (i) the timeframe for which the procedure is valid. Applicable safety limits and IROFS will be clearly identified in the procedures.

The applicant will consider the ISA results when identifying needed procedures. The method will ensure, as a minimum, that: (a) operating limits and IROFS are specified in the procedure; (b) procedures include required actions for off-normal conditions of operation; as well as normal operations; (c) if needed, safety checkpoints are identified at appropriate steps in the procedure; (d) procedures are validated through field tests; (e) procedures are approved by management personnel responsible and accountable for the operation; (f) a mechanism is specified for revising and reissuing procedures in a controlled manner; (g) the QA elements and CM Program at the facility provide reasonable assurance that current procedures are available and used at all work locations; and (h) the facility's training program trains the required persons in the use of the latest procedures available (AES, 2009a).

The applicant will use administrative procedures to perform activities that support the process operations. The activities include management measures such as: (a) configuration management; (b) nuclear criticality, radiation, chemical, and fire safety; (c) QA; (d) design control; (e) plant personnel training and qualification; (f) audits and assessments; (g) incident investigations; (h) recordkeeping and document control; (i) reporting; and (j) procurement. Administrative procedures will also be used for the implementation of the emergency and security plans including the Emergency Plan, the Physical Security Plan, the Fundamental Nuclear Material Control Plan, and the Standard Practice Procedures Plan for the Protection of Classified Matter.

Maintenance procedures will address the preventive and corrective maintenance, surveillance functional testing, and the requirements for pre-maintenance activity involving reviews of the work to be performed and reviews of procedures of QA Level 1 and QA Level 2 activities or items. The emergency procedures will address the pre-planned actions of operators and other plant personnel in the event of an emergency.

The applicant will establish and implement procedures for the NCS in accordance with ANSI/ANS-8.19-1996 (ANSI, 1996). The applicant states that NCS procedures will be written such that no single, inadvertent departure from a procedure could cause an inadvertent criticality. The applicant will establish NCS postings identifying the administrative controls applicable to the activity or area in question.

The applicant will conduct periodic reviews of procedures to assure their continued accuracy and usefulness. The applicant, at a minimum, will review all operating procedures every five years and emergency procedures every year. In addition, the applicant will review applicable procedures after unusual incidents, such as an accident, unexpected transient, significant operator error, equipment malfunction, or after any modification to a system. Procedures will be revised as needed.

11.3.4.1 Preparation of Procedures

As described in Section 11.4.1 of the SAR (AES, 2009a), the procedures will be assigned to a member of the facility staff or contractor for development. The applicant will verify the technical accuracy of procedures by interdisciplinary reviews. The applicant will perform a walkdown of the procedure in the field or a tabletop walkthrough to verify that the procedure can be performed as written. The procedures for the operation of QA Level 1 and QA Level 2 items will be subjected to an independent review that will be performed by personnel not having direct responsibility for the work function under review. The applicant states that the designated approver will determine if any additional cross-disciplinary review is required. The Plant Manager or designee will have the responsibility of approving all procedures. However, any procedures that involve QA directly will require the QA Manager's approval.

11.3.4.2 Administrative Procedures

The applicant describes in Section 11.4.2 of the SAR (AES, 2009a) administrative procedures as those written by each department as necessary to control activities that support process operations, including management measures. The applicant will write administrative procedures for several areas, such as:

A. The operator's authority and responsibility,

B. Activities affecting facility operation or operating indications,

C. Manipulation of facility control,

D. Relief of duties,

E. Equipment control,

F. Master surveillance testing schedule,

G. A Control Room Operations Logbook, and

H. Fire Protection Procedures.

The applicant describes the process for administrative control of maintenance. The applicant will establish a maintenance program for QA Level 1 and QA Level 2 items. The personnel performing maintenance activities will be qualified in accordance with applicable codes and standards and procedures. The work will be conducted in accordance with written procedures that conform to applicable codes, standards, specifications, and other appropriate criteria. The applicant will schedule maintenance appropriately and without jeopardizing the facility operation or the safety of facility personnel. The applicant will maintain maintenance records for QA Level 1 and QA Level 2 items.

11.3.4.3 Procedures

The applicant states in Section 11.4.3 of the SAR (AES, 2009a) that activities involving licensed materials or QA Level 1 and QA Level 2 items and activities will be conducted in accordance with approved procedures. These procedures' intent is to provide a pre-planned method of conducting operations of systems to eliminate errors due to on-the-spot analysis and judgments. Procedures, as described by the applicant, will be sufficiently detailed so that qualified individuals can perform the required functions without direct supervision and will contain a degree of flexibility appropriate to the activities being performed (AES, 2009a). In addition, the applicant has procedural guidance to identify the manner in which procedures are to be implemented.

The applicant will write plant-specific procedures for abnormal events for the facility. These procedures will be based on a sequence of observations and actions, with emphasis placed on operator responses to indications in the Control Room. The actions outlined by the applicant in abnormal event procedures will be based on a conservative course of action to be followed by the operating crew.

The applicant will issued temporary changes to procedures for operating activities that are of a non-recurring nature. The applicant will use the temporary changes to procedures when revision of an operating or other permanent procedure is not practical. The applicant states that temporary changes to procedures shall not involve a change to the ISA and shall not alter the intent of the original procedure. The temporary changes to procedures will be approved by two members of the facility management staff, at least one of whom is a Production Manager. The applicant will document, review and approve temporary changes to procedures in accordance with the process described in Section 11.4.4 of the SAR, "Changes to Procedures," and will identify the approved duration for the temporary change in the temporary procedure change in accordance with 10 CFR 70.72(a)(5).

The applicant will have written procedures, documented instructions, checklists, or drawings for the maintenance of the facility's SSCs—including testing and calibration. The procedures will be written in accordance to the appropriate circumstances and will conform to applicable codes, standards, specifications, and other appropriate criteria. The Maintenance Department, under the Maintenance Manager, will be responsible for the preparation and implementation of maintenance procedures. In addition, the Maintenance Manager will have the overall responsibility for assuring that the periodic testing is in compliance with the requirements.

The applicant will provide for compensatory measures to be put into place when testing IROFS that are not redundant to ensure that the IROFS will perform until it is put back into service. The periodic test procedures will be performed by the Operations and Maintenance departments.

Chemical and radiochemical activities associated with facility IROFS will be performed in accordance with approved, written procedures. The Radiation Protection/Chemistry Manager will have the responsibility for preparation and implementation of chemistry procedures. The radioactive waste management activities associated with the facility's liquid, gaseous, and solid waste systems will also be performed in accordance with approved, written procedures. In this case, the facility's operations and radiation protection/chemistry departments will have the responsibility for preparation and implementation of the radioactive waste management procedures.

Procedures, as stated by the applicant, will include provisions for operations to stop and place the process in a safe condition if a step of a procedure cannot be performed as written.

11.3.4.4 Changes to Procedures

As described in Section 11.4.4 of the SAR (AES, 2009a), the process for the changes to procedures will have the preparer of the change document the change and the reason for the change. The applicant will perform an evaluation in accordance with 10 CFR 70.72, as appropriate, to determine if a change to the license is needed to implement the proposed changes. If a change to the license is needed, the change is not going to be implemented until prior approval is received from the NRC. The procedure will then be reviewed by a qualified reviewer. The Plant Manager, a functional area manager, or a designee approved by the Plant Manager will be responsible for approving procedure changes, and for determining whether a cross-disciplinary review is necessary: and by which department(s). The applicant will perform interdisciplinary reviews, as a minimum, for changes involving chemical safety, radiation safety, criticality safety, and changes involving nuclear material control and accounting. The applicant will maintain the records for cross-functional reviews for all changes to procedures involving licensed materials or QA Level 1 and QA Level 2 items and activities in accordance with Section 11.7 of the SAR, Records Management.

11.3.4.5 Distribution of Procedures

As described in Section 11.4.5 of the SAR (AES, 2009a), the applicant will distribute originally issued, approved procedures and approved procedure revisions in a controlled manner by Document Control. Document Control, as stated by the applicant, will establish and maintain an index of the distribution of copies of facility procedures. Revisions will be controlled and distributed in accordance with this index. Indexes will be reviewed and updated on a periodic basis or as required. The Department Managers or their designees will be responsible for ensuring that personnel doing work which require the use of the procedures have ready access to controlled copies of the procedures.

11.3.4.6 Evaluation Findings

As described in the preceding sections of this SER, the staff reviewed the applicant's procedures development and implementation strategy following the guidance in Section 11.4.3.4 of NUREG-1520 (NRC, 2002).

The applicant provided in Section 11.4 of the SAR (AES, 2009a) information regarding the procedure categories used at the facility including operating, administrative, maintenance, and emergency procedures for activities involving licensed materials or QA Level 1 and QA Level 2 items and activities. The applicant provided a listing of the types of activities that are covered, or are planned to be covered, by written procedures including topics of administrative procedures; system procedures that address startup, operation, and shutdown; abnormal operation or alarm response; maintenance activities that address system repair, calibration, inspection, and testing; and emergency procedures. The applicant also described methods for identifying, developing, approving, implementing, and controlling operating procedures. This method will consider the ISA in identifying needed procedures; it requires the procedures to

(1) specify operating limits and IROFS, (2) include required actions for off-normal conditions of operation, as well as normal operations, and (3) identify safety checkpoints, as required; and use field tests to validate procedures. In addition, the method will also ensure that (1) management personnel who are responsible and accountable for the operation will approve the procedures; (2) mechanism for revising and reissuing procedures in a controlled manner is specified; (3) QA elements and CM functions at the facility provide reasonable assurance that current procedures are available and used at all work locations; and (4) training program instructs the required personnel in the use of the latest procedures.

In addition, in Section 11.4 of the SAR (AES, 2009a) the applicant (1) included provisions for periodic reviews of the procedures to ensure their continued accuracy and usefulness and established the timeframe for reviews of the various types of procedures; (2) described the use and control of procedures; and (3) included provisions for the review of procedures after unusual incidents. The applicant provided in Section 11.4.1 of the SAR (AES, 2009a) the method for preparation of procedures. The applicant described the verification of procedures and those steps that ensure that the procedures are technically accurate and can be performed as written.

The applicant described in Section 11.4.3 of the SAR (AES, 2009a) the requirements governing the use of temporary procedures. The requirements include provisions for the issuance of temporary procedures only when permanent procedures do not exist to (1) direct operations during testing, maintenance, and modifications, (2) provide guidance in unusual situations not within the scope of permanent procedures, and (3) provide assurance of orderly and uniform operations for short periods when the facility, system, or component is performing in a manner not covered by permanent procedures. The applicant also described the approval and timeframe for use of the temporary procedure. Section 11.4.3 included provisions that allow for operations to stop and place the process in a safe condition if a step of a procedure cannot be performed as written.

Based on the above, the staff found that the description provided by the applicant for the procedures that will be used at the facility in Section 11.4 and its subsections is consistent with the guidance provided in Section 11.4.3.4 of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

11.3.5 Audits and Assessments

The NRC staff has reviewed the applicant's program for audits and assessments following the regulatory acceptance criteria in Section 11.4.3.5 of NUREG-1520 (NRC, 2002).

Section 11.5 of the SAR (AES, 2009a) describes audits and assessments and states that audits will be focused on verifying compliance with regulatory and procedural requirements, and licensing commitments. Furthermore, the applicant states that assessments will be focused on effectiveness of activities and ensuring that QA Level 1 and QA Level 2 items are reliable and are available to perform their intended safety functions. The applicant will perform audits and assessments on critical work activities associated with facility safety, environmental protection and other areas that they identify via trends. The QA Department will be responsible for audits and that these are conducted in accordance with a written plan that identifies and schedules audits to be performed. The applicant provides a description of the audit team. The audit team members will not have direct responsibility for the function and area being audited, and will have technical expertise or experience in the area being audited. In addition, team members will be indoctrinated in audit techniques. Audits will be conducted on an annual basis.

Assessments will be divided by the applicant into two categories: management assessments that will be conducted by the line organizations responsible for the work activity and independent assessments will be conducted by individuals not involved in the area being assessed. Audits of work activities associated with QA Level 1 and QA Level 2 items and activities will be the responsibility of the QA Department.

Audits and assessments will be performed to assure that facility activities are conducted in accordance with the written procedures and that the processes reviewed are effective. The applicant will conduct audits and assessments in the areas of: radiation safety, nuclear criticality safety, chemical safety, industrial safety including fire protection, environmental protection, emergency management, QA, CM, maintenance, training and qualification, procedures, CAP/Incident investigation and records management.

The applicant will perform audits and assessments routinely by qualified staff personnel that are not directly responsible for production activities. Any deficiencies identified during the audit or assessment requiring corrective action will be forwarded to the responsible manager of the applicable area or function for action in accordance with the CAP procedure. In addition, future audits and assessments will include a review to evaluate if corrective actions regarding any deficiencies found have been effective.

As discussed in the applicant's September 30, 2009 response to the staff's Request for Additional Information (RAI) (NRC, 2009), the activities of the QA Program which the QA Organization has direct responsibility for implementing will be performed by an independent third-party, Certified Lead Auditor that will be contracted or obtained from affiliate companies to an appropriate level of independence from AES (AES, 2009b). In addition, the QA Organization performs self-assessments of their areas of responsibility.

The results of the audits will be provided in a written report to the AES President, Plant Manager, the Safety Review Committee (SRC), and the Managers responsible for the activities audited. The applicant will maintain records of the instructions and procedures, persons conducting the audits or assessments, and any identified violations of license conditions and corrective actions taken.

11.3.5.1 Scheduling of Audits and Assessments

As described in Section 11.5.2 of the SAR (AES, 2009a), a schedule will be established identifying the audits and assessments to be performed and the responsible organization assigned to conduct the activity. The frequency of audits and assessments will be based on the status and safety importance of the activities being performed and upon work history. The system of audits and assessments will be designed by the applicant to ensure program oversight every three years. The schedule will be reviewed periodically and will be revised as necessary to ensure coverage commensurate with current and planned activities. NCS audits will be conducted and documented quarterly such that all aspects of the Nuclear Criticality Safety Program will be audited at least every two years. The Operations Department will be assessed to ensure that NCS procedures are being followed and that the process conditions have not been altered to adversely affect NCS. Assessments will be conducted at least semi-annually. In addition, the applicant will conduct and document weekly NCS walkthroughs of UF_6 process areas.

11.3.5.2 Procedures for Audits and Assessments

Section 11.5.3 of the SAR (AES, 2009a) states that internal and external audits and assessments will be conducted using approved procedures that meet the QA Program requirements. The procedures will provide requirements for audit and assessment activities, such as scheduling and planning of the audit and assessment; certification requirements of audit personnel; development of audit plans and audit and assessment checklists as applicable; performance of the audit and assessment; reporting and tracking of findings to closure; and closure of the audit and assessment. The applicable procedures will emphasize reporting and correction of findings to prevent recurrence.

The applicant will conduct audits and assessments by using approved audit and assessment checklists, interviewing responsible personnel, performing plant area walkdowns, reviewing controlling plans and procedures, observing work in progress, and reviewing completed QA documentation. The audit and assessment results data will be tracked in the CAP and periodically analyzed for potential trends.

The audits and assessments team leader will be required to develop the audit and/or assessment report to management with documented verification of performance against established performance criteria for QA Level 1 and QA Level 2 items and activities documenting the findings, observations, and recommendations for program improvement. The applicant will require that responsible managers review the reports and provide any required responses to reported findings. Audit reports will require an effectiveness evaluation and statement for each of the applicable QA program elements reviewed during the audit. The audit/assessment will be considered closed with the proper documentation as required by the applicable audit and assessment procedure.

11.3.5.3 Qualification and Responsibilities for Audits and Assessments

As described in Section 11.5.4 of the SAR (AES, 2009a), the QA Manager will initiate the audits; and in coordination with the responsible Lead Auditor, they will determine the scope of each audit. The QA Manager may initiate special audits or expand the scope of audits.

The Lead Auditor, as described by the applicant, will direct the audit team in developing checklists, instructions, or plans and how to perform the audit. Audits will be conducted in accordance with the checklists, but the scope may be expanded by the audit team during the audit. The audit team will consist of one or more auditors that are responsible for performing audits in accordance with the applicable QA procedures. The auditors and lead auditors will have to hold certifications as required by the QA Program. Prior to certification under the AES's QA Program, auditors will have complete training on: AES's QA Program; audit fundamentals, including audit scheduling, planning, performance, reporting, and follow-up action involved in conducting audits; objectives and techniques of performing audits; and OJT training. The certification of auditors and lead auditors will be based on the QA Manager's evaluation of education, experience, professional qualifications, leadership, sound judgment, maturity, analytical ability, tenacity, and past performance and completion of QA training courses.

Lead auditors will have to participate in a minimum of five QA audits, or audit equivalent within a period of time, not to exceed three years prior to the date of certification. Audit equivalents include assessments, pre-award evaluations or comprehensive surveillances (provided the prospective Lead Auditor took part in the planning, checklist development, performance, and reporting of the audit equivalent activities). One of the audits must be a nuclear-related QA audit or audit equivalent within the year prior to certification.

Personnel performing assessments will not require certification; but they will be required to complete QA orientation training, as well as training on the assessment process. The NCS assessments will be performed under the direction of the criticality safety staff.

11.3.5.4 Evaluation Findings

As described in the preceding sections of this SER, the staff reviewed the applicant's audits and assessments program following the guidance in Section 11.4.3.5 of NUREG-1520 (NRC, 2002).

In Section 11.5 of the SAR (AES, 2009a) the applicant described the program directives covering the audit and assessment function including the activities to be audited, audit frequency, guidance in conducting the audit or assessment, assigned responsibilities for each phase of the work, and procedures for recording the results and recommending actions to be taken. It provided for the conduct of internal audits and independent assessments of activities significant to facility safety and environmental protection. The applicant described the audits as those performed to verify that operations are being conducted in accordance with regulatory requirements and license commitments and independent assessments as those conducted by offsite groups or individuals not involved in the licensed activity to verify that the health, safety, and environmental compliance functions are effectively achieving their designed purposes.

In Section 11.5.1 the applicant provided a list of the areas where audits and assessments will be conducted. The areas listed include: radiation safety, nuclear criticality safety, chemical safety, fire safety, environmental protection, emergency management, QA, CM, maintenance, training and qualification, procedures, incident investigation, and records management.

The applicant also included provisions for qualified personnel without direct responsibility for the function and area being audited or assessed to perform the audits and assessments. The subsections on Section 11.5 included specifications with regards to staff positions and committees responsible for audits and assessments and describe the levels of management to which results are reported.

Based on the above, the staff found that the audits and assessment program description provided by the applicant in Section 11.5 and its subsections is consistent with the guidance provided in Section 11.4.3.5 of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

11.3.6 Incident Investigations and Corrective Action Process

The NRC staff has reviewed the applicant's process for conducting incident investigations and its corrective action process following the regulatory acceptance criteria in Section 11.4.3.6 of NUREG-1520 (NRC, 2002).

11.3.6.1 Incident Investigations

The applicant describes in Section 11.6.1 of the SAR (AES, 2009a) the incident investigation process as a simple mechanism available for use by any person at the facility to report deficiencies, abnormal events, and potentially unsafe conditions or activities. Any abnormal events that potentially threaten or lessen the effectiveness of health, safety or environmental protection will be identified, and reported to, and investigated by the EHS&L Manager. The applicant will consider each event in terms of its requirements for reporting in accordance with regulations and will evaluate the event to determine the level of investigation required. The applicant's process of incident identification, investigation, root cause analysis, environmental protection analysis, recording, reporting, and follow-up will be addressed in and performed by written CAP procedures. CAP procedures will include guidance for classifying occurrences, including examples of threshold off-normal occurrences. The depth of the investigation will depend, as explained by the applicant, upon the severity of the classified incident in terms of the levels of uranium released and/or the degree of potential for exposure of workers, the public, or the environment.

The EHS&L Manager will be responsible for maintaining a list of agencies to be notified in case of the event; determining if a report to any agencies is required and notifying the appropriate agencies when required. In addition, the EHS&L Manager or designee will maintain a record and track to completion corrective actions to be implemented as a result of off-normal occurrence investigations in accordance with CAP procedures. The licensing organization will be responsible for all appropriate communications with government agencies.

The incident investigation process, as described by the applicant, will establish a process 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 report to the NRC as required by 10 CFR 70.50 (CFR, 2008d) and 70.74 (CFR, 2008e). The investigation process will include a prompt, risk-based evaluation and, depending on the complexity and severity of the event, one individual may suffice to conduct the evaluation. The investigator(s) will be independent from the line function(s) involved with the incident under investigation—and are assured of no retaliation for participating in investigations. Investigations will begin within 48 hours of the abnormal event, or sooner, depending on safety significance of the event. The record of IROFS failures required by 10 CFR 70.62(a)(3) (CFR, 2008f) will be reviewed as part of the investigation. Record revisions necessitated by post-failure investigation conclusions will be made within five working days of the completion of the investigation.

Qualified internal or external investigators will be appointed to serve on investigating teams when required. The teams will include at least one process expert and at least one team member trained in root cause analysis.

The applicant will monitor and document corrective actions through completion. The applicant will maintain auditable records and documentation related to abnormal events, investigations, and root cause analyses so that "lessons learned" may be applied to future operations of the facility. For each abnormal event, the incident report will include a description, contributing factors, a root cause analysis, findings, and recommendations. Relevant findings will be reviewed with all affected personnel. Details of the event sequence will be compared with accident sequences already considered in the ISA, and the ISA Summary will be modified to include evaluation of the risk associated with accidents of the type actually experienced. The applicant will develop CAP procedures for conducting an incident investigation, and the procedures will contain elements such as: (a) a documented plan for investigating an abnormal event; (b) a description of the functions, qualifications, and/or responsibilities of the manager who would lead the investigative team and those of the other team members; the scope of the team's authority and responsibilities; and assurance of cooperation of management; (c) assurance of the team's authority to obtain all the information considered necessary and its independence from responsibility for or to the functional area involved in the incident under investigation; (d) retention of documentation relating to abnormal events for two years or for the life of the operation, whichever is longer; (e) 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; (f) requirements to make available original investigation reports to the NRC on request; and (g) a system for monitoring the completion of appropriate corrective actions.

11.3.6.2 Corrective Action Process

Section 11.6.2 of the SAR (AES, 2009a) states that AES QA Program will identify the responsibilities and provide authority for those individuals involved in quality activities to identify any condition adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective materials and equipment, and non-conformances. These individuals will identify and document conditions adverse to quality, analyze and determine how the conditions can be corrected or resolved, and take such steps as necessary to implement corrective actions in accordance with documented procedures.

The QA Program will require regularly scheduled audits and assessments to ensure that needed corrective actions are identified and that employees have the authority and responsibility to initiate the corrective action process if they discover deficiencies. The QA Program will contain procedures for identifying, reporting, resolving, documenting, and analyzing conditions adverse to quality. The reports of conditions adverse to quality will be analyzed to identify trends in quality performance. Significant conditions adverse to quality and significant trends will be reported to senior management in accordance with CAP procedures, and follow-up actions will be taken by the QA Manager to verify proper and timely implementation of the corrective actions.

11.3.6.3 Evaluation Findings

As described in the preceding sections of this SER, the staff reviewed the applicant's incident investigations and corrective action processes following the guidance in Section 11.4.3.6 of NUREG-1520 (NRC, 2002).

In Section 11.6.1 of the SAR (AES, 2009a) the applicant described the incident investigation program as a mechanism available for use by any person to report abnormal events that may occur during operation of the facility to be investigated to determine their specific or generic root cause(s), generic implications, and risk significance; to recommend corrective actions; and to report to the NRC as required by 10 CFR 70.50, "Reporting Requirements," and 10 CFR 70.74, "Additional Reporting Requirements." In addition, the applicant include provision within this section to monitor and document corrective actions through completion and ensure that corrective actions are taken within a reasonable period to resolve findings from abnormal event investigations. The applicant also provided for the maintenance of documentation related to abnormal events for the life of the operation so that "lessons learned" may be applied to future operations of the facility. The applicant specified that details of the event sequence will be compared with accident sequences already considered in the ISA, and the ISA Summary will be modified to include evaluation of the risk associated with accidents of the type actually experienced.

Based on the above, the staff found that the incident investigation program description provided by the applicant in Section 11.6 and its subsections is consistent with the guidance provided in Section 11.4.3.6 of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

11.3.7 Records Management

11.3.7.1 Records Management program

The NRC staff has reviewed the applicant's records management program following the regulatory acceptance criteria in Section 11.4.3.7 of NUREG-1520 (NRC, 2002).

As described in Section 11.7 of the SAR (AES, 2009a), records management will be performed in a controlled and systematic manner to provide identifiable and retrievable documentation. The applicant will identify the QA Records to be generated, supplied or held in applicable design specifications, procurement documents, or other documents in accordance with approved procedures. The applicant will consider QA records valid after they are stamped, initialed, signed, or otherwise authenticated and dated by authorized personnel. For computer codes and computerized data used for activities relied on for safety, the applicant will establish procedures for maintaining readability and usability of older codes and data as computing technology changes.

The AES QA Program will require procedures for reviewing, approving, handling, identifying, retention, retrieval and maintenance of QA records. The applicant includes records such as results of tests and inspections required by applicable codes and standards, construction, procurement, and receiving records; personnel certification records; design calculation; purchase orders; specifications and amendments; procedures; incident investigation results and approvals or corrective action taken; various certification forms; source surveillance and audit reports; component data packages; and any other QA documentation required by specifications or procedures. The applicant will maintain records at locations where they can be reviewed and audited to establish that the required quality has been assured.

The applicant will maintain a Master File within the facility. The applicant will control the access and use of the Master File. Documents in the Master File will be legible and identifiable as to the subject to which they pertain. The applicant states that documents in the Master File may be originals or reproduced copies. The applicant may use computer storage of data in the Master File. The Master File as described by the applicant will provide for accurate retrieval of information without undue delays and will preclude deterioration of records. The applicant will provide written instructions for the storage of records in the Master File and will designate a supervisor to be responsible of the implementation of these requirements. The applicant will include in the instructions: (a) a description of the location(s) of the Master File and an identification of the location(s) of the various record types within the Master File; (b) the filing system to be used; (c) a method for verifying that records received are in agreement with any applicable transmittal documents and are in good condition. This is not required for documents generated within a section for use and storage in the same sections' satellite files; (d) a method for maintaining a record of the records received; (e) the criteria governing access to and control of the Master File; (f) a method for maintaining control of and accountability for records removed from the Master File; and (g) a method for filing supplemental information and for disposing of superseded records.

If the applicant uses a single records storage facility, the applicant will review the storage facility for adequacy of protecting the records by a person competent in the technical field of fire protection and fire extinguishing. The applicant states that dual records storage facilities will not be subject to the same review. Records related to health and safety will be maintained in accordance with the applicable requirements of Title 10, *Code of Federal Regulations*. The applicant will retain the records in accordance with the Records Management procedures.

The applicant provides a list of examples of records that will be retained.

11.3.7.2 Evaluation Findings

As described in the preceding sections of this SER, the staff reviewed the applicant's records management program following the guidance in Section 11.4.3.7 of NUREG-1520 (NRC, 2002). In Section 11.7 of the SAR (AES, 2009a) the applicant described the records management program which prepares, verifies, characterizes, and maintains records and ensures that records are legible, identifiable, and retrievable for their designated lifetimes. The applicant included provisions to (1) categorize records by relative safety importance to identify record protection and storage needs and to designate the retention period for individual kinds of records; (2) protect records against tampering, theft, loss, unauthorized access, damage, or deterioration while in storage; (3) establish and document procedures specifying the requirements and responsibilities for record selection, verification, protection, transmittal, distribution, retention, maintenance, and disposition; (4) implement procedures that assign responsibilities for records management, specify the authority needed for records retention or disposal, specify which records must have controlled access and provide the controls needed, provide for the protection of records from loss, damage, tampering, and theft or during an emergency, and specify procedures for ensuring that the records management system remains effective; and (5) have procedures in place to promptly detect and correct any deficiencies in the records management system or its implementation.

For computer codes and computerized data used for activities relied on for safety, as specified in the ISA Summary, the applicant described that procedure(s) will be established for maintaining readability and usability of older codes and data as computing technology changes. Based on the above, the staff found that the records management program description provided by the applicant in Section 11.7 is consistent with the guidance provided in Section 11.4.3.7 of NUREG-1520 (NRC, 2002) and is, therefore, acceptable.

11.3.8 Other QA Elements

11.3.8.1 Description of other QA Elements

The applicant described other QA elements in Section 11.8 of the SAR (AES, 2009a). The applicant's QA Program and its supporting manuals, procedures and instructions are applicable to items and activities designated as QA Level 1 and QA Level 2. The QA Program specifies mandatory requirements for performing activities affecting quality and is set forth in procedures and its revisions which are distributed on a controlled basis to organizations and individuals responsible for quality.

Prior to undertaking an activity, the applicant will document, approve and implement the applicable portions of the QA Program. The applicant states that a management assessment of the QA program is performed at least six months prior to scheduled receipt of licensed material on the site. Prior to this initial management assessment, it is monitored through project review meetings and annual assessments. Any items identified by the applicant as needing completion or modification will be entered into the CAP and corrective action completed before scheduled receipt of licensed material.

While construction activities are in progress, the applicant will simultaneously use the QA Program for design, construction, and pre-operational testing—and the QA Program for operations. The applicant will plan and schedule system turnover as construction is completed. Prior to system turnover, written procedures are developed for control of the transfer of SSCs and associated documentation. The procedures include checklists, marked drawings, documentation lists, system status, and receipt control.

The applicant states that anyone may propose changes to the QA Program supporting manuals and procedures. The changes will be implemented after the QA Manager reviews them and found acceptable and compatible with applicable requirements, guidelines and AES policy. The QA Program and supporting manuals and procedures will be reviewed periodically to ensure compliance with applicable regulations, codes, and standards.

Personnel performing activities covered by the QA Program will perform work in accordance with approved procedures, and will demonstrate suitable proficiency in their assigned tasks.

The applicant will establish formal training programs for QA policies, requirements, procedures, and methods. The applicant will provide ongoing training to ensure continuing proficiency as procedural requirements change. New employees will be required to attend a QA indoctrination class on authority, organization, policies, manuals, and procedures. The applicant will provide additional formal training in specific topics such as NRC regulations and guidance, procedures, auditing, and applicable codes and standards. The applicant will perform supplemental training as required. OJT training will be performed by the employee's supervisor in QA area-specific procedures and requirements. Training records will be maintained by the applicant for each person performing quality-related job functions.

Any activities that will be contracted will be identified and controlled by the applicant. The principal contractors will be required to comply with the portions of QA Program applicable to the scope of their work and their performance of contracted activities will be formally evaluated by the applicant commensurate with the importance of the activities to safety.

11.3.8.2 Evaluation Findings

In a letter dated October 30, 2009, AES submitted a request to the NRC for the expedited review and approval of the QAPD for the EREF (AES, 2009c). AES requested the expedited approval in order to be able to apply the QAPD language during its procurement of services and material. The staff completed a technical review of the QAPD on April 8, 2010 (NRC, 2010). Based on the review, documented in the accompanying Staff Evaluation Report, the staff found the program acceptable for application to the design, construction, operation, including maintenance and modification, and decommissioning of the proposed EREF (NRC, 2010). The Staff Evaluation Report discusses the staff review of the QAPD which was based on NUREG-1520 (NRC, 2002) and documents the staff's conclusion that the QAPD adequately describes the application of other QA elements and has adequately established other QA elements as part of Management Measures as required by 10 CFR Part 70.62(d) (NRC, 2010).

11.4 Evaluation Findings

This section is a summary of the staff's evaluation findings concerning management measures.

11.4.1 Configuration Management

As described in Section 11.3 of this SER, the staff has reviewed the CM function for the EREF following the criteria in Section 11.4.3.1 of the NUREG-1520. The staff's evaluation found that the applicant's description of the overall CM program appropriately covered CM Policy, design requirements, document control, change control, and assessments. The applicant has 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. Management-level policies and procedures, including an analysis and independent safety review of any proposed activity involving IROFS, are described and will provide reasonable assurance that consistency among design requirements, physical configuration, and facility documentation is maintained as part of a new activity or change in an existing activity involving licensed material. The management measures will include the following elements of CM:

<u>CM Management</u>

The applicant will develop the organizational structure, procedures, and responsibilities necessary to effectively implement CM.

<u>Design Requirements</u>

The applicant's design requirements and bases will be documented and supported by analyses. All design documentation will be maintained current.

<u>Document Control</u>

The applicant's documents, including drawings, will be appropriately stored and accessible. Drawings and related documents captured by the system will include those necessary and sufficient to adequately describe IROFS.

Change Control

Responsibilities and procedures will adequately describe how the applicant will achieve and maintain strict consistency among the design requirements, the physical configuration, and the facility documentation. Methods will be in place for suitable analysis, review, approval, and implementation of identified changes to IROFS, including appropriate CM controls.

Assessments

The applicant has committed to an adequate function that includes both initial and periodic assessments. The assessments will verify and ensure the adequacy of the CM function.

The CM program description provided by the applicant in Section 11.1 and its subsections is consistent with the guidance provided in Section 11.4.3.1 of NUREG-1520 (NRC, 2002) requiring the applicant to describe the overall CM function, how design requirements are established and maintained, document control, change control, and assessments of the CM function, and is therefore acceptable.

11.4.2 Maintenance

As described in Section 11.3.2 of this SER, the staff reviewed the applicant's description of its program to maintain the availability and reliability of IROFS identified in the ISA Summary and compared it to the guidance in Section 11.4.3.2 of NUREG-1520 (NRC, 2002). The maintenance program will include: surveillance/monitoring, corrective maintenance, PM, functional testing, equipment calibration, and work control for maintenance of IROFS. The applicant's maintenance function is proactive and uses maintenance records, PM records, and surveillance tests to analyze equipment performance and to seek the root causes of repetitive failures.

The surveillance and monitoring, PM, and functional testing activities described in the license application provide reasonable assurance that the IROFS identified in the ISA Summary will be available and reliable to prevent or mitigate accident consequences.

The maintenance function will: (1) be based on approved procedures; (2) employ work control methods that properly consider personnel safety, awareness of facility operating groups, QA, and the rules of CM; (3) use the ISA Summary to identify IROFS that require maintenance and determine the level of maintenance needed; (4) justify the PM intervals in terms of the equipment reliability goals; (5) provide for training that emphasizes the importance of IROFS identified in the ISA Summary, regulations, codes, and personnel safety; and (6) will create documentation that includes records of all surveillance, inspections, equipment failures, repairs, and replacements of IROFS.

The staff concludes that the applicant's description of the maintenance program meets the requirements of 10 CFR Part 70 and provides reasonable assurance of public health and safety and the protection of the environment.

11.4.3 Training and Qualification

As described in Section 11.3.3, the staff reviewed the applicant's strategy for development and implementation of the training and qualification program and compared it to the acceptance criteria guidance in Section 11.4.3.3 of NUREG-1520 (NRC, 2002). Based on this review, the NRC staff has concluded that the applicant has adequately described and assessed its personal training and qualification in a manner that: (1) satisfies regulatory requirements and (2) is consistent with the guidance in Section 11.4.3.3 of NUREG-1520 (NRC, 2002) which requires the applicant cover its training program appropriately covered: (a) organization and management of the training function, (b) analysis and identification of functional areas requiring training, (c) position training requirements, (d) training basis and objectives, (e) organization of instruction, (f) evaluation of trainee learning, (g) conduct OJT training, (h) evaluation of training effectiveness, (i) personnel qualification, and (j) periodic personnel evaluation. There is reasonable assurance that implementation of the described training and qualification will result in personnel who are qualified and competent to design, construct, start up, operate, maintain, modify, and decommission the facility safely. The staff concludes that the applicant's plan for personnel training and qualification meets the requirements of 10 CFR Part 70

11.4.4 Procedures

As described in Section 11.3.4 of this SER, the staff evaluated the applicant's description of the process for the development, approval, and implementation of procedures in accordance with the criteria in Section 11.4.3.4 of NUREG-1520 (NRC, 2002) to ensure that the applicant's procedures will address the operation of IROFS and all management measures supporting those IROFS and the applicant has described a process for developing, implementing, and controlling procedures.

The staff concludes that the applicant's plan for procedures meets the requirements of 10 CFR Part 70.

11.4.5 Audits and Assessments

As described in Section 11.3.5 of this SER, staff reviewed Section 11.5 of the SAR (AES, 2009) and compared the applicant's plan and processes to the review acceptance criteria in Section 11.4.3.5 of NUREG-1520 (NRC, 2002). Based on this review, the NRC staff has concluded that the applicant has adequately described its audits and assessments process and its policy directives, plans, and procedural requirements considering: (a) the general structure of the audits and assessments program, (b) the activities to be audited or assessed, (c) the scheduling of audits and assessments, (d) the procedures for audits and assessments; and (e) the qualifications and responsibilities for audits and assessments.

The staff concludes that the applicant's plan for audits and assessments meets the requirements of 10 CFR Part and provides reasonable assurance of protection of the health and safety of the public, the workers, and the environment.

11.4.6 Incident Investigations

As described in Section 11.3.6 of this SER, the staff reviewed the applicant's description of the organization responsible for: (1) performing incident investigations of abnormal events that may occur during operation of the facility; (2) determining the root cause(s) and generic implications of the event; and (3) recommending corrective actions for ensuring a safe facility and safe facility operations and found that it is consistent with the acceptance criteria in Section 11.4.3.6 of NUREG-1520 (NRC, 2002).
The applicant will monitor and document corrective actions through to completion. The applicant will also take follow-up actions to verify the proper and timely implementation of the corrective actions.

The applicant will maintain documentation so that "lessons learned" may be applied to future operations of the facility.

Accordingly, the staff concludes that the applicant's description of the incident investigation process complies with applicable NRC regulations, and is adequate.

11.4.7 Records Management

As described in Section 11.3.7 of this SER, the staff has reviewed the applicant's records management system against the acceptance criteria in Section 11.4.3.7 of NUREG-1520 (NRC, 2002) and concluded that the system: (1) will be effective in collecting, verifying, protecting, and storing information about the facility and its design, operations, and maintenance and will be able to retrieve the information in readable form for the designated lifetimes of the records; (2) will provide a records storage area(s) with the capability to protect and preserve health and safety records that are stored there during the mandated periods, including protection of the stored records against loss, theft, tampering, or damage during and after emergencies; and (3) will provide reasonable assurance that any deficiencies in the records management system or its implementation will be detected and corrected promptly.

11.4.8 Other QA Elements

In a letter dated October 30, 2009, AES submitted a request to the NRC for the expedited review and approval of the QAPD for the EREF (AES, 2009c). AES requested the expedited approval in order to be able to apply the QAPD language during its procurement of services and material. The staff completed a technical review of the QAPD on April 8, 2010 (NRC, 2010). Based on that review, documented in the accompanying Staff Evaluation Report, the staff found the program acceptable for application to the design, construction, operation, including maintenance and modification, and decommissioning of the proposed EREF (NRC, 2010). The Staff Evaluation Report discusses the staff review of the QAPD which was based on NUREG-1520 (NRC, 2002) and documents the staff's conclusion that the QAPD adequately describes the application of other QA elements and has adequately established other QA elements as part of Management Measures as required by 10 CFR Part 70.62(d) (NRC, 2010).

11.5 References

(AES, 2009a) AREVA Enrichment Services LLC, "Eagle Rock Enrichment Facility Safety Analysis Report," 2009.

(AES, 2009b) AREVA Enrichment Services LLC, "AREVA Enrichment Facility Responses to U.S. Nuclear Regulatory Commission's Requests for Additional Information," 2009.

(AES, 2009c) AREVA Enrichment Services LLC, "Request for Expedited Approval of Quality Assurance Program Description," 2009.

(NRC, 2010) U.S. Nuclear Regulatory Commission, "AREVA Enrichment Services Quality Assurance Program Description for the Eagle Rock Enrichment Facility (TAC L32738)," 2010.

(NRC, 2009) U.S. Nuclear Regulatory Commission, "Request for Additional Information – AREVA Enrichment Services LLC License Application for the Eagle Rock Enrichment Facility (TAC L32707)," 2009.

(NRC, 2002) U.S. Nuclear Regulatory Commission, NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," 2002.

CHAPTER 12.0 PHYSICAL PROTECTION

By letter, dated April 23, 2009, AREVA Enrichment Services, LLC (AES) submitted Revision 1 of the license application for review and approval by the U.S. Nuclear Regulatory Commission (NRC) in accordance with Title 10 of the *Code of Federal Regulations* (CFR) 70.22(k). The staff subsequently sent a Request for Additional Information (RAI), by letter, dated August 26, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092190320), that contained 12 issues to be addressed by AES. AES provided a response to the RAI. By letter, dated January 20, 2010, AES submitted the Physical Security Plan (PSP) Revision 2, which incorporated the responses to the RAI. The PSP also included the Transportation Security Plan (TSP) for the staff's review.

12.1 Regulatory Requirements

The following regulatory requirements apply to physical protection at the proposed facility:

1. 10 CFR 73.67 governs licensee fixed site and in-transit requirements for the physical protection of special nuclear material (SNM) of moderate and low strategic significance (LSS).

2. 10 CFR 73.71 requires reporting of safeguards events.

3. 10 CFR 73.73 specifies the requirements for advance notice and protection of export shipments of SNM of LSS.

4. 10 CFR 73.74 describes the requirement for advance notice and protection of import shipments on nuclear material from countries that are not party to the Convention on the Physical Protection of Nuclear Material.

12.2 Regulatory Guidance

The following guidance applies to physical protection at the proposed facility:

1. Regulatory Guide 5.59, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance." (NRC, 1983)

2. NRC Regulatory Issue Summary 2005-22, "Requirements for the Physical Protection During Transportation of Special Nuclear Material of Moderate and Low Strategic Significance: 10 CFR Part 73 vs. Regulatory Guide 5.59 (1983)." (NRC, 2005)

12.3 Staff Review and Analysis

The PSP and TSP were reviewed to ensure that the applicant has an acceptable understanding of the security requirements as outlined in 10 CFR Part 73; and has committed to implementing the general performance objectives of 10 CFR 73.67(a), the specific fixed site and in-transit security requirements of 10 CFR 73.67(c), (f), and (g), and the notification requirements of 10 CFR 73.71, 10 CFR 73.73, and 10 CFR 73.74.

The PSP was reviewed to ensure that the applicant committed to storing SNM only in a controlled access area, monitoring the protection of SNM by an intrusion detection system or other approved procedures, maintaining both a guard force and offsite response force available to respond to unauthorized penetrations or activities, establishing and maintaining response procedures for dealing with threats of theft or theft of SNM, and reporting safeguards events as required by regulation.

The TSP for the transportation of SNM of LSS was reviewed to ensure that the applicant committed to providing advance notification, receiving confirmation from receivers prior to shipping material; providing in-transit security; transporting the material in tamper-indicating, sealed containers; verifying the integrity of seals on shipments received; notifying shippers of receipt of shipment; and implementing the proper reporting and investigations in the event of lost or unaccounted for material.

12.4 Evaluation Findings

The NRC staff's review of the applicant's PSP for the protection of SNM of LSS contains information that has been marked as "Security-Related Information" by the applicant, pursuant to 10 CFR 2.390. The methods and procedures as outlined in the PSP satisfy the performance objectives, systems capabilities, and reporting requirements specified in 10 CFR 73.67 and 73.71. The PSP for the facility is acceptable and provides reasonable assurance that the requirements for the physical protection of SNM of LSS will be met.

The NRC staff's review of the applicant's TSP for the transportation of SNM of LSS contains information that has been marked as "Security-Related Information" by the applicant, pursuant to 10 CFR 2.390. The NRC staff also reviewed the applicant's TSP for SNM of LSS shipments originating from, or arriving at, the facility. The approaches and procedures as outlined in the TSP satisfy the performance objectives, systems capabilities, and event and advance notification requirements specified in 10 CFR 73.67(a), 10 CFR 73.67(f) and (g)(1)-(5), 10 CFR 73.71, 10 CFR 73.73, and 10 CFR 73.74. The NRC staff concludes that the facility TSP is acceptable and provides reasonable assurance that the requirements for the physical protection of SNM of LSS in transit will be met.

CHAPTER 13.0 SAFETY EVALUATION REPORT PREPARERS

The individuals and organizations listed below are the principal contributors to the preparation of this Safety Evaluation Report. The U.S. Nuclear Regulatory Commission's (NRC's) staff directed the effort and contributed to the technical evaluations. The staff also used contractor support from the Center for Nuclear Waste Regulatory Analyses and ICF Consulting in the preparation of this document.

U.S. NUCLEAR REGULATORY COMMISSION CONTRIBUTORS

Breeda Reilly, Office of Nuclear Material Safety and Safeguards (NMSS)	NRC Project Manager General Information and Organization and Administration
Thomas Hiltz, NMSS	Branch Chief, AFCB
Mary Adams, NMSS	Decommissioning
Damaris Arroyo, NMSS	Management Measures and Quality Assurance
James Bongarra, Office of New Reactors	Human Factors Engineering
Gregory Chapman, NMSS	Health Physics, Accident Analysis, and Environmental Protection
Jonathan De Jesus-Segarra, NMSS	Chemical Safety
Ira Dinitz, Office of Nuclear Reactor Regulation (NRR)	Liability Insurance
James Downs, NMSS	Fire Safety
Stan Echols, NMSS	Environmental Protection
Keith Everly, Office of Nuclear Security and Incident Response (NSIR)	Protection of Classified Matter
Herman Graves, Office of Nuclear Regulatory Research	Seismic and Structural
Shawn Harwell, NRR	Financial Qualification
Doug Hase, NSIR	Protection of Classified Matter
Kenneth Kline, Office of Federal and State Materials and Environmental Management Programs (FSME)	Decommissioning Financial Assurance

Mike Norris, NSIR	Emergency Management
Tom Pham, NMSS	Material Control and Accountability
Blake Purnell, NMSS	Nuclear Criticality Safety
Roman Przygodzki, FSME	Decommissioning Financial Assurance
David Rahn, NRR	Electrical Engineering and Instrumentation and Controls
Christopher Tripp, NMSS	Nuclear Criticality Safety
Glenn Tuttle, NMSS	Material Control and Accountability
Steve Ward, NMSS	Material Control and Accountability
Rex Wescott, NMSS	Integrated Safety Analysis
Barry Wray, NSIR	Physical Protection

CENTER FOR NUCLEAR WASTE REGULATORY ANALYSES CONTRIBUTORS

Asadul Chowdhury	Structural Design
Sui-Min (Simon) Hsiung	Natural Phenomena
John Stamatakos	Seismic Engineering

ICF CONSULTING CONTRIBUTORS

Craig Dean	Decommissioning Financial Assurance
Paul Bailey	Decommissioning Financial Assurance

APPENDIX A
INTEGRATED SAFETY ANALYSIS (ISA) AND ISA SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) staff's review of the applicant's ISA and ISA Summary is documented in Chapter 3 and Appendix A to this safety evaluation report. Appendix A contains information that has been marked by the applicant as "Security-Related Information" and "Export Controlled Information," pursuant to Title 10 of the *Code of Federal Regulations* (CFR) 2.390 and 10 CFR 810, respectively.

As discussed in Chapter 3 of this report, the NRC staff concluded that the applicant's commitment to conduct and maintain an ISA is in accordance with the requirements of 10 CFR 70.62(c)(1). Based on its review of the ISA Summary, the staff also concludes that the applicant provided an acceptable ISA Summary for the proposed facility that will meet the requirements of 10 CFR 70.65(b)(1) through (9); 10 CFR 70.23(a); and 10 CFR 70.64 (b).

APPENDIX B
ACCIDENT ANALYSIS

The U.S. Nuclear Regulatory Commission's (NRC) staff independently evaluated the consequences of a subset of potential accident sequences identified in the applicant's Integrated Safety Analysis (ISA) Summary. This evaluation is documented in Appendix B to this safety evaluation report. Appendix B contains information that has been marked by the applicant as "Security-Related Information," pursuant to Title 10 of the *Code of Federal Regulations* (CFR) 2.390.

A list of the six accident sequences that the staff evaluated is presented in Appendix B. The selected accident consequences vary in severity from high- to low-consequence events, and include accidents initiated by natural phenomena, operator error, and equipment failure. The most significant consequences are associated with the release of uranium hexafluoride and nuclear criticality.

The NRC staff concluded that the proposed Eagle Rock Enrichment Facility (EREF) design would reduce the risk (likelihood) of these accidents by using items relied on for safety (IROFS) and defense-in-depth measures. In addition, the facility Emergency Plan addresses these types of events. The NRC staff independently verified the applicant's accident analysis by performing confirmatory calculations and modeling. The NRC staff concluded that through the combination of plant design, passive- and active-engineered IROFS, administrative IROFS, and defense-in-depth features, the consequences of potential accidents at the proposed EREF will pose an acceptably low safety risk to workers, the public, and the environment.

APPENDIX C
COST ESTIMATE TO CONSTRUCT AND OPERATE

The U.S. Nuclear Regulatory Commission staff's review of the applicant's cost estimate to construct and operate the Eagle Rock Enrichment Facility is documented in Section 1.2.3.3.1 of this report. Appendix C contains the estimated costs of the two phases of construction which that have been marked by the applicant as "Proprietary Information," pursuant to Title 10 of the *Code of Federal Regulations* (CFR) 2.390.

The NRC staff finds that, based on the financial information submitted in the application and subsequent RAI response, AES meets the financial qualifications for the proposed activities in accordance with 10 CFR 70.23(a)(5).

APPENDIX D
HUMAN FACTORS

The U.S. Nuclear Regulatory Commission's (NRC) staff independently evaluated the human factors engineering. This evaluation is documented in Appendix D to this safety evaluation report. Appendix D contains information that has been marked by the applicant as "Security-Related Information," pursuant to Title 10 of the *Code of Federal Regulations* (CFR) 2.390.

The NRC staff reviewed general design criteria and applicable standards related to human-system interfaces for the proposed Eagle Rock Enrichment Facility (EREF). The applicant will establish and maintain a process to design and implement the human-system interfaces for the EREF. The staff concluded that the design and implementation of humans-system interface activities will ensure that the human-system interfaces perform their intended functions, thereby, meeting the regulatory requirements in 10 CFR 70.61(e).

APPENDIX E
ELECTRICAL SYSTEM AND INSTRUMENTATION AND CONTROL

The U.S. Nuclear Regulatory Commission (NRC) staff's review of the electrical system and instrumentation and controls (I&C) is documented in Appendix E to this safety evaluation report. Appendix E contains information that has been marked by the applicant as "Security-Related Information," pursuant to Title 10 of the *Code of Federal Regulations* (CFR) 2.390.

The staff reviewed the electrical system and I&C and their application to items relied on for safety (IROFS) needed to prevent or mitigate the identified accident sequences. This included reviewing how the design of these systems address the baseline design criteria, general design criteria, applicable Institute of Electrical and Electronics Engineers' standards, and applicable American National Standards Institute's standards. The proposed Eagle Rock Enrichment Facility (EREF) design reduces the risk (likelihood) of the accident by identifying IROFS and defense-in-depth features. The NRC staff concluded that through the combination of plant design, passive- and active-engineered IROFS, administrative IROFS, defense-in-depth features, and the application of management measures, electrical system and I&C at the proposed EREF will enable the facility to pose an acceptably low safety risk to workers, the public, and the environment. The staff reviewed the ISA summary and other information as it pertains to the electrical power system and I&C used as IROFS, and finds that it provides reasonable assurance that the applicant has identified appropriate IROFS and established engineered and administrative controls to ensure compliance with the performance requirements of 10 CFR 70.61.

APPENDIX F
STRUCTURAL DESIGN

The U.S. Nuclear Regulatory Commission (NRC) staff independently evaluated the protection of building structures against natural phenomena in the applicant's Integrated Safety Analysis Summary. This evaluation is documented in Appendix F to this safety evaluation report. Appendix F contains information that has been marked by the applicant as "Security-Related Information," pursuant to Title 10 of the *Code of Federal Regulations* (CFR) 2.390.

The NRC staff reviewed the structural design of the proposed facility. This included reviewing the building designs and structural design criteria, bases, and methodology. The proposed design criteria, bases, and methodology are based on industry-accepted codes, standards, and procedures. As a result, the staff determined that the applicant meets the requirements in 10 CFR 70.64(a)(2), 70.64(a)(4) and 70.64(b), as they apply to structural design.

APPENDIX G
NUCLEAR CRITICALITY SAFETY

The U.S. Nuclear Regulatory Commission (NRC) staff's Nuclear Criticality Safety (NCS) review of the applicant's Integrated Safety Analysis (ISA) Summary is documented in Appendix G to this safety evaluation report. Appendix G contains information that has been marked by the applicant as "Security-Related Information" and "Export Controlled Information," pursuant to Title 10 of the *Code of Federal Regulations* (CFR) 2.390 and 10 CFR 810, respectively.

The staff reviewed the ISA Summary and selected ISA documents related to NCS. Based on the review, the staff determined that the applicant's ISA met the applicable requirements of 10 CFR 70.70 through 70.65, as they apply to criticality safety.

APPENDIX H
MATERIAL CONTROL AND ACCOUNTING

The U.S. Nuclear Regulatory Commission (NRC) staff's review of the applicant's nuclear Material Control and Accountability (MC&A) program is documented in Appendix H to this safety evaluation report. Appendix H contains information that has been marked by the applicant as "Security-Related Information," pursuant to Title 10 of the *Code of Federal Regulations* (CFR) 2.390.

The NRC staff concluded that the applicant provided an acceptable Fundamental Nuclear Material Control (FNMC) Plan for the proposed facility that will meet the applicable 10 CFR Part 74 requirements. The FNMC Plan describes acceptable methods for achieving the performance objectives in 10 CFR 74.33(a), and the system capabilities of 10 CFR 74.33(c). As a result, the staff determined that the applicant meets the requirements in the area of MC&A to operate the proposed facility under 10 CFR Part 74.

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Same as above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report documents the U.S. Nuclear Regulatory Commission (NRC) staff's review and safety and safeguards evaluation of the AREVA Enrichment Services LLC (AES) application for a license to construct a gas centrifuge uranium enrichment facility and possess and use byproduct material, source material, and special nuclear material (SNM). The proposed facility is known as the Eagle Rock Enrichment Facility (EREF). AES proposes that the EREF be located in Bonneville County, Idaho, about 32 kilometers (20 miles) west northwest of the city of Idaho Falls. The EREF will possess natural, depleted, and enriched uranium, and will be authorized to enrich uranium up to a maximum of 5 weight percent uranium-235.

The objective of the NRC's review is to evaluate the facility's potential adverse impacts 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 management organization, administrative programs, and financial qualifications provided to ensure safe design and operation of the facility.

The NRC concludes, in this safety evaluation report, that the applicant's descriptions, specifications, 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.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT
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