AREVA

10 CFR 70.5

September 28, 2009

AES-O-NRC-09-00140-0

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

AREVA Enrichment Services LLC
Eagle Rock Enrichment Facility
NRC Docket No: 70-7015

Subject: Response to Requests for Additional Information – AREVA Enrichment Services LLC License Application for the Eagle Rock Enrichment Facility

On April 23, 2009, AREVA Enrichment Services LLC (AES) submitted a revised License Application to the U.S. Nuclear Regulatory Commission (NRC) to construct and operate the Eagle Rock Enrichment Facility (EREF) in Bonneville County, Idaho (Ref. 1).

On August 27, 2009, the NRC transmitted to AES Requests for Additional Information (RAI) regarding the EREF License Application (Ref. 2). AES hereby submits the responses to those NRC RAI.

Enclosure 2 provides the non-proprietary and non-security-related AES responses to the RAI and supporting information, including markups of the Safety Analysis Report and Environmental Report. This enclosure does not contain any proprietary, security-related sensitive unclassified non-safeguards information (SUNSI), or Export Control Information (ECI) as controlled under 10 CFR 810.

Enclosure 3 provides the proprietary and security-related AES responses to the RAI and supporting information, including markups to the Integrated Safety Analysis Summary, Emergency Plan, Physical Security Plan, Standard Practice Procedure Plan for the Protection of Classified Matter, Fundamental Nuclear Material Control Plan, Safety Analysis Report, and Environmental Report. This enclosure contains those portions of the submittal that contain either proprietary or SUNSI information that AES is requesting be withheld from public disclosure in accordance with 10 CFR 2.390. This information contains quotes from or markups of information that the NRC previously identified as SUNSI. Additional information was identified as SUNSI by AES using the guidance in NRC Regulatory Issue Summary (RIS) 2005-31, "Control of Security-Related Sensitive Unclassified Non-Safeguards Information Handled by Individuals, Firms, and Entities Subject to NRC Regulation of the Use of Source, Byproduct, and Special Nuclear Material." In accordance with 10 CFR 2.390(b), an affidavit supporting our request to withhold this proprietary and security-related information is enclosed.

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Enclosure 4 contains those markups of the Integrated Safety Analysis Summary that AES has identified as containing information that is deemed ECI under 10 CFR 810, "Assistance to foreign atomic energy activities" and other regulations. AES requests that this information be withheld from public disclosure in accordance with 10 CFR 2.390, "Public inspections, exemptions, request for withholding," paragraphs (a)(1) and (a)(3).

The AES responses include a description of each RAI, the AES response, markups of the EREF License Application as appropriate, additional supporting information (reports, figures, tables, etc.), and associated commitments.

As noted above, some AES responses contain proprietary information, SUNSI, or export control information that AES is requesting be withheld from public disclosure in accordance with 10 CFR 2.390. Enclosure 1 provides an affidavit supporting our request to withhold the information identified in Enclosures 3 and 4 in accordance with 10 CFR 2.390(b).

The EREF License Application will be revised to include the changes identified in the markups provided in Enclosures 2, 3, and 4 in Revision 2 of the EREF License Application.

If you have any questions regarding this submittal, please contact me at (508) 573-6554.

Respectfully,

Jim A. Kay
Licensing Manager

References:

1) S. Shakir (AES) Letter to the U.S. Nuclear Regulatory Commission, Revision 1 to License Application for the Eagle Rock Enrichment Facility, dated April 23, 2009.
2) B. Reilly (U.S. Nuclear Regulatory Commission) Letter to Jim Kay, Licensing Manager, Eagle Rock Enrichment Facility, AREVA Enrichment Services LLC, Request for Additional Information - AREVA Enrichment Services LLC License Application for the Eagle Rock Enrichment Facility, dated August 27, 2009.

Enclosures:

1) Affidavit of Jim Kay
2) Public Responses to NRC Requests for Additional Information and Supporting Information
3) Proprietary and Security-Related Responses to NRC Requests for Additional Information and Supporting Information to be withheld in accordance with 10 CFR 2.390
4) Export Control Information – Markups of Integrated Safety Analysis Summary

Commitments:
The EREF License Application will be revised to include the changes identified in the markups provided in Enclosures 2, 3, and 4 in Revision 2 of the EREF License Application.

Additional commitments are identified in the RAI responses contained in Enclosures 2 and 3.

CC:
Breeda Reilly, U.S. NRC Senior Project Manager
Gloria Kulesa, U.S. NRC Senior Project Manager

a) I am the Licensing Manager for the AREVA Enrichment Services LLC (AES), and as such have the responsibility of reviewing the proprietary and confidential information sought to be withheld from public disclosure in connection with our application to construct and operate a uranium enrichment facility. I am authorized to apply for the withholding of such proprietary and confidential information from public disclosure on behalf of AES.

b) I am making this affidavit in conformance with the provisions of 10 CFR 2.390 of the regulations of the Nuclear Regulatory Commission (NRC), and in conjunction with AES's request for withholding, which is accompanied by this affidavit.

c) I have knowledge of the criteria used by AES in designating information as proprietary or confidential.

d) By this submittal, AES seeks to protect from disclosure certain proprietary and confidential information, sensitive unclassified security-related information (SUNSI), and export control information contained in Enclosures 3 and 4.

   1. Security-Related Markups of the Integrated Safety Analysis Report, Emergency Plan, Physical Security Plan, Standard Practice Procedure Plan for the Protection of Classified Material, Fundamental Nuclear Material Control Plan, Safety Analysis Report, and Environmental Report.
   2. Proprietary Commercial Information provided in RAI Responses.
   3. Security-Related Sketches of the Basement Floor Slab
   4. Security-Related and Export Control Information – Markups of the Integrated Safety Analysis Report

   This affidavit discusses the bases for withholding certain portions of this submittal, as indicated therein, from public disclosure.

e) Pursuant to the provisions of 10 CFR 2.390(b)(4), the following is furnished for consideration by the NRC in determining whether the proprietary information sought to be protected should be withheld from public disclosure.

   1. For the sensitive unclassified security-related information (SUNSI) in Items 1, 2, 3, and 4 the information was identified as SUNSI by the NRC in the August 27, 2009, letter and by AES using the guidance in NRC Regulatory Issue Summary (RIS) 200531, "Control of Security-Related Sensitive Unclassified Non-Safeguards Information Handled by Individuals, Firms, and Entities Subject to NRC Regulation of the Use of Source, Byproduct, and Special Nuclear Material."
   2. For the proprietary items in Item 2 in Section (d), public disclosure of the proprietary information AES seeks to protect is likely to cause substantial harm to AES's competitive position within the meaning of 10 CFR 2.390(b)(4)(v). The proprietary information has substantial commercial value to AES.
   3. For Item 4 in Section (d), this information is deemed ECI under 10 CFR 810.
   4. Information for which protection from disclosure is sought has been held in confidence by AES. This information is proprietary to AES, and AES seeks to protect it as such.

6. The information sought to be withheld is being provided to the NRC in confidence, and, under the provisions of 10 CFR 2.390, it is to be received in confidence by the NRC.
7. The information sought to be withheld is not available in public sources, to the best of AES's knowledge and belief.

For all of the reasons discussed above, AES requests that the identified proprietary information be withheld from public disclosure.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on September 28, 2009.

J. A. Kay

Mr. Jim A. Kay
Licensing Manager of AES LLC
400 Donald Lynch Boulevard
Marlborough, MA 01752

Maureen L. Lyons
Notary Public

Enclosure 2
Public Responses to
Requests for Additional Information
and Supporting Information

Enclosure	Subject or Title
2.1	Public Responses to Requests for Additional Information • General Information • Financial Qualifications • Special Exemptions and Special Authorizations • Safety Programs • External Events and Structures • Human Factors Engineering • Radiation Protection • Nuclear Criticality Safety • Fire Safety • Decommissioning • Management Measures • Quality Assurance Program Description
2.2	Public Markup Pages of the EREF Safety Analysis Report
2.3	Public Markup Pages of the EREF Environmental Report
2.4	USGS Compiled Data Contour Map, USGS Compiled Data Grid Map, and ESRP Earthquake Activity Rate Calculations.
2.5	Human System Interface Design Implementation Plan

NRC RAI Number: GI-1

Text of NRC RAI:

Table 1.2-1

Table 1.2-1 provides information on the types of materials proposed for use. The applicant should provide total quantities of licensed materials to be possessed, including any calibration sources proposed to be used and proposed possession limits.

Regulations in 10 Code of Federal Regulations (10 CFR) 70.22(a)(4) require an applicant to identify the name, amount, and specification of the material proposed for use.

AES Response to NRC RAI:

AES expects that the quantities and types for the sealed and unsealed instrument calibration sources will be similar to those that the NRC approved for use at the National Enrichment Facility in Amendment 6 to Materials License SNM-2010 on March 14, 2008. However, as stated in SAR Section 1.2.3, these sources will be determined during the design phase.

SAR Section 1.2.3 states: "It is expected that other source materials and by-product materials will also be used for instrument calibration purposes. These materials will be identified during the design phase and the SAR will be revised, accordingly." AES will revise this section to state: "Other source materials and by-product materials will also be used for instrument calibration purposes. These 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. Subsequently, the SAR will be revised to incorporate the additional sources."

Associated EREF License Application Revisions:

SAR Section 1.2.3 will be revised as depicted in the markup provided in Enclosure 2.2.

Commitments:

The revision to SAR Section 1.2.3 depicted in the markup provided in Enclosure 2.2 will be incorporated in Revision 2 of the EREF License Application.

Attachments:

Enclosure 2.2: Markup to SAR Section 1.2.3

References:

M. D. Tschiltz (NRC) letter to S. Cowne (LES), "Review of Louisiana Energy Services' Request to Amend License Related to Possession of Byproduct Material, Decommissioning Financial Assurance and Liability Insurance, and Amendment 6 to License," dated March 14, 2008.

NRC RAI Number: GI-2

Text of NRC RAI:

Safety Analysis Report, Section 1.3.3.2, Annual Precipitation – Amounts and Forms

Provide analysis for determining the ground snow load (44.2 lb/ft^2) for the Eagle Rock Enrichment Facility (EREF).

This information is needed to assess whether the ground snow load estimated by the applicant is appropriate.

10 CFR 70.64(a)(2) requires the applicant to include adequate protection against natural phenomena in its design of the facility.

AES Response to NRC RAI:

The ground snow load at 44.2 lb/ft^2 for the EREF was determined as follows:

- Snow depth data (collected from the National Weather Service) at locations within close proximity to the EREF site that had a long period of record with similar climate conditions were used to compute L-moment parameters, were applied to Generalized Extreme Value (GEV) frequency distributions and used to estimate the 50-year return values.
- Snow course data, snow depth and snow water equivalent (SWE) (collected from the Natural Resources Conservation Service) at two locations near the EREF site that had a long period of record with similar climate conditions were used to create a snow depth/density relationship between snow depth and SWE.
- The 50-year snow depth data was converted into snow loads and spatially distributed to obtain the estimate at the EREF site. The ground snow load was estimated from spatial interpolation of the station snow load values.

SAR Section 1.3.3.2 will be revised to add a discussion on how the ground snow load at 44.2 lb/ft^2 for the EREF was determined.

A copy of the analysis for the ground snow load determination, "Site-Specific Snow Load Calculations for the AREVA Site in Idaho, USA", prepared by Applied Weather Associates and dated November 6, 2008, is available for review in AREVA's Bethesda, Maryland office.

Associated EREF License Application Revisions:

The EREF License Application will be revised as follows to incorporate this RAI response:

SAR Section 1.3.3.2 will be revised to add a discussion of how the ground snow load at 44.2 lb/ft^2 for the EREF was determined.

Commitments:

The EREF License Application will be revised to include the SAR markup in Revision 2 of the EREF License Application.

Attachments:

Enclosure 2.2 shows the markup to EREF SAR Section 1.3.3.2.

NRC RAI Number: GI-3

Text of NRC RAI:

Safety Analysis Report, Section 1.3.4.5, Design Basis Flood Events Used for Accident Analysis; Integrated Safety Analysis Summary Section 3.2.4.3, Floods; and Environmental Report Sections 3.4.12.2 and 3.4.12.3

Justify that a potential flood with the 1.0×10^{-5} annual probability is not a safety concern that needs to be included in the facility design and integrated safety analysis.

AREVA appears to exclude flooding as a potential external hazard from further consideration for facility design and in integrated safety analysis for the proposed Eagle Rock Enrichment Facility. AREVA based this determination on the fact that the proposed facility will be located above the Federal Emergency Management Agency estimated 100-year and 500-year flood elevations for the region. AREVA does not consider a potential flood with the 1.0×10^{-5} annual probability (highly unlikely probability defined by the applicant for its integrated safety analysis) in making its decision to exclude a flood hazard.

10 CFR 70.64(a)(2) requires the applicant to include adequate protection against natural phenomena in its design of the facility. In addition, 10 CFR 70.62(c)(iv) requires the applicant to conduct and maintain an integrated safety analysis that identifies potential accident sequences caused by credible external events.

AES Response to NRC RAI:

For the EREF two sources of flooding were evaluated: Flooding from nearby surface waters, including river flooding and upstream dam failures, and site flooding from extreme local precipitation, including roof ponding and local site flooding. [Security-related information withheld under 10 CFR 2.390].

Flooding from Nearby Surface Waters

SAR Section 1.3.4.5 and ER Section 3.4.12.3 note that the facility is not located near any reservoirs, levees, or surface waters that could cause flooding of the plant site. [Security-related information withheld under 10 CFR 2.390].

Site Flooding due to Extreme Local Precipitation

[Security-related information withheld under 10 CFR 2.390]

As discussed in SAR Section 1.3.4.5, no special design considerations for local extreme precipitation are necessary to prevent flooding at the proposed site other than stormwater runoff controls.

[Security-related information withheld under 10 CFR 2.390]

Associated EREF License Application Revisions:

The response to the RAI does not require any changes or additions to be made to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: FQ-1

Text of NRC RAI:

Provide the proposed financial plan for the construction and operation of EREF (i.e., the proposed percentages of debt and equity to be used in the financing of the project, a brief statement on any long-term contracts in place or under negotiation, a brief statement on which entity is considered the parent company with respect to financial qualifications).

The license application (in Section 1.2.2) states:

> There are financial qualifications to be met before a license can be issued. AES acknowledges the use of the following Commission-approved criteria as described in <u>Policy Issues Associated with the Licensing of a Uranium Facility; Issue 3, Financial Qualifications</u> (LES, 2002) in determining if the project is financially feasible:
>
> Construction of the facility shall not commence before funding is fully committed. Of this full funding (equity and debt), the applicant must have in place before constructing the associated capacity: (a) a minimum of equity contributions of 30% of project costs from the parents and (b) firm commitments ensuring funds for the remaining project costs.
>
> AES shall not proceed with the project unless it has in place long-term enrichment contracts (i.e., five years) with prices sufficient to cover both construction and operation costs, including a return on investment for the entire term of the contracts.

The application, however, does not provide the supporting basis for the staff to determine if the financial qualifications can be met.

10 CFR 70.22(a)(8) requires financial qualifications of the applicant "Where the nature of the proposed activities is such as to require consideration of the applicant's financial qualifications to engage in the proposed activities in accordance with the regulations in this chapter...."

AES Response to NRC RAI:

AES anticipates that its funding for various phases of construction may come from funds from operations, capital raised by itself, potential partners, and lending. Before initiating each phase of construction, AES will make its cost estimate for such a phase available, and document to NRC the source of funds available or committed to fund that phase; these conditions will be made part of the license.

Financial Qualifications

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. The AREVA SA is a corporation formed under the laws of France ("AREVA"), is governed by the Executive Board, and its principal owners are described in Section 1.2.1.2 of the EREF Safety Analysis Report (SAR). AREVA SA is required to file with the French financial market authorities AMF (Autorité des Marchés Financiers), in accordance with articles 211-1 to 211-42 of its General Regulation, a yearly Statutory Auditors' report, issued in French language, on the consolidated financial statements. A free translation into English of the Statutory Auditors' report on the consolidated financial statement is yearly issued by AREVA SA. The title of the Statutory Auditor's report translated into English is the Reference Document. The latest version was issued for year 2008 ending on December 31, 2008. Pursuant to European Regulation 1606/2002 of July 19, 2002, AREVA's consolidated financial statements for year 2008 were prepared in accordance with *International Financial Reporting Standards* (IFRS).

For the year ending December 31, 2008, AREVA had total assets of over €34.6 billion ($48.7 billion based on the Interbank exchange rate as of December 31, 2008), total revenue of more than €13 billion ($18 billion based on the Interbank exchange rate as of December 31, 2008), net income of €498 million ($702 million based on the Interbank exchange rate as of December 31, 2008), and cash and cash equivalents of €1.05 billion ($1.48 billion based on the Interbank exchange rate as of December 31, 2008). Total assets have been above €15 billion ($21 billion based on the Interbank exchange rate as of December 31, 2008) since 2005 and growing since then. Total yearly revenue has steadily grown from €10.1 billion ($11.9 billion based on the Interbank exchange rate as of December 31, 2005) since 2005 to reach €11.9 billion ($17.5 billion based on the Interbank exchange rate as of December 31, 2007) in 2007. Net income was €1,144 million ($1,355 million based on the Interbank exchange rate as of December 31, 2005) in 2005, €672 million ($887 million based on the Interbank exchange rate as of December 31, 2006) in 2006, €882 million ($1299 million based on the Interbank exchange rate as of December 31, 2007) in 2007. Cash and cash equivalents were between €634 ($750 million based on the Interbank exchange rate as of December 31, 2005) and €1,484 million ($2,185 million based on the Interbank exchange rate as of December 31, 2007) for the years 2005-2007 (Source: AREVA 2008 Reference Document, AREVA SA).

In addition to the Statutory Auditors' report and the Reference Document showing the financial strength of the AREVA group, AREVA has also a rating issued by Standard&Poor's. As stated in the latest Standard and Poor's report, issued on July 10, 2009, AREVA is affirmed at 'A-1' on short-term corporate credit rating and at 'A' on long-term corporate credit rating (Source: AREVA RatingsDirect®, Standard&Poor's, July 10, 2009).

Revised Funding Mechanism

SAR Section 1.2.2 will be revised to change the planned method for funding the construction and operation of the facility.

Pursuant to 10 CFR 70.23(a)(5), AES is required to demonstrate that it is financially qualified to carry out the activities proposed in its application. AES proposes to satisfy this obligation 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). That approach is as follows:

- 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.

- Operation of the EREF shall not commence until AES has in place either: (1) long term contracts lasting five years or more that provide sufficient funding for the estimated cost of operating the facility for the five year period; (2) documentation of the availability of one or more alternative sources of funds that provide sufficient funding for the estimated cost of operating the facility for five years; or (3) some combination of (1) and (2).

Investment Structure

Investment in the Eagle Rock Enrichment Facility (EREF) at 6.6 MSWU will be divided in at least two-phases as follows:

- First phase at 3.3 MSWU
- Second phase of up to 3.3 MSWU

The first phase may be broken down into smaller increments based on calendar time, construction phase, etc. Depending on market conditions the second 3.3 MSWU may be constructed in one or more phases in sync with market demand.

First Phase Investment

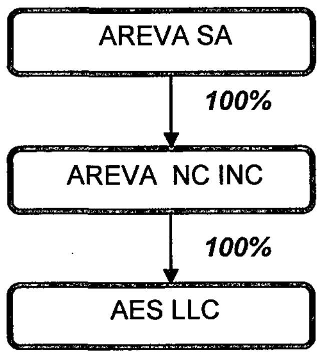
The total estimated project cost for the first phase, including inflation and interest, is $3.2 billion including inflation and financing cost. AES expects to finance the project cost of the first phase through:

- equity contribution from its parent company,
- a commitment from the parent company to provide the necessary funds for the project
- debt,
- self-generated cash during the production rampup.

The current anticipated estimated sources and uses are as follows:

[The breakdown of estimated sources and uses has been withheld as proprietary information in accordance with 10 CFR 2.390.]

Note: AREVA NC SA is a wholly owned AREVA SA subsidiary, and is the parent company of AREVA NC Inc. AREVA NC Inc is the parent company of AES LLC.



Off-Take Agreements

[The information regarding the off-take agreements has been withheld as proprietary information in accordance with 10 CFR 2.390.]

Second Phase Investment

In case that the market conditions are favorable to the expansion, AES may launch the construction of the second phase and contract at the same time. The project cost of the second phase is estimated at [Information withheld in accordance with 10 CFR 2.390], including inflation and financing cost. In such a case, AES expects to finance the construction of the second phase with (i) its self-generated cash during the production rampup for at least 30% of the project cost, i.e. at least [Information withheld in accordance with 10 CFR 2.390], and (ii) with additional debt to cover the remaining project cost.

In total, the equity is expected to be at about 30% of the total project cost, equity stemming from three different sources: (i) equity contribution from AES parents; (ii) equity commitment from AES parents; and (ii) cash flow from AES operation.

Associated EREF License Application Revisions:

SAR Section 1.2.2 is modified to as depicted in Enclosure 2.2.

Commitments:

The markup of SAR Section 1.2.2 provided in Enclosure 2.2 will be incorporated into Revision 2 of the EREF License Application.

Attachments:

Enclosure 2.2: Markup of SAR Section 1.2.2

NRC RAI Number: SE-1

Text of NRC RAI:

In Section 10.2.1, the Safety Analysis Report states: "Since, AES intends to sequentially install and operate the Separations Building Modules over time, financial assurance for decommissioning will be provided during the operating life of the EREF at a rate that is in proportion to the decommissioning liability for these facilities as they are phased in. Similarly, AES will provide decommissioning funding assurance for disposition of depleted tails at a rate in proportion to the amount of accumulated tails onsite up to the maximum amount of the tails as described in Section 10.3, Tails Disposition. An exemption request to permit this incremental financial assurance is provided in Section 1.2.5, 'Special Exemptions or Special Authorizations.'"

[1] Provide a schedule for the phase in of construction and operations, including projected dates for: commencement of operations of the first module; commencement of operations of subsequent modules; first receipt of licensed material; subsequent receipt(s) of licensed material; the initial generation of depleted uranium tails; and the accumulation of tails.

[2] Provide a schedule for submitting the initial and subsequent executed financial assurance instruments in terms of the phases of the construction and operation as well as a breakdown of the percentage of the full decommissioning funding corresponding to each phase of the construction and operations. The cost estimate for each phase must cover the full decontamination and decommission costs for that phase plus the 25 percent contingency factor. Confirm that the total decommissioning cost estimate contains the 25 percent contingency factor for each phase.

[3] In Section 1.2.5, AES "commits to updating the decommissioning cost estimates on an annual forward looking incremental basis and to providing the U.S. Nuclear Regulatory Commission (NRC) revised funding instruments that reflect these projections of depleted uranium tails production." Until the facility is at full operation, confirm that the decommissioning funding estimates and revised funding instruments would be provided annually on a forward-looking basis to reflect the aggregate cost of any facility module that would be currently in operation; that has been in operation and has not been fully decontaminated and decommissioned as approved by NRC, or would be in operation within the next 12 months; to reflect the accumulated depleted uranium tailings onsite and a projection of the amount that would be onsite within the next year; and to include the 25 percent contingency factor.

AES Response to NRC RAI:

[1] The milestones for the project to license, construct and operate the proposed Eagle Rock Enrichment Facility (EREF) are provided in Section 1.0 of the Environmental Report (ER), page 1.0-1. Based on these milestones, the preliminary schedule for the phase in of construction and operations is as follows:

Initiate Facility Construction — February 2011

Centrifuge Assembly Building (CAB)

- Delivery of UF_6 (less than or equal to 20 kgU) as testing material: April 2013

First Separations Building Module (SBM):

- First Delivery of Natural UF_6 (more than 50 kgU) as feed material — October 2013
- Start First Cascade — February 2014
- Initial Generation of Depleted Uranium Tails — February 2014

Second SBM:

- First Delivery of Natural UF_6 (more than 50 kgU) as feed material — October 2015
- Start First Cascade — February 2016
- Initial Generation of Depleted Uranium Tails — February 2016

Third SBM:

- First Delivery of Natural UF_6 (more than 50 kgU) as feed material — October 2017
- Start First Cascade — February 2018
- Initial Generation of Depleted Uranium Tails — February 2018

Fourth SBM:

- First Delivery of Natural UF_6 (more than 50 kgU) as feed material — October 2019
- Start First Cascade — February 2020
- Initial Generation of Depleted Uranium Tails — February 2020

Complete Construction — February 2022
Achieve Full Nominal Production Output — March 2022

The accumulation of Tails during ramp-up is given in Table 10.3-1 of the EREF Safety Analysis Report (SAR).

[2] AES will provide to the NRC final executed copies of the reviewed financial assurance instruments at least 21 days prior to:

- The receipt of UF_6 for testing material for the CAB. The Decommissioning Funding Plan update shall cover the decontamination and decommissioning of the CAB and all other plant areas where the testing material is used. The amount of the Decommissioning

Funding Plan, associated with this phase, is $12,782,500 (2007 U.S. dollars), including the 25 percent contingency factor, or 3% of the full Decommissioning Cost.

- The receipt of Natural UF_6 as feed material for the first SBM. The Decommissioning Funding Plan update shall cover the decontamination and decommissioning of the first SBM and all other plant areas where feed material is used. The amount of the Decommissioning Funding Plan, associated with this phase, is $131,098,750 (2007 U.S. dollars), including the 25 percent contingency factor, or 29% of the full Decommissioning Cost.

- The receipt of Natural UF_6 as feed material for the second SBM. The Decommissioning Funding Plan update shall cover the decontamination and decommissioning of the second SBM and all other plant areas where feed material is used. The amount of the Decommissioning Funding Plan, associated with this phase, is $235,875,000 (2007 U.S. dollars), including the 25 percent contingency factor, or 53% of the full Decommissioning Cost.

- The receipt of Natural UF_6 as feed material for the third SBM. The Decommissioning Funding Plan update shall cover the decontamination and decommissioning of the third SBM and all other plant areas where feed material is used. The amount of the Decommissioning Funding Plan, associated with this phase, is $340,651,250 (2007 U.S. dollars), including the 25 percent contingency factor, or 76% of the full Decommissioning Cost.

- The receipt of Natural UF_6 as feed material for the fourth SBM. The Decommissioning Funding Plan update shall cover the decontamination and decommissioning of the fourth SBM and all other plant areas where feed material is used. The amount of the Decommissioning Funding Plan, associated with this phase, is $445,427,500 (2007 U.S. dollars), including the 25 percent contingency factor, or 100% of the full Decommissioning Cost.

The details of the calculation of the Decommissioning Funding Plan update are described in the Table here below:

		Unit Decommission Cost		Cumulative Decommission Cost		
		Reference Cost (2007 U.S. Dollars)	Including 25% contingency (2007 U.S. Dollars)	Reference Cost (2007 U.S. Dollars)	Including 25% contingency (2007 U.S. Dollars)	Percentage of total decommissioning Funding plan
CAB and other plant areas	Testing Material	10,226,000 (note 1)	12,782,500	10,226,000	12,782,500	3%
First SBM and other plant areas	Feed Material	94,653,000 (note 2)	118,316,250	104,879,000	131,098,750	29%
Second SBM and other plant areas	Feed Material	83,821,000 (note 3)	104,776,250	188,700,000	235,875,000	53%
Third SBM and other plant areas	Feed Material	83,821,000 (note 3)	104,776,250	272,521,000	340,651,250	76%
Fourth SBM and other plant areas	Feed Material	83,821,000 (note 3)	104,776,250	356,342,000	445,427,500	100%

Note 1: referred as Cost to Decommission Other Buildings in Table 10.1-14 of the EREF SAR

Note 2: given in Table 10.1-16 of the EREF SAR

Note 3: from Table 10.1-14 and Table 10.1-16 of the EREF SAR, we can determine that the cost to decommission the second, third and fourth SBM is equal to $346,116,000 (total decommissioning cost) minus $10,226,000 for the CAB and $94,653,000 for the first SBM, i.e. $251,463,000. Thus, the unit decommissioning cost for SBM 2, 3, or 4 is $83,821,000 (i.e., $251,463,000 divided by 3).

AES confirms that the total decommissioning cost estimate contains the 25% contingency factor for each phase.

[3] Until the facility is at full operation, AES confirms that the decommissioning funding estimates and revised funding instruments would be provided annually on a forward-looking basis to reflect the aggregate cost of any facility module that would be currently in operation; that has been in operation and has not been fully decontaminated and decommissioned as approved by NRC, or would be in operation within the next 12 months; to reflect the accumulated depleted uranium tailings onsite and a projection of the amount that would be onsite within the next year; and to include the 25 percent contingency factor.

Associated EREF License Application Revisions:

The response to the RAI does not require any changes or additions to be made to the EREF License Application.

Commitments:

1) AES will provide to the NRC final executed copies of the reviewed financial assurance instruments at least 21 days prior to:

   - The receipt of UF_6 as testing material for the CAB. The Decommissioning Funding Plan update shall cover the decontamination and decommissioning of the CAB and all other plant areas where the testing material is used.

   - The receipt of Natural UF_6 as feed material for the first SBM. The Decommissioning Funding Plan update shall cover the decontamination and decommissioning of the first SBM and all other plant areas where feed material is used.

   - The receipt of Natural UF_6 as feed material for the second SBM. The Decommissioning Funding Plan update shall cover the decontamination and decommissioning of the second SBM and all other plant areas where feed material is used.

   - The receipt of Natural UF_6 as feed material for the third SBM. The Decommissioning Funding Plan update shall cover the decontamination and decommissioning of the third SBM and all other plant areas where feed material is used.

   - The receipt of Natural UF_6 as feed material for the fourth SBM. The Decommissioning Funding Plan update shall cover the decontamination and decommissioning of the fourth SBM and all other plant areas where feed material is used.

2) Until the facility is at full operation, AES confirms that the decommissioning funding estimates and revised funding instruments would be provided annually on a forward-looking basis to reflect the aggregate cost of any facility module that would be currently in operation; that has been in operation and has not been fully decontaminated and decommissioned as approved by NRC, or would be in operation within the next 12 months; to reflect the

accumulated depleted uranium tailings onsite and a projection of the amount that would be onsite within the next year; and to include the 25 percent contingency factor.

Attachments:

None

NRC RAI Number: SP-1

Text of NRC RAI:

Safety Analysis Report, Section 3.3.4, Structural Design Criteria, and Integrated Safety Analysis Summary Section 3.2.3.4.4, Extreme Precipitation

Define normal roof design live load for Separations Building Modules; Blending, Sampling, and Preparation Building; Cylinder Receipt and Shipping Building; and Technical Support Building.

Qualitative design criteria are discussed in both the Safety Analysis Report and Integrated Safety Analysis Summary. AREVA uses the term normal roof design live load. However, AREVA does not indicate what the roof design live load is. 10 CFR 70.64(a)(2) requires the applicant to include adequate protection against natural phenomena in its design of the facility. In addition, 10 CFR 70.62(c)(iv) requires the applicant to conduct and maintain an integrated safety analysis that identifies potential accident sequences caused by credible external events.

AES Response to NRC RAI:

The normal roof design live load definition and corrections to the SAR and ISAS are provided.

Normal Roof Design Live Load Definition

The minimum normal roof design live load to be applied to the IROFS structures [security-related information withheld under 10 CFR 2.390] *represents the maintenance activities* required to access roof ventilation equipment. The value of this load may vary for each building depending on the activities performed and will be determined in accordance with the International Building Code (Section 1607) and ASCE 7-05 (Chapter 4.0 and Table 4-1). The codes specify a uniform load of 40 psf and a concentrated load of 300 pounds. However, any unique specific loads that are larger than the code standard load values will be incorporated into the roof design.

Corrections to the SAR and ISAS

In EREF SAR Section 3.3.4, the term "normal roof design live load" was improperly used. When describing the roof design for extreme local precipitation, the documents should state that the loads associated with the extreme local precipitation shall not exceed the design load (or capacity) of the roof and not the "normal roof design live load".

SAR Section 3.3.4, 10th bullet will be revised to state "The roof drainage systems (including secondary roof drainage path) will be designed such that the amount of rainfall that can collect on the roof does not exceed the design load for the roof."

In EREF ISA Summary Section 3.3.2.2.6, [security-related information withheld under 10 CFR 2.390]

ISA Summary Section 3.3.2.2.6, 3rd paragraph, will be revised to state [security-related information withheld under 10 CFR 2.390]

Associated EREF License Application Revisions:

The EREF License Application will be revised as follows to incorporate this RAI response:

SAR Section 3.3.4, 10th bullet will be revised to state "The roof drainage systems (including secondary roof drainage path) will be designed such that the amount of rainfall that can collect on the roof does not exceed the design load for the roof."

ISA Summary Section 3.3.2.2.6, 3rd paragraph, will be revised to state [security-related information withheld under 10 CFR 2.390]

Commitments:

The EREF License Application will be revised to include the SAR and ISA Summary markups in Revision 2 of the EREF License Application.

Attachments:

Enclosure 2.2 shows the markup to EREF SAR Section 3.3.4 and Enclosure 3.2 shows the ISA Summary Section 3.3.2.2.6.

NRC RAI Number: SP-2

Text of NRC RAI:

<u>Safety Analysis Report, Section 3.3.4, Structural Design Criteria, and Integrated Safety Analysis Summary Section 3.2.3.4.4, Extreme Precipitation</u>

Provide numerical value for the Extreme Environmental Rainfall.

SAR Section 3.3.4 indicates that roofs will be designed to avoid water ponding due to extreme local precipitation to a depth exceeding the extreme environmental rainfall. As stated in the ISA Summary Section 3.2.3.4.4, there are three types (1, 24, and 48 hours) of all-season extreme local precipitation hazards for the annual probability of 1.0×10^{-5}. It is not clear which of the three all-season extreme local precipitations is the extreme environmental rainfall.

10 CFR 70.64(a)(2) requires the applicant to include adequate protection against natural phenomena in its design of the facility.

AES Response to NRC RAI:

The numerical value for the Extreme Environmental Rainfall is the 24-hour extreme local precipitation estimate of 112 mm (4.39 in). The 24-hour also accounts for the peak 1-hour extreme local precipitation estimate of 60 mm (2.37 in). Roof ponding from extreme local precipitation, EE-LOCAL PRECIP-ROOFS, is addressed in ISA Summary Table 3.7.4, page 12 of 34. As discussed in this table, [security-related information withheld under 10 CFR 2.390].

As stated in SAR Section 3.3.4, roofs will be designed so that the depth of ponded water during the extreme local precipitation does not exceed the Extreme Environmental Rainfall. SAR Section 3.3.4 will be revised to add a statement that the Extreme Environmental Rainfall is equivalent to the 24-hr all season extreme local precipitation estimate of 112 mm (4.39 in).

As stated in ISA Summary Section 3.3.2.2.6, [security-related information withheld under 10 CFR 2.390].

Associated EREF License Application Revisions:

The EREF License Application will be revised as follows to incorporate this RAI response:

SAR Section 3.3.4 will be revised to add a statement that the Extreme Environmental Rainfall is equivalent to the 24-hr all season extreme local precipitation estimate of 112 mm (4.39 in).

ISA Summary Section 3.3.2.2.6 will be revised to add a statement that [security-related information withheld under 10 CFR 2.390].

Commitments:

The EREF License Application will be revised to include the SAR and ISAS markups in Revision 2 of the EREF License Application.

Attachments:

Enclosure 2.2 shows the markup to SAR Section 3.3.4. Enclosure 3.2 shows the markup to ISAS Section 3.3.2.2.6.

NRC RAI Number: SP-3

Text of NRC RAI:

Integrated Safety Analysis Summary, Section 3.2.3.4.4, Extreme Precipitation

Provide technical basis for the 1, 24, and 48 hours all-season extreme local precipitation estimates.

This information is needed to assess whether the extreme load precipitations the applicant estimated are appropriate.

10 CFR 70.64(a)(2) requires the applicant to include adequate protection against credible natural phenomena in its design of the facility. In addition, 10 CFR 70.62(c)(iv) requires the applicant to conduct and maintain an integrated safety analysis that identifies potential accident sequences caused by credible external events.

AES Response to NRC RAI:

As noted in ISA Summary Section 3.2.3.4.4, [security-related information withheld under 10 CFR 2.390].

The 2-year, 5-year, 10-year, 25-year, 50-year, and 100-year 24-hour and 48-hour precipitation values for the EREF site coordinates were extracted from National Oceanic and Atmospheric Administration (NOAA) Atlas 2 Volume 5, *Precipitation Frequency Atlas of the Western United States (1973)*, and Technical Paper 49, *Two-to-Ten-Day Precipitation for Return Periods of 2 to 100 Years in the Contiguous United States (1964)*. The 2-year and 100-year 6-hour and 24-hour precipitation values from NOAA Atlas 2 Volume 5 were also extracted. This information was used to calculate the 2-year and 100-year 1-hour return values. NOAA Atlas 2 was then utilized to determine the 5-year, 10-year, 25-year and 50-year 1-hour precipitation values. The National Weather Service's least squares regression procedure was applied to extrapolate the 10,000-year and 100,000-year return frequencies for 1-hour, 24-hour and 48-hour durations.

This is the same approach reviewed by the NRC for USEC in NUREG 1851, "Safety Evaluation Report for the American Centrifuge Plant in Piketon, Ohio," September 2006.

A copy of the analysis for the extreme precipitation determination, "Calculated 10^{-4} and 10^{-5} Return Frequencies for the 1 Hour, 24 Hour and 48 hour Durations at the AREVA Site in Idaho, USA", prepared by Applied Weather Associates and dated July 24, 2008, is available for review in AREVA's Bethesda, Maryland office.

ISA Summary Section 3.2.3.4.4 will be revised to [security-related information withheld under 10 CFR 2.390].

Associated EREF License Application Revisions:

The EREF License Application will be revised as follows to incorporate this RAI response:

ISA Summary Section 3.2.3.4.4 will be revised to [security-related information withheld under 10 CFR 2.390]. ISA Summary Section 3.2.9 will be revised to [security-related information withheld under 10 CFR 2.390].

Commitments:

The EREF License Application will be revised to include the ISA Summary markups in Revision 2 of the EREF License Application.

Attachments:

Enclosure 3.2 shows the markups to ISAS Sections 3.2.3.4.4 and 3.2.9.

NRC RAI Number: SP-4

Text of NRC RAI:

Integrated Safety Analysis Summary, Section 3.2.6.1, Probabilistic Seismic Hazard Analysis Results

Provide a description of the methodology and the resulting analyses used to develop scaled earthquake time histories.

The information is needed to assess the applicant's seismic design.

10 CFR 70.64(a)(2) requires the applicant to include adequate protection against credible natural phenomena in its design of the facility. In addition, 10 CFR 70.62(c)(iv) requires the applicant to conduct and maintain an integrated safety analysis that identifies potential accident sequences caused by credible external events.

AES Response to NRC RAI:

Methodology

Results of the site-specific probabilistic seismic hazard assessment for the EREF site include uniform hazard response spectra (UHRS) determined for annual probabilities of exceedance that range from $2.1x10^{-3}$ to 10^{-5}. The "design response spectrum" for the project will be developed from the site-specific UHRS using a performance based method described in Regulatory Guide 1.208 (US NRC March, 2008) and in ASCE/SEI 43-05 (American Society of Engineers (2005). Certain engineering analyses could require acceleration time histories that closely match the earthquake motion characteristics represented in the frequency domain by the "design response spectrum."

[Security-related information withheld under 10 CFR 2.390].

Seismic hazard at the EREF site is influenced by its central location within a considerably less seismically active region of the Eastern Snake River Plain (ESRP) with exposure to more distant but more seismically active regions of the "Yellowstone Parabola (YP)" (Petersen et al., 2008) located west, north and east of the EREF site.

[Security-related information withheld under 10 CFR 2.390].

AES will develop response spectra using these time histories to verify that the generated response spectra envelope the DBE Ground Response Spectra (developed by ASCE 43-05 methodology).

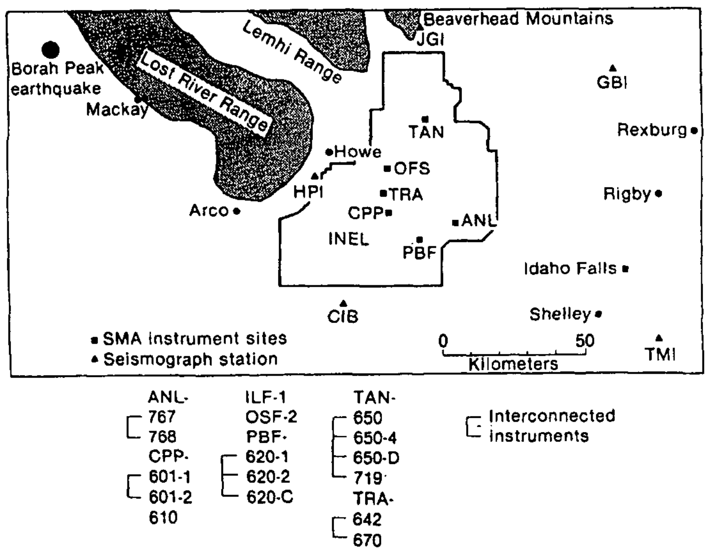


Figure 1. Map of the Idaho National Engineering Laboratory (INEL) relative to the epicenter of the Borah Peak earthquake showing the location of strong-motion accelerographs with notation of interconnected instruments and the location of the seismograph stations.

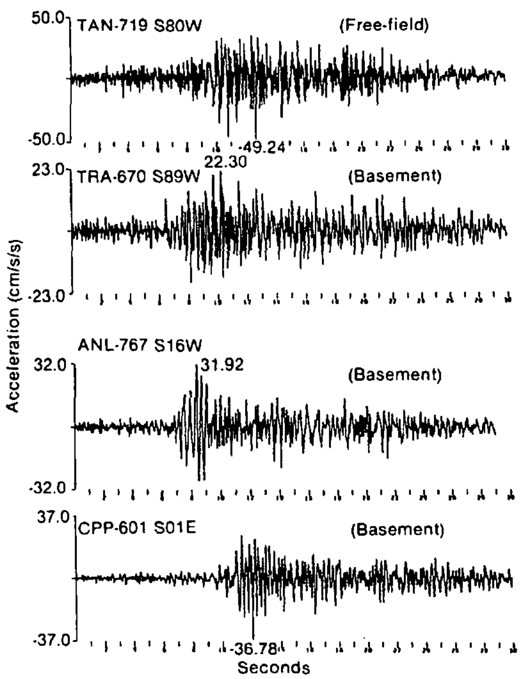


Figure 2. INEL acceleration time-histories from the Borah Peak earthquake displaying the similarity in an envelope of peak amplitudes for TRA, ANL, and CPP.

__Resulting Analyses__

While the resulting analyses from the time history analyses have not been developed at this time, these analyses will be available for the NRC to conduct the operability readiness and management measures verification review to verify that management measures that ensure compliance with the performance requirements of 10 CFR 70.61 have been implemented and confirm that the facility has been constructed and will be operated safely and in accordance with the requirements of the license.

Similar to the Materials Licenses for the National Enrichment Facility and the USEC Facility, AES expects that the NRC will include a license condition in the Materials License that states:

"Introduction of UF6 into any module of the EREF shall not occur until the Commission completes an operational readiness and management measures verification review to verify that the management measures that ensure compliance with the performance requirements of 10CFR 70.61 have been implemented and confirms that the facility has been constructed and will be operated safely and in accordance with the requirements of the license. The licensee shall provide the Commission with 120 days advance notice of its plan to introduce UF6 in any module of the EREF."

Associated EREF License Application Revisions:

A new ISA Summary Section 3.2.6.4 titled "Earthquake Time History Development" will incorporate this methodology.

Commitments:

The EREF License Application will be revised to include the ISA Summary markup in Revision 2 of the EREF License Application.

Attachments:

Enclosure 3.2 shows the markup to EREF ISA Summary Section 3.2.6.

References:

American Society of Civil Engineers (2005), Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities, ASCE/SEI 43-05, 81p.

Gasparini, D. and E.H. Vanmarcke (1976), Simulated Earthquake Motions Compatible with Prescribed Response Spectra, M.I.T. Department of Civil Engineering Research Report R76-4, Order No. 527, January 1976.

Jackson, Suzette M. and J. Boatwright (1983), The Borah Peak Earthquake of October 28, 1983 – Strong Ground Motion, in Earthquake Spectra, Vol. 2, No. 1, 1985, pp. 51-69.

Silva, W.J. and K. Lee (1987), WES RASCAL Code for Synthesizing Earthquake Ground Motions, Miscellaneous Paper S-73-1, Report No. 24 in the Series, "State-of-the-Art for Assessing Earthquake Hazards in the United States, US Army Engineers Waterways Station, Vicksburg, MS.

U.S. Nuclear Regulatory Commission (March, 2007), A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion, Regulatory Guide 1.208.

NRC RAI Number: SP-5

Text of NRC RAI:

Integrated Safety Analysis Summary, Section 3.2.6.3, Selection of the Design-Basis Earthquake

Provide a calculation showing how the methodologies in ASCE 43-05 will be implemented to determine the design basis earthquake and to demonstrate compliance with the target seismic performance goals.

The information is needed to assess the applicant's seismic design and integrated safety analysis results.

10 CFR 70.64(a)(2) requires the applicant to include adequate protection against credible natural phenomena in its design of the facility. In addition, 10 CFR 70.62(c)(iv) requires the applicant to conduct and maintain an integrated safety analysis that identifies potential accident sequences caused by credible external events.

AES Response to NRC RAI:

NTS Calculation E000-00-S-CAL-0001, "Seismic Ground Design Response Spectra: Design Basis Earthquakes for Structures (DBE-B) and for UF_6 Process Piping and Systems (DBE-P)", Revision 00 is available for the NRC in the AES Bethesda, Maryland office.

This calculation documents the development of the horizontal and vertical Design Ground Response Spectra based on adjusting the initial Uniform Hazard Response Spectras in accordance with the methodologies outlined in ASCE 43-05, "Standard Seismic Design Criteria", 2005.

Associated EREF License Application Revisions:

The response to the RAI does not require any changes or additions to be made to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: SP-6

Text of NRC RAI:

Integrated Safety Analysis Summary, Section 3.2.8, Site-Specific Volcanic Hazard Analysis

Provide technical basis to characterize the potential hazards from ash eruptions of Cascade Range volcanoes.

The information is needed to assess the potential roof loads from a Cascade Range eruption.

10 CFR 70.64(a)(2) requires the applicant to include adequate protection against credible natural phenomena in its design of the facility. In addition, 10 CFR 70.62(c)(iv) requires the applicant to conduct and maintain an integrated safety analysis that identifies potential accident sequences caused by credible external events.

AES Response to NRC RAI:

Maximum ash thickness deposited at the EREF from a Cascade Range Eruption

The maximum ash thickness that could be deposited at the EREF from future Cascade tephra eruptions is less than 8 cm, assuming a hypothetical eruption from the nearest Cascade volcano, a maximum credible eruptive volume and explosivity, and the dispersal of ash directly toward the EREF. Therefore, the maximum ash thickness that could be deposited on building roofs at the EREF from future Cascade tephra eruptions is less than 8 cm.

Potential roof loads from a Cascade Range Eruption

Blong (1981, 1984) discusses the effects of ash loading on structures. Snow loading can be used as an analog when making calculations and comparisons. Assume an 8 cm thickness of ash (upper bound; maximum credible ash fall at the site). Assume it has the dry- and wet-ash densities of the Mount St Helens May 18 1980 ash, those being 0.5 g/cc dry and 1.25g /cc wet (Shipley and Sarna-Wojcicki, 1983). The load of the ash (8 cm) would range from 4 g/cm^2 (dry) to 10 g/cm^2 (wet) – that is, the load of ash would range from 8.2 lb/ft^2 (dry) to 20.5 lb/ft^2 (wet).

For the EREF, the design basis extreme environmental snow load was developed by combining the normal ground snow load of 44.2 lb/ft^2 (216 kg/m^2) with the additional surcharge from an extreme winter precipitation event. The extreme winter precipitation event results in a load of 19 lb/ft^2 (93 kg/m^2). The EREF extreme environmental snow load is 63.2 lb/ft^2 (309 kg/m^2). *The load of ash would range from 8.2 lb/ft^2 (40 kg/m^2) (dry) to 20.5 lb/ft^2 (100 kg/m^2) (wet), which is below both the extreme environmental snow load and normal ground snow load for the EREF.*

Determination of maximum ash thickness deposited at the EREF from a Cascade Range Eruption

The Volcanism Working Group (VWG) (1990; Chapter 5) reviewed published literature, conducted interviews of field researchers in eastern Idaho, and compiled information related to air-fall ash potential at the Idaho National Laboratory (INL), a site immediately west of the proposed Eagle Rock Enrichment Facility. The VWG addressed the 21 most likely sites for ash-fall producing eruptions in the western United States, including all of the Cascade Range volcanoes. The U.S. Geological Survey subsequently published hazard assessments for the five active Cascade volcanoes in Washington, including Mt Adams (Scott, et al., 1995), Mt Baker (Gardner, et al., 1995), Glacier Peak (Waitt, et al., 1995), Mt Rainier (Hoblitt, et al., 1995), Mount St Helens (Wolfe and Pierson, 1995), and the Medicine Lake volcano in northern California (Donnelly-Nolan, et al., 2007). These investigations provide additional detail on eruption probabilities and the distribution of near-vent volcanic products. However, they do not alter the fundamental conclusions of the VWG (1990), which found that prehistoric Cascade tephra deposits on the Eastern Snake River Plain (ESRP) do not exceed 5 cm in thickness. Apparently, the Cascade volcanoes are sufficiently far from the INL (675 to 790 km) or located such that the prevailing westerly winds aloft did not disperse ash directly toward the INL. It was further concluded that the maximum ash thickness that could be deposited at the INL from future Cascade tephra eruptions is less than 8 cm, assuming a hypothetical eruption from the nearest Cascade volcano, a maximum credible eruptive volume and explosivity, and the dispersal of ash directly toward the INL. Because of the proximity of the EREF to INL, the maximum ash thickness that could be deposited at the EREF from future Cascade tephra eruptions is also estimated to be less than 8 cm.

The supporting technical basis for these conclusions is summarized as follows.

1. Empirical curves of compacted air-fall ash thickness vs. distance for the largest late Pleistocene and Holocene Cascade eruptions (Miller, 1989, Figure 3; Hoblitt, et al., 1987, Figure 3-1) include the Mazama (Crater Lake, ca. 8,800 years B.P., 40 km^3); Glacier Peak G (ca. 11,000 – 12,000 years B.P.); Mt St Helens Yn (3,300 – 4,000 years B.P., 3 km^3); and Mt St Helens May 18, 1980 (uncompacted, 1 km^3) tephras. Assuming the most adverse conditions of closest proximity (a hypothetical major eruption of Newberry Volcano, Oregon, 675 km from the INL), ash dispersed directly toward the EREF, and an eruption magnitude of Mt Mazama 8,800 years B.P. (the largest known eruption from a Cascade volcano), the thickness vs. distance data show that about 6 cm of tephra would be deposited at the EREF.

2. The observed thicknesses of 13 late Pleistocene and Holocene compacted air-fall tephras at field localities on or near the ESRP range from 0.5 to 5 cm, and most are less than 2 cm thick (Volcanism Working Group, 1990; Table 7). Blong (1984) suggests that the initial uncompacted thicknesses of such tephras may have been up to twice as great. These deposits include tephras from the largest eruptions of Cascade volcanoes listed in item 2 above. The observed thicknesses of compacted Mazama ash (the largest known Cascade eruption) at five localities on or near the ESRP range from 0.5 to 3 cm.

Associated EREF License Application Revisions:

The EREF License Application will be revised as follows to incorporate this RAI response:

SAR Section 1.3.3.2 will be revised to provide a comparison of the ash load to snow load for the EREF. A new Appendix D.1, Maximum Ash Thickness Deposited at the EREF from a Cascade Range Eruption, will be added to the ER.

ISA Summary Section 3.2.3.3.1 (new) and Tables 3.7-3 (page 1 of 10) and 3.7-4 (page 1 of 34), will be revised to [security-related information withheld under 10 CFR 2.390].

Commitments:

The EREF License Application will be revised to include the markups in Revision 2 of the EREF License Application.

Attachments:

Enclosure 3.2 shows the markups to the ISA Summary Section 3.2.3.3.1 (new) and Tables 3.7-3 (page 1 of 10) and 3.7-4 (page 1 of 34), and Enclosure 2.2 shows the markup to SAR Section 1.3.3.2.

Enclosures 2.3 and 3.2 shows the new Appendix D.1, Maximum Ash Thickness Deposited at the EREF from a Cascade Range Eruption, will be added to the ER and ISA Summary, respectively.

References:

Blong, R.J., 1984, Volcanic Hazards – a Sourcebook on the Effects of Eruptions: Academic Press, New York, 424 p.

Blong, R.J., 1981, Some effects of tephra falls on buildings. In: Tephra Studies, Self, S. and Sparks, R.S.J., editors, Reidel Publishing Co., p. 405-420.

Donnelly-Nolan, J.M., Nathenson, M., Champion D.E., Ramsey, D.W., Lowenstern, J.B., and Ewert, J.W., 2007, Volcano hazards assessment for Medicine Lake Volcano, northern California: U.S. Geological Survey Scientific Investigations Report 2007–5174-A, 33 p., 1 plate.

Hoblitt, R.P, Miller, C.D., and Scott, W.E., 1987, Volcanic hazards with regard to siting nuclear-power plants in the Pacific Northwest: U.S. Geological Survey Open-File Report 87-297, 196 p.

Hoblitt, R.P., Wilder, J.S., Driedger, C.L., Scott, K.M., Pringle, P.T., and Vallance, J.W., 1995, Volcano hazards from Mount Rainier, Washington: U.S. Geological Survey Open-File Report 95-273, 10 p., 1 plate.

Miller, C.D., 1989, Potential hazards from future volcanic eruptions in California: U.S. Geological Survey Bulletin 1847, 17 p., 2 plates.

Scott 1995, Volcano hazards in the Mount Adams region, Washington: U.S. Geological Survey Open-File Report 95-492, 11 p., plates.

Shipley, S., and Sarna-Wojcicki, A.M., 1983, Distribution, thickness and mass of tephra from volcanoes – Pacific Northwest United States: Assessment of hazards to nuclear reactors. U.S. Geological Survey Misc Field Studies Map MF-1435, 27 p.

Volcanism Working Group, 1990, Assessment of potential volcanic hazards for New Production Reactor site at the Idaho National Engineering Laboratory: EG&G Informal Report, EGG-NPR-10624, 98 p. [Original document identifier is UCRL-ID-104722, Technical Information Department, Lawrence Livermore National Laboratory, University of California, Livermore, CA 94551.]

Waitt, R.B., Mastin, Larry, and Beget, J.E., 1995, Volcanic-hazard zonation for Glacier Peak Volcano, Washington: U.S. Geological Survey Open-File Report 95-499, 9 p., 2 plates.

Warrick, R.A., Anderson, J., Dowing, T., Lyons, J., Ressler, J., Warrick, M., and Warrick, T., 1981, Four communities under ash, after Mount St Helens: Program on Tech Envir and Man, Monograph 34, Instutute of Behavioral Science, University of Colorado, Boulder, CO, 143 p.

Wolfe, E.W., and Pierson, T.C., 1995, Volcanic-hazard zonation for Mount St Helens, Washington: U.S. Geological Survey Open-File Report 95-497, 12 p., 2 plates.

NRC RAI Number: SP-7

Text of NRC RAI:

Integrated Safety Analysis Summary and Environmental Report Appendix F, Probabilistic Seismic Hazard Assessment

Provide a complete list of the reference earthquake catalogs used to compile the list of earthquakes in Appendix F, Table 3 that the applicant used in the probabilistic seismic hazard assessment.

The information is needed to assess the results of the applicant's probabilistic seismic hazard assessment.

10 CFR 70.64(a)(2) requires the applicant to include adequate protection against credible natural phenomena in its design of the facility. In addition, 10 CFR 70.62(c)(iv) requires the applicant to conduct and maintain an integrated safety analysis that identifies potential accident sequences caused by credible external events.

AES Response to NRC RAI:

List of the Reference Earthquake Catalogs

The list of earthquakes in Appendix F, Table 3 represents a catalog of earthquakes of Magnitude 5.0 and larger that was assembled from the following two catalogs. Assembling earthquake parameters from the following two sources created the "project earthquake catalog" and Table 3 represents the subset of magnitude 5.0 and larger earthquakes.

1. Advanced National Seismic System (ANSS) (http://www.ncedc.org/anss/catalog-search.html)

2. Significant U.S Earthquakes (1568 – 1989) (http://neic.usgs.gov/neis/epic/epic_rect.html)

   The project earthquake catalog assembled from the above sources was compared to other regional earthquake lists. These additional lists include:

   a. Petersen et. al. (2008) USGS Open File 2008-1128 – Western U.S. Catalog #2 with duplicates deleted and with dependent and man-made events included (http://earthquake.usgs.gov/research/hazmaps/products_data/2008/catalogs/)
   b. Earthquake History of Idaho, USGS (http://earthquake.usgs.gov/regional/states/idaho/history.php)
   c. Current and Historical Earthquake Activity, Idaho National Laboratory (https://inlportal.inl.gov/portal/server.pt?open=514&objID=4311&parentname=Gateway&parentid=None&mode=2&in_hi_userid=2&cached=true)

Additional Reference Information

1. **Advanced National Seismic System:**

The ANSS catalog represents a compilation of earthquake data recorded by and contributed to the ANSS database by 15 regional networks operated by the USGS and its partnering networks. For the study region in the inter-mountain Western U.S., the contributing networks include (1) the University of Utah Seismograph Network, (2) the University of Reno Seismic Lab, and (3) the Montana Bureau of Mines and Geology. The seismic detection capability for this region is specified that earthquakes of magnitude 2.0 and larger can be located by the combined resources of the regional networks.

(http://earthquake.usgs.gov/research/monitoring/anss/regions/imw/#capabilities)

The ANSS Catalog includes data entries spanning the time period from 1898 to the Present. The catalog is updated in real time as new earthquakes are recorded by the regional networks and location information is transmitted to ANSS.

The ANSS database was searched using the following parameters to obtain an earthquake list of magnitude 3.0 and larger earthquakes for a rectangular block surrounding a 200-mile radius of the EREF site. The search parameters include:

- catalog=ANSS
- start_time=1898/01/01,00:00:00
- end_time=2007/11/01,00:00:00
- minimum_latitude=40.5
- maximum_latitude=46.5
- minimum_longitude=-116.6
- maximum_longitude=-108.5
- minimum_magnitude=3.0
- maximum_magnitude=9.0
- minimum_depth=0.
- maximum_depth=200.
- event_type=E

The resulting catalog included 2,843 earthquakes for the time period from August 30, 1962 through October 31, 2007.

An additional list of earthquakes of magnitudes 1.0 to 3.0 was obtained using a similar regional search of the ANSS database. This produced a list of more than 40,000 earthquakes. This information was used only to illustrate epicenter locations on a map, and not for analytical purposes.

2. Significant U.S Earthquakes (1568 – 1989):

The ANSS Catalog did not contain earthquake information for the study region prior to 1962. The Catalog of Significant U.S. Earthquakes was searched using the search parameters listed below to obtain information on historical earthquakes.

- FILE CREATED: Fri Jul 25 22:07:28 2008
- Geographic Grid Search Earthquakes= 166
- Latitude: 47.000N - 40.000N
- Longitude: 108.000W - 117.000W
- Catalog Used: USHIS
- Data Selection: Significant U.S. Earthquakes (USHIS)

The database search resulted in 166 earthquakes located in the study region that dated from October 1871 through July 1989.

3. **Petersen et. al. (2008) USGS Open File 2008-1128**

The earthquake catalog assembled by the USGS for preparation of the 2008 National Seismic Hazard Maps (USGS Open File Report 2008-1128) was compared to the project earthquake catalog assembled from sources 1. and 2. above. The catalogs agree with respect to the list of earthquakes of magnitude 5.0 and larger, e.g. Appendix F Table 3. Small differences in magnitude of "tenths" of a magnitude unit exist for certain earthquakes, but these differences are attributed to magnitudes being reported using various magnitude scales, mb, ML, or MW. Processing of the earthquake catalogs currently involves conversion to a common scale, moment magnitude, MW, in order to apply the catalogs in a seismic hazard assessment using current ground motion prediction equations that scale ground motion amplitudes relative to the MW magnitude scale. The catalogs were compared for earthquakes reported for the Eastern Snake River Plain. It is noted that the USGS Western US catalog has a lower bound magnitude threshold of 4.0 MW. A comparison of the project earthquake catalog (sources 1. and 2.) with the USGS WUS catalog showed an agreement on the 2 earthquakes of magnitude 4 and larger that are known for the Eastern Snake River Plain. These include the November 11, 1905 "Shoshone" earthquake and an un-named earthquake on August 18, 1964. Both the project catalog and the USGS WUS catalog list identical location coordinates for the Shoshone earthquake. The project catalog lists the 1905 earthquake as 5.7 ML, the USGS reports the 1905 earthquake as 5.5 MW. Also, the 1964 earthquake is reported in both the project catalog and USGS WUS catalog with identical location coordinates and identical magnitude of 4.2.

The USGS WUS catalog was not used as a contributing catalog to the project catalog due to its low magnitude threshold of 4.0. This version of the USGS WUS catalog has been processed by the USGS to remove duplicate entries that result from merging data from several catalogs. Included in the catalog are dependent events (i.e. foreshocks and aftershocks) as well as man-made events including mining explosions and mine collapse events.

4. **Earthquake History of Idaho, USGS**

Following is a description of historical seismicity of Idaho including bordering states. This information was used to verify that all important earthquakes known for Idaho were accounted for in the compiled project earthquake catalog.
http://earthquake.usgs.gov/regional/states/idaho/history.php

Idaho

Earthquake History

The first earthquake causing damage in Idaho's earthquake history occurred on November 9, 1884, apparently centering in northern Utah. Six shocks were reported felt at Paris, Idaho, causing considerable damage to houses. People suffered from nausea.

A shock on November 11, 1905, was felt in the southern half of Idaho and parts of Utah and Oregon. At Shoshone, Idaho, walls cracked and plaster fell.

On May 12, 1916, Boise was hit by a shock which wrecked chimneys and caused people to rush into the streets. Reclamation ditches were damaged and the flow of natural gas altered. It was felt at Loon Creek, 120 miles northeast, and in eastern Oregon - an area of 50,000 square miles.

An intensity VII earthquake occurred within the State on July 12, 1944. The Seafoam Ranger Station building shook so hard the occupants thought it was coming apart. Several people reported that the shaking was so violent they were unable to walk. Another observer reported that rocks rose at least a foot in the air and looked like a series of explosions up the hill. Part of the canyon wall collapsed near Lime Creek. Cracks opened 100 yards long in Duffield Canyon and cracks one to three inches across and several hundred yards long opened on the road below Seafoam. Two chimneys fell at Cascade. This shock was felt over 70,000 square miles, including all of central Idaho, and parts of Washington, Oregon, and Montana.

The magnitude 7.1 earthquake at Hebgen Lake, Montana, on August 17, 1959, which killed 28 people, formed "Quake Lake," and did $11 million damage to roads and timber, also caused some damage in Idaho. Intensity VII was experienced in the Henry's Lake, Big Springs, and Island Park areas. Big Springs increased its flow 15 percent and became rusty red colored. A man was knocked down at Edward's Lodge. There was considerable damage to building in the Henry's Lake area. Trees swayed violently, breaking some roots, and cars jumped up and down. Chimneys fell and a 7-foot-thick rock-and-concrete dock cracked.

In the Island Park area chimneys were toppled and wells remained muddy for weeks. At Mack's Inn, a small girl was thrown from bed and hysteria occurred among some guests. Dishes were broken.

An intensity VII earthquake occurred on August 30, 1962, in the Cache Valley area of Utah. Two large areas of land totaling four acres, five feet thick, slid 300 yards downhill at Fairview, Idaho, opening new springs. Plaster walls, and chimneys were cracked and a chimney fell at Franklin. Falling brick at the Franklin School cracked through the roof and plaster was cracked in every room. Additional damage occurred at Preston. This magnitude 5.7 earthquake was felt over an area of 65,000 square miles in five states and cause approximately $1 million in damage.

An intensity VI shock, on November 1, 1942, centered near Sand Point and affected 25,000 square miles of Washington, Montana, and Idaho. The Northern Pacific Railroad partially suspended operations to inspect the right of way for boulders and slides. Church services were interrupted, but only minor damage was reported by homes.

A February 13, 1945, shock near Clayton, felt over a 60,000 square mile area, broke some dishes at Idaho City and cracked plaster at Weisner.

A locally sharp shock was felt at Wallace on December 18, 1957, damaging the Galena Silver Mine and frightening miners working 3,400 feet underground.

Soda Springs was shaken by a shock on August 7, 1960, which cracked plaster and a concrete foundation. It was only felt over a 900 square mile area.

Two intensity VI shocks were reported in 1963. The first on January 27, was felt over 6,000 square miles and centered near Clayton, where plaster and windows were cracked. Large boulders rolled down the hill near Camp Livingston and aftershocks were felt for a week. The second occurred on September 10 and was a magnitude 4.1 shock. It caused minor damage at Redfish Lake. Thunderous earth noises were heard.

A magnitude 4.9 shock on April 26, 1969, cracked a foundation at Ketchum, plaster at Livingston Mills, and a cement floor at Warm Springs. It was felt over 9,000 square miles.

Abridged from Earthquake Information Bulletin, Volume 4, Number 2, March - April 1972.

5. **Current and Historical Earthquake Activity, Idaho National Laboratory**

The following is a description of historical seismicity in the region of the Idaho National Laboratory that is posted on the INL web page. This information was used to verify that all important earthquakes known for Idaho were accounted for in the compiled project earthquake catalog.

https://inlportal.inl.gov/portal/server.pt?open=514&objID=4342&parentname=CommunityPage&parentid=6&mode=2&in_hi_userid=2&cached=true

Historical Earthquake Activity

The historical catalog from 1872 to 2007 shows earthquakes of magnitude (M) 2.0 and greater occurred primarily in the Basin and Range (mountains and valleys). This dramatically contrasts the small number of earthquakes that have occurred in the Snake River Plain. Earthquakes occur along normal faults that formed the Basin and Range as a result of the tensional stresses in the earth's crust. This process has occurred for about 16 million years and is still active today. Two large historic earthquakes, the 1983 surface-wave magnitude (Ms) 7.3 Borah Peak, Idaho and 1959 Ms 7.5 Hebgen Lake, Montana earthquakes ruptured along such normal faults (where the mountains move up and the valleys drop down). Since crustal extension in this region occurs slowly over millions of years, repeat times of earthquakes with M > 7.0 are on the order of thousands of years. For example, the earthquake that occurred prior to the 1983 Borah Peak earthquake occurred 5,000 to 7,000 years earlier.

The historical earthquake catalog also shows that the Snake River Plain is seismically quiet (or aseismic) relative to the surrounding active Basin and Range region. The largest earthquake that may have occurred within the Snake River Plain is the 1905 M 5.7 Shoshone earthquake. The November 11, 1905, Shoshone earthquake occurred before there was instrumental monitoring in Idaho and, because its location was based on felt reports, it may have an error of 100 km or more. Modified Mercalli Intensity (MMI) zones that were assigned based on damage reports documented at the time of the earthquake indicate the epicenter may be to the south of the Snake River Plain. Because the official epicenter is located within the Snake River Plain, INL seismic design criteria include ground motion contributions from an earthquake similar in size to 1905 Shoshone earthquake occurring at INL.

Associated EREF License Application Revisions:

The response to the RAI does not require any changes or additions to be made to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: SP-8

Text of NRC RAI:

Environmental Report, Section 3.3.7, Seismic Hazard Assessment, and Appendix F, Probabilistic Seismic Hazard Assessment

Verify the tentatively worded conclusions in the Environmental Report and Appendix F that explain why the applicant's seismic hazard assessment results in smaller ground motion amplitudes than the ground motions the U.S. Geological Survey predicted in the 2008 U.S. National Seismic Hazard Maps.

The information is needed to assess the results of the applicant's probabilistic seismic hazard analysis.

10 CFR 70.64(a)(2) requires the applicant to include adequate protection against credible natural phenomena in its design of the facility. In addition, 10 CFR 70.62(c)(iv) requires the applicant to conduct and maintain an integrated safety analysis that identifies potential accident sequences caused by credible external events.

AES Response to NRC RAI:

A. Background

Probabilistic seismic hazard is determined for a specific site location using a complex analysis of the three basic input models listed below.

1. Geometries of regional seismic sources including area sources and fault sources that describe the geographic locales of seismic activity.
2. Annual earthquake recurrence rates as a function of magnitude for seismic source zones and faults defined in Item 1.
3. Ground motion prediction equations that describe ground motion amplitudes as functions of frequency given occurrence of an earthquake of a specific magnitude and located at a specific distance from a site underlain with a known surface geologic condition (e.g. soil, bedrock, or known V_{S30} value).

The probabilistic seismic hazard assessment is conducted using information that is affected by two principal types of uncertainty. These include *epistemic* uncertainty that results from incomplete knowledge or understanding of geologic and seismologic processes in the specific geographic region being studied, and *aleatory* uncertainty that results from random processes. *Epistemic* uncertainties are accommodated in the probabilistic seismic hazard assessment by using multiple hypotheses to represent the range of credible models for Items 1 through 3, above. *Aleatory* uncertainty, or randomness, in ground motion predictions is accommodated by specifying the median ground motion prediction as well as the standard error (σ "sigma") determined for each ground motion prediction equation. In addition, random uncertainty is accommodated by specifying the number of standard errors (e.g. 1, 2, 3, or unbounded) that will be analyzed in the seismic hazard calculations.

Parametric variations of each of the three basic input models are analyzed to yield a number of individual seismic hazard assessments (i.e. seismic hazard curves), each of which is the result of a specific combination of varied input models. A final hazard assessment can be obtained by using statistical processing of the individual seismic hazard curves to determine mean or median hazard, or hazard at a higher statistical fractile (e.g. 85^{th} percentile). Also, a final hazard assessment can be obtained using a logic-tree system in which each individual parametric variation is assigned a weight and application of the weights yields the final seismic hazard assessment.

Results of the site-specific seismic hazard assessment performed for the EREF site were compared to new seismic hazard estimates released by the US Geological Survey in Open File Report 2008-1128 (Petersen et al., 2008). Information included in OF 2008-1128 (Reference 1) was used to qualitatively describe several reasons why the site-specific hazard results were 30% to 40% lower than new hazard results published by the USGS.

Following are the qualitative reasons that were cited in the site-specific study that were listed as possible causes for the seismic hazard differences.

> "The difference in seismic hazard estimates can result from the following possible causes."

- The site-specific PSHA used ground motion models for normal slip fault mechanisms; the USGS possibly used various fault mechanisms, or unspecified fault mechanisms, which predict higher amplitude seismic ground motions.
- The weighted result for the site-specific PSHA includes hazard results for hard bedrock attenuation models, which leads to lower amplitude seismic ground motions. The USGS 2008 results are for the NEHRP B-C Boundary site condition which is a firm bedrock condition that results in higher amplitude seismic ground motions relative to hard bedrock site conditions.
- The site-specific PSHA used a local earthquake frequency model determined for the ESRP; the USGS possibly used a larger background seismicity model for the Basin and Range province and a local cell earthquake activity rate that could exceed the historical earthquake rate (Petersen et al., 2008).

B. <u>Verification of Site Specific Ground Motion Amplitudes</u>

Previously unavailable information has been recently obtained. The new information includes gridded earthquake activity rates that were only recently posted to the USGS web page dedicated to the 2008 update of the National Seismic Hazard Maps and is provided in this response to further support the three bulleted explanations above for the USGS (2008) hazard estimates being higher than that determined in the site-specific probabilistic seismic hazard assessment.

1. Ground Motion Prediction Equations

The site-specific probabilistic seismic hazard study for the EREF site used ground motion prediction equations published by Spudich et al. (1999) and by Boore and Atkinson (2008). Each of these equations was applied to predict ground motion for earthquake activity associated with normal faulting. The EREF site is located within the intermountain west tectonic zone that is characterized by extensional tectonism with seismic activity being the result of crustal movements predominantly on normal faults.

Spudich et al. (1999, p. 1161) conclude that "extensional regime ground motions are systematically smaller than non-extensional regime motions." Their equations are applicable only to normal faulting tectonic settings. Boore and Atkinson (2008) provide equations to predict ground motions based on fault mechanism. Following are comparisons of ground motion amplitudes for normal vs. strike-slip faulting using equations from Boore and Atkinson (2008).

Magnitude, M_W	Distance, km	Fault Type	Peak Ground Acceleration, g	Spectral Acceleration, g, 5 Hz
7.0	100.0	Normal	0.023	0.055
7.0	100.0	Strike-Slip	0.030	0.066

Boore and Atkinson (2008) illustrate that ground motions are increased by 30% for peak ground acceleration and by 20% for spectral acceleration at 5 Hz for strike-slip vs. normal faults for a magnitude 7 earthquake located at a distance of 100 km.

The USGS used three equally-weighted ground motion prediction equations for their 2008 update of hazard maps for the extensional tectonic region of the Western United States. These include Boore and Atkinson (2008), Campbell and Bozorgnia (2008), and Chiou and Youngs (2008). Petersen et al. (2008, Figure 12.) illustrate that each of the three ground motion equations was applied with equal weight of 0.5 for strike-slip and normal fault motion predictions. The USGS (2008) hazard assessment covers a much broader geographic region of the WUS than the site-specific study performed for the EREF site. This larger region was modeled to include ground motion resulting equally from normal and strike-slip fault movements. USGS (2008) hazard estimates thus are higher than those determined in the site-specific study due to the inclusion of ground motions predicted for strike-slip faults, which produce higher ground motions. Results in the table above for the Boore and Atkinson (2008) models illustrate that ground motion could increase by 20% to 30% if ground motion was attributed exclusively to strike-slip faults. Ground motion variations by fault type attributed to the other two USGS (2008) ground motion models were not determined, but are believed to also predict lower ground motions for normal faulting earthquakes relative to strike-slip faulting. It is therefore estimated that the increase in USGS (2008) seismic hazard relative to the EREF site-specific study, attributed to usage of both strike-slip and normal faults, is about 10% to 15% based on examination of predictions by the Boore and Atkinson (2008) models.

2. Site Conditions for Ground Motion Predictions

The site-specific seismic hazard assessment was determined for site conditions in the upper 30 meters of the earth's crust (i.e. V_{S30}) that included 620 m/sec (2,034 ft/sec) for the Spudich et al. (1999) models, and that ranged from 760 m/sec (2,493 ft/sec) to 1,300 m/sec (4,265 ft/sec) for the Boore and Atkinson (2008) ground motion models. This range of site conditions brackets the one site condition used by the USGS (2008) hazard assessment; that being the NEHRP Site Class B-C Boundary characterized by a V_{S30} of 760 m/sec. It was estimated in the site-specific study using shear wave velocity measurements made at the INL that the actual V_{S30} site condition at EREF was closer to the 1,300 m/sec than to the USGS (2008) B-C site condition of 760 m/sec. The USGS converts all ground motion models that make hard bedrock predictions (e.g. Site Class A, V_{S30} of 1,500 m/sec) to the B-C Boundary site condition using the following factors (Petersen et al., 2008, p. 18).

> *"For several of these models, we used frequency-dependent factors to convert from hard rock (NEHRP site class A) to firm rock (NEHRP site class BC). These factors are: 1.52 for peak ground acceleration, 1.74 for 0.1-s, 1.76 for 0.2-s, 1.72 for 0.3-s, 1.58 for 0.5-s, 1.34 for 1.0-s, and 1.20 for 2.0-s spectral acceleration (see Frankel and others, 1996)."*

Ground motion amplitudes at hard rock sites range from about 60% to 80% of amplitudes predicted for Site Class B-C. Seismic hazard results in the site-specific study were weighted as 75% for Site Class B-C and 25% to a hard bedrock site condition just below the Site Class A condition (e.g. 1,300 m/sec vs. 1,500 m/sec). This weighting used in the site-specific study would slightly decrease the hazard estimate relative to the USGS (2008) which is based on only the lower velocity (V_{S30} of 760 m/sec) B-C Boundary Site Class.

3. Earthquake Activity Rates

Differences in the seismic hazard between the EREF site-specific study and the USGS (2008) study were attributed to possible differences in earthquake activity rates determined for the regions surrounding the EREF site. USGS (2008) input files (include 10^a grids) were not available until recently to make a comparison with activity rates applied in the USGS (2008) hazard assessments. The input files are provided in the USGS website:
http://earthquake.usgs.gov/research/hazmaps/products_data/2008/software/

These technical input data files defined the geometries of the USGS (2008) background sources for the Eastern Snake River Plain and the Yellowstone Parabola, and the gridded and smoothed earthquake activity rates for the region. Petersen et al. (2008, p.21) describe the method for determining activity rates for "Uniform Background Zones" which include the Eastern Snake River Plain and the Yellowstone Parabola (see below). The earthquake activity rates are incremental per grid cell per year, for a magnitude increment of 0.1 mag unit (so 10^a is the rate of earthquakes with magnitude between -0.05 and 0.05) and a grid cell size of 0.1 x 0.1 degree (so the area of the grid cell varies with latitude). A uniform b value of 0.8 was applied for this region.

Uniform Background Zones (Petersen et al. 2008, p.21)

> In contrast to the gridded-(smoothed-) seismicity model, regional background zones account for earthquake potential spread uniformly across tectonic or geologic regions with constant geologic or strain characteristics. These zones are designed to provide a hazard floor and account for future random earthquakes in areas with little or no historical seismicity. The earthquake rate for each WUS background zone is determined by counting earthquakes with Mw≥4 since 1963, computing an annualized rate, and prorating this rate uniformly across the entire zone. As in the 1996 and 2002 maps, we model background seismicity (WUS Model 2) in five non-overlapping regional zones: the Basin and Range Province extended to include the Rio Grande rift, parts of Arizona and New Mexico, western Texas, eastern Washington, and northern Montana and Idaho; the Cascade volcanic province; the Snake River Plain province; the Yellowstone seismicity parabola province; and a region of southeastern California and southwestern Arizona (figs. 14 and 15). These regions are geologically and seismologically distinct, and the reasoning behind the zonation is discussed in detail in the 1996 documentation. As in 1996 and 2002, the regional zone model (Model 2) is implemented in a way that does not penalize areas of high seismicity in order to provide a hazard floor in areas of low seismicity. In each grid cell, the historical gridded-seismicity rate (Model 1) is compared with the floor value from Model 2. If the historical rate exceeds the floor value, the final cell rate simply equals the historical rate. If, however, the floor value exceeds the historical rate, Models 1 and 2 are combined with respective weights 0.67 and 0.33 to calculate the final cell rate. Nowhere is the final cell rate less than the historical rate, and the total modeled seismicity rate in the WUS exceeds the total historical rate by about 16 percent.

The information described above and from OF 2008-1128 (Reference 1) was synthesized into a regional map in order to make a comparison of earthquake activity rates applied in the EREF site-specific study and in the USGS (2008) seismic hazard update for the Western United States.

Figure SP-8-1 (Enclosure 2.4) shows a map prepared using the following USGS (2008, 2009) information.

1. USGS (2008) WUS Earthquake Catalog, $M_W > 4$, duplicates, clustered events eliminated
2. USGS (2009) Gridded a-values, file agrd.out.txt
3. USGS (2009) Coordinates for Eastern Snake River Plain
4. USGS (2009) Coordinates for the Yellowstone Parabola

Figure SP-8-1 (Enclosure 2.4) includes a contour map calculated from the data file (Item 2.) of USGS (2008) gridded a-values. The agrd.out.txt includes earthquake activity rates for cell sizes of 0.1^{O} Latitude and 0.1^{O} Longitude. The rate of earthquakes of magnitude -0.5 to 0.5 centered at magnitude 0 is determined as 10^{agrd}. The rate of earthquakes at higher magnitudes is determined using the uniform b-value of 0.8 used in USGS (2008) throughout the region shown on Figure SP-8-1. Figure SP-8-2 (Enclosure 2.4) shows the gridded a-value locations with annotated a-values. Gridded a-values for the EREF site region range from 0.034 to 0.058 with a value of 0.0406 interpolated for the EREF site location. The grid cell area for the EREF site region is 89.7 sq. km. Figure SP-8-2 illustrates that the gridded a-values range from a low value of 0.015 about 40 km northwest of the EREF site near the western boundary of the ESRP with the Yellowstone Parabola zone to a high value of 0.182 about 40 km southeast of the EREF site near the contact with the eastern branch of the Yellowstone Parabola.

Earthquake activity rates predicted for the Eastern Snake River Plain by the range of USGS (2008) gridded a-values and associated uniform b-value of 0.8 are compared to earthquake activity rates determined for the ESRP in the site-specific hazard study. The calculation procedure used for this comparison is to normalize gridded a-values to the entire area of the ESRP and to compare predicted annual rates of magnitude 4.0 and larger earthquakes, and also for magnitude 5.0 and larger earthquakes. Calculation sheets are provided in Enclosure 2.4.

The following table summarizes calculations of earthquake activity rates for the ESRP provided in Enclosure 2.4.

Model	a-value 33,264 km^2	b-value	Annual Rate Mw ≥ 4.0	Annual Rate Mw ≥ 5.0	Return Period Mw ≥ 4.0	Return Period Mw ≥ 5.0
Site-Specific	2.931	0.945	0.142	0.016	7.1	62.2
USGS (2008) min a-value	2.584	0.80	0.242	0.038	4.1	26.1
USGS (2008) value at EREF	2.61	0.80	0.257	0.041	3.9	24.6
USGS (2008) max a-value	2.751	0.80	0.356	0.056	2.8	17.7

Earthquake activity rates compared in the above table for the ESRP illustrate that USGS (2008) activity rates for magnitude 4.0 and larger earthquakes are greater than predictions made by the site-specific model by factors that range from 1.7 (minimum gridded a-values) to 2.5 (maximum gridded a-values). Differences increase for larger magnitudes due to the less-steep b-value of 0.8 applied for this region by the USGS. This difference in local activity rates for the seismic zone containing the EREF site is a direct cause for the higher ground motion estimates determined by the USGS (2008) hazard assessment.

Predictions of earthquake activity rates by the site-specific and USGS (2008) recurrence models are compared to the Western US earthquake catalog (Petersen et al., 2008). Figure SP-8-1 illustrates 2 earthquakes of magnitude 4 and larger located within the ESRP. These include the M=5.5 Shoshone earthquake located at the southwest end of the ESRP, and an earthquake of magnitude 4.1 that occurred in 1964 near the eastern boundary with the

Yellowstone Parabola zone. Both the site-specific and USGS (2008) recurrence models overestimate the number of earthquakes actually observed in the ESRP. For example, the USGS (2008) models estimate the return period of magnitude 4 and larger earthquakes to be in the range of 3 to 4 years; therefore about 11 to 15 earthquakes of this size would be expected to have occurred in the 45-year duration of the earthquake catalog. The site-specific recurrence model predicts about 6 occurrences of magnitude 4 and larger earthquakes since 1963. Similarly, the USGS (2008) models predict about 4 or 5 occurrences of magnitude 5 and larger earthquakes during a 100-year time frame, compared to 1 to 2 for the site-specific model. The only known earthquake of this size attributed to the ESRP is the 1905 historical Shoshone earthquake with uncertain location, and possibly not located in the ESRP (INL, 2008).

Comparisons shown above indicate conservative representations of seismic activity rates specified for the ESRP zone which is characterized by a very low seismicity rate. The USGS (2008) earthquake activity rate estimates are substantially more conservative than the site-specific model, most likely due to their procedure of combining gridded and smoothed historical activity rates as described in the Uniform Background Zones section of Petersen et al. (2008) presented at the top of this response.

Associated EREF License Application Revisions:

This supporting documentation for the ground motion amplitudes for the EREF PSHA will be documented in (new) Appendix G for the ER. The reference to (new) Appendix G will be added to ER Section 3.3.7.

This supporting documentation for the ground motion amplitudes for the EREF PSHA will be documented in (new) Appendix G for the ISA Summary. The reference to (new) Appendix G will be added to Appendix F of the ISA Summary.

Commitments:

The EREF License Application will be revised to include the ER and ISA Summary markups in Revision 2 of the EREF License Application.

Attachments:

Enclosure 2.3 shows the markup to EREF ER Section 3.3.7 and Appendix F of the ER and Enclosure 3.2 shows the markup to Appendix F of the ISA Summary.

Enclosures 2.3 and 3.2 show the (new) Appendix G to the ER and ISA Summary.

Enclosure 2.4 includes the following:

1. USGS Compiled Data Contour Map
2. USGS Compiled Data Grid Map
3. ESRP Earthquake Activity Rate Calculations

References:

1. Petersen, Mark D., Frankel, Arthur D., Harmsen, Stephen C., Mueller, Charles S., Haller, Kathleen M., Wheeler, Russell L., Wesson, Robert L., Zeng, Yuehua, Boyd, Oliver S., Perkins, David M., Luco, Nicolas, Field, Edward H., Wills, Chris J., and Rukstales, Kenneth S., 2008, Documentation for the 2008 Update of the United States National Seismic Hazard Maps: U.S. Geological Survey Open-File Report 2008–1128, 61 p.

NRC RAI Number: SP-9

Text of NRC RAI:

Integrated Safety Analysis Summary, Section 3.3.1.1, Separations Building Modules

Provide drawings showing the vertical cross-sections of the basement floor slab of the UF6 Handling Area.

AREVA provided a basement floor plan for the UF6 Handling Area in Figure 3.3-21. Vertical cross-sections in the longitudinal and transverse directions are needed to assess the applicant's design of the basement floor slabs of the UF6 Handling Area.

10 CFR 70.64(a)(2) requires the applicant to include adequate protection against credible natural phenomena in its design of the facility. In addition, 10 CFR 70.62(c)(iv) requires the applicant to conduct and maintain an integrated safety analysis that identifies potential accident sequences caused by credible external events.

AES Response to NRC RAI:

[Security-related information withheld under 10 CFR 2.390]

While the detailed design of the seismic slab has not been developed at this time, its design drawings will be available for the NRC to conduct the operability readiness and management measures verification review to verify that management measures that ensure compliance with the performance requirements of 10 CFR 70.61 have been implemented and confirm that the facility has been constructed and will be operated safely and in accordance with the requirements of the license.

Similar to the Materials Licenses for the National Enrichment Facility and the USEC Facility, AES expects that the NRC will include a license condition in the Materials License that states:

> "Introduction of UF_6 into any module of the EREF shall not occur until the Commission completes an operational readiness and management measures verification review to verify that management measures that ensure compliance with the performance requirements of 10 CFR 70.61 have been implemented and confirms that the facility has been constructed and will be operated safely and in accordance with the requirements of the license. The licensee shall provide the Commission with 120 days advance notice of its plan to introduce UF_6 in any module of the EREF."

Associated EREF License Application Revisions:

The response to the RAI does not require any changes or additions to be made to the EREF License Application.

Commitments:

None

Attachments:

Enclosure 3.9 provides the following sketches:

Sketch 1 – "Partial North or South facing Longitudinal Cross-Section"

Sketch 2 – "Partial East or West Facing Transverse Cross-Section"

NRC RAI Number: SP-10

Text of NRC RAI:

<u>Integrated Safety Analysis Summary, Section 3.3.1.1.2, Functional Areas and Major Components</u>

Provide a description of the seismic isolators to be used in the UF_6 Handling Area, including design specifications.

This information is needed to assess the applicant's design of seismic isolators to analyze the building structures.

10 CFR 70.64(a)(2) and 70.64(a)(4) require the applicant to include adequate protection against credible natural phenomena in its design of the facility. In addition, 10 CFR 70.62(c)(iv) requires the applicant to conduct and maintain an integrated safety analysis that identifies potential accident sequences caused by credible external events.

AES Response to NRC RAI:

[Security-related information withheld under 10 CFR 2.390]

Similar to the Materials Licenses for the National Enrichment Facility and the USEC Facility, AES expects that the NRC will include a license condition in the Materials License that states:

> "Introduction of UF_6 into any module of the EREF shall not occur until the Commission completes an operational readiness and management measures verification review to verify that management measures that ensure compliance with the performance requirements of 10 CFR 70.61 have been implemented and confirms that the facility has been constructed and will be operated safely and in accordance with the requirements of the license. The licensee shall provide the Commission with 120 days advance notice of its plan to introduce UF_6 in any module of the EREF."

Associated EREF License Application Revisions:

The response to the RAI does not require any changes or additions to be made to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: SP-11

Text of NRC RAI:

Integrated Safety Analysis Summary, Section 3.3.2.3, Structural Design Loads

Provide a description of the methodology to be used for the structural analyses of items relied on for safety buildings for the design loads.

This information is needed to assess the applicant's approach to analyze the building structures.

10 CFR 70.64(a)(2) and 70.64(a)(4) require the applicant to include adequate protection against credible natural phenomena, environmental conditions, and dynamic effects in its design of the facility. In addition, 10 CFR 70.62(c)(iv) requires the applicant to conduct and maintain an integrated safety analysis that identifies potential accident sequences caused by credible external events.

AES Response to NRC RAI:

The IROFS structures will be analyzed using linear elastic analysis and numerical modeling methods. The IROFS buildings will be modeled using approved software that has been verified and controlled. For building framing (beam elements) and concrete walls and slabs (plate elements) the GT-Strudl program will be used. The design loads are applied to the pertinent structural elements, members, and joints on the model. The design loads are combined using the load combinations listed in ISA Summary Section 3.3.2.3.4. Resulting element and member moments and forces from the load combinations are reviewed to determine the controlling design loads for the buildings, foundations and other structural components. The governing loads are converted to stresses and compared to the allowable stresses specified in the codes identified in ISA Summary Section 3.3.2.1.

Associated EREF License Application Revisions:

The EREF License Application will be revised as follows to incorporate this RAI response:

ISA Summary Section 3.3.2.3 will be revised to [security-related information withheld under 10 CFR 2.390].

Commitments:

The EREF License Application will be revised to include the ISA Summary markups in Revision 2 of the EREF License Application.

Attachments:

Enclosure 3.2 shows the markup to EREF ISA Summary Section 3.3.2.3.

NRC RAI Number: HFE-1

Text of NRC RAI:

[SAR Section 3.3.1]

Describe the process that will be used to conduct a "human factors engineering review of the human-system interfaces" for those items relied on for safety (IROFS) requiring operator actions.

The license application states, on page 3.3-1:

For those IROFS requiring operator actions, a human factors engineering review of the human-system interfaces shall be conducted using applicable guidance in NUREG-0700, "Human-System Interface [Design] Review Guidelines," Revision 2, dated May 2002 [(NRC, 2000a)], and NUREG-0711, "Human Factors Engineering Program Review Model," Revision 2, dated February 2004 (NRC, 2004a).

A detailed description is needed of how the guidance will be applied specifically to the design and implementation of the human system interfaces for the AES-EREF (i.e., a level of detail as comparable to implementation plans as described in NUREG-0711). At a minimum, the application should provide an implementation plan level of detail that addresses the criteria contained in NUREG-0711, Element 8, "Human-System Interface Design", or an alternative supported by justification determined to be acceptable by the staff.

10 CFR 70.62(d) requires, in part, that ". ...**engineered and administrative controls and control systems that are identified as items relied on for safety pursuant to § 70.61(e) of this subpart 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 § 70.61 of this subpart" [emphasis added].

In addition, 10 CFR 70.64(a)(10) requires that, "The design must provide for inclusion of instrumentation and control systems to monitor and control the behavior of items relied on for safety." Given that the AES-EREF application contains many IROFS that rely on human action, the instrumentation and control systems associated with these IROFS must be designed to adequately support operator task performance.

Further, staff guidance contained in NUREG-1513, "Integrated Safety Analysis Guidance Document," identifies that for administrative controls (e.g., certain human actions), "...the man-machine interface for that individual should be carefully designed."

AES Response to NRC RAI:

SAR Section 3.3.1 states:

> "*For those IROFS requiring operator actions, a human factors engineering review of* the human-system interfaces shall be conducted using applicable guidance in

> NUREG-0700, "Human-System Interface Design Review Guidelines," Revision 2, dated May 2002 (NRC, 2000a), and NUREG-0711, "Human Factors Engineering Program Review Model," Revision 2, dated February 2004 (NRC, 2004a)."

During detailed design, the process that will be used to conduct a "human factors engineering review of the human-system interfaces" for those items relied on for safety (IROFS) requiring operator actions is described in the attached implementation plan. This plan addresses the criteria contained in NUREG-0711, Element 8, "Human-System Interface Design."

Associated EREF License Application Revisions:

None

Commitments:

During detailed design, the process that will be used to conduct a "human factors engineering review of the human-system interfaces" for those items relied on for safety (IROFS) requiring operator actions is described in the attached implementation plan. This plan addresses the criteria contained in NUREG-0711, Element 8, "Human-System Interface Design."

Attachments:

Enclosure 2.5: Human System Interface Design Implementation Plan

NRC RAI Number: HFE-2

Text of NRC RAI:

Integrated Safety Analysis Summary Section 3.3.1

Explain the purpose(s) and function(s) of the overview screen, control desk, and fire alarm system contained in the Control Room. Describe the method(s) used to design the overview screen, control desk, and fire alarm system. Describe the composition of the overview screen (e.g., the layout of the viewing area, information to be displayed, physical characteristics of the screen, etc.). Describe the composition of the control desk (e.g., presentation media used [visual display units (VDUs), soft-controls, computerized procedures, etc,). information to be displayed, etc.). Describe the composition of the fire alarm system (e.g., presentation medium used, information to be displayed, etc.). The explanations and descriptions should focus on how these control room components support the role of the operator in controlling and maintaining the facility in a safe condition and under upset/accident conditions.

The ISA Summary states, on page 3.3-10,

> The OSB [Operations Support Building] contains the following functional areas located on the second floor.
>
> Control Room
>
> The Control Room is the main monitoring point for the entire facility. The Control Room provides all of the facilities for the control of the plant, operational requirements, and personnel comfort. It is a permanently staffed area that contains the following equipment:
>
> - Overview screen
> - Control desk
> - Fire alarm system
> - Storage facilities
> - Communication systems

10 CFR 70.62(d) requires, in part, that ". ...**engineered and administrative controls and control systems that are identified as items relied on for safety pursuant for §70.61(e) of this subpart 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 § 70.61 of this subpart" [emphasis added].

In addition, 10 CFR 70.64(a) (10) requires that, "The design must provide for inclusion of instrumentation and control systems to monitor and control the behavior of items relied on for safety." Given that the AES-EREF application contains many IROFS that rely on human action, the instrumentation and control systems associated with these IROFS must be designed to adequately support operator task performance.

Further, staff guidance contained in NUREG-1513, "Integrated Safety Analysis," identifies that for administrative controls (e.g., certain human actions), "...the man-machine interface for that individual should be carefully designed."

AES Response to NRC RAI

[Security-related information withheld under 10 CFR 2.390]

SAR Section 7.5.1.7 provides additional information regarding the design and description of the main fire alarm control panel in the Control Room. The main fire alarm control panel monitors all building alarm panels and the fire pump controllers. All fire alarms, suppression system actuation alarms, supervisory alarms, and trouble alarms are audibly and visually annunciated by the main fire control alarm panel. This will permit the Control Room to activate the Fire Brigade. The failure of the main fire alarm control panel will not result in failure of any building's local fire alarm control panel and its associated local control functions. The main fire alarm control panel is not required to support any IROFS in the performance of their safety function. Additional design details will be developed during detailed design.

The specific composition of the overview screen, the control desk, and the fire alarm system will be determined during the detail design of the plant. These details will be available to enable the NRC to conduct the operability readiness and management measures verification review to verify that measures that ensure compliance with the performance requirements of 10 CFR 70.61 have been implemented and confirm that the facility has been constructed and will be operated safely and in accordance with the requirements of the license.

Similar to the Materials Licenses for the National Enrichment Facility and the USEC American Centrifuge Plant, AES expects that the NRC will include a license condition in the Materials License that states:

> "Introduction of UF_6 into any module of the EREF shall not occur until the Commission completes an operational readiness and management measures verification review to verify that the design and management measures that ensure compliance with the performance requirements of 10 CFR 70.61 have been implemented and confirms that the facility has been constructed and will be operated safely and in accordance with the requirements of the license. The licensee shall provide the Commission with 120 days advance notice of its plan to introduce UF_6 in any module of the EREF."

Associated EREF License Application Revisions:

None

Commitments:

None

Attachments:

None

NRC RAI Number: HFE-06

Text of NRC RAI:

Integrated Safety Analysis Summary, Section 3.5.7.1.3

Explain the purpose and functions of the Alarm Annunciation System and describe the method(s) used to design it.

The license application states, on page 3.5-36,

....Facility alarm systems which provide security, safety, and environmental protection such as fire alarm, radiation monitoring, gas release, equipment failure, etc. all provide audio and visual annunciation in either the Control Room or central alarm station. Control Room and or security personnel will respond to the alarm condition directly and if applicable annunciate the condition over the PA system.

10 CFR 70.62(d) requires, in part, that ". ...**engineered and administrative controls and control systems that are identified as items relied on for safety pursuant to § 70.61(e) of this subpart 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 § 70.61 of this subpart" [emphasis added].

In addition, 10 CFR 70.64(a) (10) requires that, "The design must provide for inclusion of instrumentation and control systems to monitor and control the behavior of items relied on for safety." Given that the AES-EREF application contains many IROFS that rely on human action, the instrumentation and control systems associated with these IROFS must be designed to adequately support operator task performance.

Further, staff guidance contained in NUREG-1513, "Integrated Safety Analysis," identifies that for administrative controls (e.g., certain human actions), "... the man-machine interface for that individual should be carefully designed."

AES Response to NRC RAI:

[Security-related information withheld under 10 CFR 2.390]

Associated EREF License Application Revisions:

This response to the RAI does not require any revisions to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: HFE-9

Text of NRC RAI

Integrated Safety Analysis Summary, Section 3.5.9.2.2

Explain how operator actions taken from the Local Control Centers (LCCs) are coordinated with the Central Control System (CCS).

The ISA Summary states, on page 3.5-40,

> "Each LCC has sufficient functionality to completely operate and protect its associated process system without any CCS intervention."

The application indicates that each local control station can "completely operate and protect its associated process system with out CCS intervention." What measures are in place to ensure coordination between the local control operators and control center operators (e.g., isolation devices/lock-outs present, administrative procedures, etc.).

10 CFR 70.62(d) requires, in part, that ". ...**engineered and administrative controls and control systems that are identified as items relied on for safety pursuant to § 70.61(e) of this subpart 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 § 70.61 of this subpart" [emphasis added].

In addition, 10 CFR 70.64(a) (10) requires that, "The design must provide for inclusion of instrumentation and control systems to monitor and control the behavior of items relied on for safety." Given that the AES-EREF application contains many IROFS that rely on human action, the instrumentation and control systems associated with these IROFS must be designed to adequately support operator task performance.

Further, staff guidance contained in NUREG-1513, "Integrated Safety Analysis," identifies that for administrative controls (e.g., certain human actions), "...the man-machine interface for that individual should be carefully designed."

AES Response to NRC RAI:

Normal operation of the plant does not rely upon operator intervention. Under normal circumstances, each Local Control Center will continue to monitor and control the systems/components under its control without any intervention from either Central Control Room systems or operators. There is a local operator interface (LOI) at each Local Control Center. This device allows an operator access to the Supervisory Control and Data Acquisition System (SCADA) relating to that Local Control Center mimicking exactly the view from the Central Control Room. From here, the LOI can be used to change equipment modes and sequences just as in the Central Control Room provided that the Central Control Room has electronically given permission to do so by authorizing 'local' control. When equipment has been given to local control both mechanical and electrical lockouts and interlocks are in place to ensure inadvertent commands from the Central Control Room are eliminated. Supervisory control is only given back when the local operator has completed his task and removed all protection devices.

Operator actions performed at the Local Control Centers. that are credited as an IROFS, consist of scheduled monitoring functions with the potential of operator intervention if the process parameters are not within system specifications.

Associated EREF License Application Revisions:

None

Commitments:

None

Attachments:

None

NRC RAI Number: RP-1

Text of NRC RAI:

Clarify the frequency of training program evaluation and review/updates. There appears to be inconsistency in the frequency of evaluation/review/update of the Radiation Protection training program. The last two sentences of the next to last paragraph of Section 4.5 state an annual frequency while Paragraph 7 of Section 4.5.1 and Paragraph D of Section 11.3.3.1.1 state a frequency of 2 years.

AES Response to NRC RAI:

SAR (Revision 1), Section 4.5.1, paragraph 7, will be revised to change the frequency for which the contents of the formal radiation protection training program are reviewed and updated from "at least every two years" to "at least annually".

SAR (Revision 1), Section 11.3.3.1.1, paragraph D, states that the frequency for which the contents of the radiation protection sections of the nuclear safety training program are reviewed and updated is "at least annually" and requires no change.

Associated EREF License Application Revisions:

The EREF license application will be revised as shown on the SAR Section 4.5.1 markup presented in Enclosure 2.2.

Commitments:

The EREF License Application will be revised to include the SAR Section 4.5.1 markup in Revision 2 of the EREF License Application.

Attachments:

Enclosure 2.2 provides the markup page for SAR Section 4.5.1.

NRC RAI Number: RP-2

Text of NRC RAI:

Clarify in Section 4.6.1, of the Safety Analysis Report (SAR), whether exhaust ventilation serving potentially contaminated areas of dispersible materials are treated for radiological and hydrogen fluoride (HF) contamination in the air stream via filters, similar to the Gaseous Effluent Ventilation Systems (GEVSs), and whether these filters (pre-, high efficiency particulate air (HEPA), and carbon adsorption) are continuously monitored for differential pressure and that the air stream is monitored for alpha and HF concentrations. This is needed so that staff can determine compliance with 10 CFR 20.1701, 70.23(a)(3), and NUREG-1520 Section 4.4.6.3(4) which require adequate process and engineering controls affecting the concentration of radioactive materials in air.

AES Response to NRC RAI:

The potentially contaminated areas for the EREF served by the HVAC systems in question are the Ventilated Room in the Blending, Sampling, and Preparation Building (BSPB); the Decontamination Workshop, Chemical Trap Workshop, Mobile Unit Disassembly & Reassembly Workshop, Valve and Pump Dismantling Workshop, and Maintenance Facility in the Technical Support Building (TSB); and the Centrifuge Test and Post Mortem Facility in the Centrifuge Assembly Building (CAB).

The HVAC systems serving these potentially contaminated areas for the EREF include filtration for removal of radiological and HF contamination from the air stream, and include alpha and HF monitoring of the exhaust air stream. This information is contained in the following ISA Summary Sections:

- The HVAC system for the Ventilated Room is described in ISA Summary Section 3.5.1.1.7 and is illustrated on ISA Summary Figure 3.5.6.B.
- The HVAC system for the Decontamination Workshop, Chemical Trap Workshop, Mobile Unit Disassembly & Reassembly Workshop, Valve and Pump Dismantling Workshop, and Maintenance Facility is described in ISA Summary Section 3.5.1.1.8, and is illustrated on ISA Summary Figure 3.5-7.
- The HVAC system for the Centrifuge Test and Post Mortem Facility is presented in ISA Summary Section 3.5.1.1.9, and is illustrated on ISA Summary Figure 3.5-13.

As discussed in SAR Section 4.6.1 (5^{th} and 6^{th} paragraphs), several measures are taken to ensure effective operation of the ventilation filtration systems. Differential pressure across HEPA filters in potentially contaminated ventilation exhaust systems is monitored monthly or automatically monitored and alarmed. Operating procedures specify limits and setpoints on the differential pressure consistent with manufacturers' recommendations. Additionally, filter inspection, testing, maintenance and change out criteria are specified in written procedures approved by the Operations Manager, or a designated alternate. Change out frequency is based on considerations of filter loading, operating experience, differential pressure data and any UF_6 releases indicated by HF alarms.

Associated EREF License Application Revisions:

The response to the RAI does not require any changes or additions to be made to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: RP-3

Text of NRC RAI:

Explain the effluent sampler mechanism for the Liquid Effluent Collection and Treatment System and summarize this equipment's operating history. This is needed for staff to determine compliance with 10 CFR 20.1501(a)(2), 70.23(a)(3), and NUREG-1520, Section 4.4.7.3(11) which require adequate equipment and facilities to quantify radiological hazards.

AES Response to NRC RAI:

The liquid effluent treatment system is a batch process in which effluents are sent through precipitation tanks and filters to remove uranium and fluoride. The filtrate ultimately is sent to a pot evaporator. The evaporator produces a chemically decontaminated gaseous effluent. This gaseous effluent will be sampled using an air pump. The pump will be connected to a sampling device such as a bubbler, condenser or desiccant. The selected sampling mechanism will be reliable based on operating experience with similar systems.

The design of the effluent sampler mechanism will be developed in detail design. This detail will be available for the NRC to conduct the operability readiness and management measures verification review to verify that management measures that ensure compliance with the performance requirements of 10 CFR 70.61 have been implemented and confirm that the facility has been constructed and will be operated safely and in accordance with the requirements of the license.

Similar to the Materials Licenses for the National Enrichment Facility and the USEC Facility, AES expects that the NRC will include a license condition in the Materials License that states:

> "Introduction of UF_6 into any module of the EREF shall not occur until the Commission completes an operational readiness and management measures verification review to verify that management measures that ensure compliance with the performance requirements of 10 CFR 70.61 have been implemented and confirms that the facility has been constructed and will be operated safely and in accordance with the requirements of the license. The licensee shall provide the Commission with 120 days advance notice of its plan to introduce UF_6 in any module of the EREF."

Associated EREF License Application Revisions:

None

Commitments:

None

Attachments:

None

NRC RAI Number: RP-4

Text of NRC RAI:

Section 4.4, final paragraph, of the SAR does not discuss distribution of Radiation Protection procedures in accordance with NUREG-1520, Section 4.4.4.3(2). This is needed for staff to determine compliance with 10 CFR 70.22(a)(8) which requires adequate procedures.

AES Response to NRC RAI:

As specified in SAR Section 4.4, SAR Chapter 11 describes the program implemented for the control of procedures.

SAR Section 11.4.5, Distribution of Procedures, describes the requirements for distribution of procedures.

Associated EREF License Application Revisions:

The response to the RAI does not require any changes or additions to be made to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: RP-5

Text of NRC RAI:

SAR Section 4.8.1.2

Clarify in the SAR, Section 4.8.1.2, what equipment will be used for airborne activity monitoring and, if there are alarm functions, what criteria will be used to establish alarm settings. In Section 4.8.1.2, it is unclear whether the applicant is referring to continuous air monitors (which typically provide real time analysis of airborne radioactivity with alarm capability), continuous air samplers (where filters are analyzed after being exchanged), or both. This may be indicative of some slight inconsistency with text in SAR Section 4.7, paragraphs 7 and 8. This is needed for staff to determine compliance with 10 CFR 20.1502(b), 70.23(a)(3), and NUREG-1520 Section 4.4.7.3(4) which require adequate equipment to monitor the occupation intake of radioactive material.

AES Response to NRC RAI:

SAR Section 4.8.1.2 will be revised to clarify the type of equipment used for airborne radioactivity monitoring, and where there are alarm functions, the criteria that will be used to establish alarm settings. These changes are consistent with the text in SAR Section 4.7.

The monitoring equipment for airborne radioactivity monitoring in SAR Section 4.8.1.2 are continuous air monitors and continuous air samplers. Continuous air monitors are permanently located in Restricted Areas of the facility. Continuous air monitors in locations classified as Airborne Radioactivity Areas are equipped with alarms. When deemed necessary, portable air samplers may be used to collect a sample on filter paper for subsequent analysis in the laboratory.

Monitor data is collected for regular analysis and documentation. Monitors in locations classified as Airborne Radioactivity Areas are continuous air monitors equipped with alarms. The alarm is activated when airborne radioactivity levels exceed predetermined limits. The limits are set with consideration being given to both toxicity and radioactivity. The volume of air sampled may have to be adjusted to ensure adequate sensitivity with minimum sampling time. The operating history of the facility, changes in technology, changes in room functions and design, and changes in regulations may necessitate adjustment of the monitors.

Associated EREF License Application Revisions:

The EREF License Application will be revised as follows to incorporate this RAI response:

SAR Section 4.8.1.2 will be revised as described in the response to this RAI.

Commitments:

The EREF License Application will be revised to include the SPPP markups in Revision 2 of the EREF License Application.

Attachments:

Enclosure 2.2 shows the markup to EREF SAR Section 4.8.1.2.

NRC RAI Number: RP-6

Text of NRC RAI:

Section 4.11.2 of the SAR does not include reference to reporting requirements in 10 CFR 30.50 and 10 CFR 40.60, as applicable. Please revise the SAR or provide additional justification as to why this information is not needed. This is needed to confirm compliance with these reporting requirements which may differ slightly and be subject to change from those in 10 CFR 20 and 10 CFR 70.

AES Response to NRC RAI:

SAR Section 4.11.2 will be revised to reference reporting requirements in 10 CFR 30.50 and 10 CFR 40.60. SAR Section 4.12, References, will be revised to add 10 CFR 30.50 and 10 CFR 40.60 as references.

Associated EREF License Application Revisions:

The EREF License Application will be revised as follows to incorporate this RAI response:

SAR Section 4.11.2 will be revised to reference reporting requirements in 10 CFR 30.50 and 10 CFR 40.60. SAR Section 4.12 will be revised to add 10 CFR 30.50 and 10 CFR 40.60 as references.

Commitments:

The EREF License Application will be revised to include the SPPP markups in Revision 2 of the EREF License Application.

Attachments:

Enclosure 2.2 shows the markup to EREF SAR Sections 4.11.2 and 4.12.

NRC RAI Number: RP-7

Text of NRC RAI:

Section 4.1.1 of the SAR does not discuss any professional staff, supervision, and/or technicians considered to be key personnel for the Radiation Protection program with defined qualifications and responsibilities. Please revise the SAR or provide additional justification as to how 10 CFR 70.22(a) is met. This is needed for staff to determine compliance with 10 CFR 70.22(a) and NUREG-1520 Section 4.4.1.3(2) and 4.4.1.3(3) which require the licensee to specify the qualifications and responsibilities of key program personnel.

AES Response to NRC RAI:

SAR Section 4.3 provides the requirements for the technical qualifications, including training and experience, of the facility Radiation Protection Program staff. As stated in SAR Section 4.3, the Radiation Protection Program staff is assigned responsibility for implementation of the Radiation Protection Program functions, and staffing is consistent with the guidance provided in Regulatory Guides 8.2 (NRC, 1973a) and 8.10 (NRC, 1977).

As stated in SAR Sections 4.1.2 and 4.3, members of the Radiation Protection Program staff are trained and qualified consistent with the guidance provided in American National Standards Institute (ANSI) standard 3.1, Selection, Qualification and Training of Personnel for Nuclear Power Plants (ANSI, 1993).

As stated in SAR Section 4.1.2 and 2.2.4, the Radiation Protection/Chemistry Manager has, as a minimium, a bachelor's degree (or equivalent) in an engineering or scientific field and at least four years of responsible nuclear experience. The responsibilities of the Radiation Protection Manager and his staff are provided in SAR Section 4.1.1.4.

As stated in SAR Section 4.1.2, at least one member of the Radiation Protection/Chemistry Manager's staff shall have at least two years of experience at a facility that processes uranium, including uranium in soluble form.

Associated EREF License Application Revisions:

The response to the RAI does not require any changes or additions to be made to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: RP-8

Text of NRC RAI:

Section 4.2.1

Section 4.2.1 of the SAR (or other section if appropriate) does not appear to consider guidance in Regulatory Guide 4.21 when performing facility construction or modifications. Please provide additional information or an alternate approach to Regulatory Guide 4.21. This is needed for staff to determine compliance with 10 CFR 20.1406 and 70.23(a)(3).

AES Response to NRC RAI:

SAR Section 4.2 will be revised to state:

> "The guidance of Regulatory Guide 4.21 will be followed to minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste (NRC, 2008)."

SAR Section 4.12 will be revised to include a reference to Regulatory Guide 4.21.

Associated EREF License Application Revisions:

The EREF License Application will be revised as depicted in Enclosure 2.2.

Commitments:

The EREF License Application will be revised to include the SAR markups in Revision 2 of the EREF License Application.

Attachments:

Enclosure 2.2 shows the markup to EREF SAR Sections 4.2 and 4.12.

References:

Regulatory Guide 4.21, Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning

NRC RAI Number: RP-9

Text of NRC RAI:

Section 4.6.1

Revise Section 4.6.1 of the SAR to include discussion that ventilation and containment systems will be designed and sized appropriately to reduce airborne concentrations below the applicable DAC values as described in NUREG-1520 Section 4.4.6.3(1). This is needed for staff to determine compliance with 10 CFR 20.1101(b) and 70.23(a)(3) which require adequate facilities to keep public exposures ALARA and protect health and the environment.

AES Response to NRC RAI:

SAR Section 4.6.1 will be revised to include that ventilation and containment systems in the areas of the facility identified as having potential airborne concentrations of radionuclides that could exceed the occupational, derived air concentration (DAC) values specified in 10 CFR 20, Appendix B, during normal operations, will be designed and sized appropriately to reduce airborne concentrations below the applicable DAC values.

Associated EREF License Application Revisions:

The EREF License Application will be revised as follows to incorporate this RAI response:

SAR Section 4.6.1 will be revised to include that ventilation and containment systems (serving contaminated and potentially contaminated areas of the facility) will be designed and sized appropriately to reduce airborne concentrations below the occupational, derived air concentration (DAC) values specified in 10 CFR 20, Appendix B, during normal operations.

Commitments:

The EREF License Application will be revised to include the SAR markups in Revision 2 of the EREF License Application.

Attachments:

Enclosure 2.2 shows the markup to EREF SAR Sections 4.6.1.

NRC RAI Number: NCS-1

Text of NRC RAI:

SAR Section 5.2.1.5

Identify which acceptance criteria in NUREG-1520, Section 5.4.3.4 will be used to analyze nuclear criticality safety (NCS) accident sequences in operations and processes.

SAR Section 5.2.1.5 states:

> NCS analyses also meet the following: The NCS methodologies and technical practices in NUREG-1520 (NRC, 2002), Section 5.4.3.4, are used to analyze NCS accident sequences in operations and processes.

Not all the criteria in NUREG-1520, Section 5.4.3.4 appear to be applicable to NCS analyses. This information is needed to clarify a commitment made in the SAR.

AES Response to NRC RAI:

The acceptance criteria that will not be used in NUREG-1520, Section 5.4.3.4.2, Technical Practices, are provided below and listed at the end of SAR Section 5.1.2:

- Item 9: There is no planned use of density as a control parameter,
- Item 13: There is no planned use of concentration as a control parameter, and
- Item 16: There is no planned use of neutron absorption (e.g., borosilicate-glass raschig rings, cadmium) as a control parameter.

The applicable Technical Practices will be determined on a system-by-system basis.

Also note that the methodologies applicable to NCS analysis described in NUREG-1520, Section 5.4.3.4.1 are applicable except for Item (4) which states:

- Item 4 - An inadvertent nuclear criticality will be detected promptly to ensure that radiation exposures to workers are minimized.

The AES Eagle Rock Enrichment Facility NCS controls are preventative in nature; therefore, detector response is not applicable to analyze NCS accident sequences. However, as stated in the SAR, Section 5.3, a Criticality Accident Alarm System will be installed at the facility to meet the requirements of 10 CFR 70.24. Additionally, as stated in SAR, Section 5.1.1, all required personnel will be trained for proper response to the activation of the CAAS.

Associated EREF License Application Revisions:

The following section of the EREF License Application will be revised to incorporate the above response:

SAR Section 5.2.1.5, Additional Nuclear Criticality Safety Analyses Commitments (page 5.2-6).

Commitments:

Revise the EREF License Application to include these markups in Revision 2.

Attachments:

Enclosure 2.2 provides the markup page for SAR Section 5.2.1.5, Additional Nuclear Criticality Safety Analyses Commitments.

NRC RAI Number: NCS-2

Text of NRC RAI:

SAR Section 5.1.2

Justify the use of partial reflection (2.5 cm of water) to account for humans and other spurious reflectors. Clarify what is meant by "2.5 cm of water reflection around vessels."

The SAR states:

> (Section 5.1.2) Partial reflection of 2.5 cm (0.984 in) of water is assumed where limited moderating materials (including humans) may be present.
>
> (Section 5.2.1.3.1) . . . where appropriate, spurious reflection due to walls, fixtures, personnel, etc. has been accounted for by assuming 2.5 cm (0.984 in) of water reflection around vessels.

It is not clear how partial reflection will be used in NCS analyses and that it is sufficient to account for humans or other reflectors in close proximity to SNM. 10 CFR 70.61(d) requires all nuclear processes to be subcritical under normal and credible abnormal conditions with an approved margin of subcriticality for safety.

AES Response to NRC RAI:

For clarification, the 2.5 cm of water reflection around vessels is intended to model the presence of a water layer in contact with the outside surface of a vessel (except on those sides which are in direct contact with another component/surface). Examples using a water reflector around a component are shown in the attached 2-D figures (top and side views, respectively). The figures display a cylinder of fissile material with water reflection sitting on a concrete slab.

The use of 2.5 cm of water (H_2O) to simulate spurious reflection is a common practice. Typically, this water thickness is used to simulate hands being placed on a component bearing fissile material and/or to simulate a water sheath on the surface of a container. Additionally, 2.5 cm of water around the model boundary can be used as a best estimate to simulate potential reflection from a non-concrete wall (room return) or human presence.

Examples using a 2.5 cm of H_2O reflector can be found in ARH-600.

- Page III.B.4-5, "Critical Infinite Cylinder - UO_2-H_2O with Nominal (One inch) H_2O Reflection",
- Pages II.E.1 and II.E.2, "Reflector Savings - Water Reflector" uses 1-inch of H_2O as reflector around spheres of ^{233}U and ^{239}Pu (modeled separately).

An additional reference supporting this position is:

- Haire (ORNL), Jordan (ORNL), and Taylor (ORNL Y-12), "Storage Array Reflection Considerations", ANS 1997 Winter Meeting, Albuquerque, NM.

The use of 2.5 cm of water described above was also accepted by the NRC for licensing the NEF facility.

Associated EREF License Application Revisions:

The response to the RAI does not require any changes or additions to be made to the EREF License Application.

Commitments:

None

Attachments:

None

References:

The following additional public references were utilized:

1. R. D. Carter, G. R. Kiel, and K. R. Ridgway, Criticality Handbook, ARH-600, Vol. I to III, Atlantic Richfield Company, Richland, Washington, 1969/71.

2. Haire (ORNL), Jordan (ORNL), and Taylor (ORNL Y-12), "Storage Array Reflection Considerations", ANS 1997 Winter Meeting, Albuquerque, New Mexico.

NRC RAI Number: NCS-3

Text of NRC RAI:

SAR Table 5.1-2

Clarify the safety criteria for tanks listed in Table 5.1-2 to ensure consistency with other statements in the SAR.

For example, SAR Section 5.1.2 states:

> . . . the values in Table 5.1-2, Safety Criteria for Buildings/Systems/ Components, represent the limits based on 6.0 wt% enrichment except for the Contingency Dump System traps which are limited to 1.5 wt% ^{235}U.

The safety criteria for tanks listed in Table 5.1-2 is based on 5 wt% enrichment not 6 wt% enrichment. If the safety criteria for tanks will be based upon 5 wt% enrichment, justify that this is sufficiently conservative to ensure that all processes will remain subcritical.

10 CFR 70.61(d) requires all nuclear processes to be subcritical under normal and credible abnormal conditions with an approved margin of subcriticality for safety.

AES Response to NRC RAI:

AES will update SAR Table 5.1-2 to reflect the safety criteria for tanks with 6 wt% enriched material consistent with SAR Section 5.1.2.

Associated EREF License Application Revisions:

The following section of the EREF License Application will be revised to incorporate the above response:

SAR Table 5.1-2, Safety Criteria for Buildings/ Systems/ Components

Commitments:

Revise the EREF License Application to include these markups in Revision 2.

Attachments:

Enclosure 2.2 provides the markup page for SAR Table 5.1-2.

NRC RAI Number: NCS-4

Text of NRC RAI:

SAR Section 5.2.1.3

Clarify whether or not the assumptions (other than enrichment) described in SAR Section 5.2.1.3 are used for determining the safe values of geometry or volume.

The SAR states:

> (Section 5.1.2) The safe values of geometry / volume define the characteristic dimension of importance for a single unit such that nuclear criticality safety is not dependent on any other parameter assuming 6 wt% ^{235}U for safety margin.
>
> (Section 5.1.2) The values on Table 5.1-1 are chosen to be critically safe when optimum light water moderation exists and reflection is considered within isolated systems.
>
> (Section 5.2.1.3) The NCS analyses results provide values of k-effective (k_{eff}) to conservatively meet the upper safety limit. The following sections provide a description of the major assumptions used in the NCS analyses.

The assumptions described in Section 5.2.1.3 do not appear to be entirely consistent with the statements in Section 5.1.2. For example, performing analysis with an H/U ratio up to 7 may not account for optimum moderation in all cases.

10 CFR 70.61(d) requires all nuclear processes to be subcritical under normal and credible abnormal conditions with an approved margin of subcriticality for safety.

AES Response to NRC RAI:

AES will revise SAR Section 5.1.2 to clarify that the values in Table 5.1-1 are calculated with optimum moderation (i.e., various H/U ratios greater than and less than 7 are analyzed) and 30 cm water reflection.

H/U is generally limited to 7, which may not be an optimally moderated value, but is regarded as being beyond what is possible in most parts of a vacuum plant, and therefore, is regarded as the worst case H/U (see SAR Section 5.2.1.3.3). An H/U ratio of 7 is not claimed to be an optimum value.

Associated EREF License Application Revisions:

The following section of the EREF License Application will be revised to incorporate the above response:

SAR Section 5.1.2, Control Methods for Prevention of Criticality (page 5.1-3)

Commitments:

Revise the EREF License Application to include these markups in Revision 2.

Attachments:

Enclosure 2.2 provides the markup page for SAR Section 5.1.2.

NRC RAI Number: NCS-5

Text of NRC RAI:

SAR Section 5.3

Revise the SAR to clarify the commitment to provide criticality accident alarm system (CAAS) coverage.

SAR Section 5.3 states:

> Areas where Special Nuclear Material (SNM) is handled, used, or stored in amounts at or above the 10 CFR 70.24 (CFR, 2008d) mass limits are provided with CAAS coverage.

This statement is not consistent with the regulatory requirements of 10 CFR 70.24. 10 CFR 70.24 requires that licensees authorized to possess greater than a critical mass of SNM shall provide CAAS coverage in *each* area where SNM is handled, used, or stored. The license application requests authorization to possess greater than a critical mass of SNM, therefore an exemption to 10 CFR 70.24 must be requested to exclude areas from CAAS coverage where SNM is handled, used, or stored. Such a request should specify the areas where CAAS coverage may not be provided and justify that the 10 CFR 70.17 requirements for granting an exemption are met.

AES Response to NRC RAI:

10 CFR 70.24(a) states:

> "Each licensee authorized to possess special nuclear material in a quantity exceeding 700 grams of contained uranium-235...shall maintain in each area in which ***such*** [emphasis added] licensed specific nuclear material is handled, used, or stored, a monitoring system meeting the requirements of..."

Therefore, CAAS coverage is only required in those areas that contain special nuclear material at or above the mass limits provided in 10 CFR 70.24(a). This position is consistent with the NRC staff position stated in NUREG-1851, Section 5.3.4.4, Criticality Accident Alarm Coverage:

> "Non-FMOs and areas with less than 700 g ^{235}U do not require CAAS under 10 CFR 70.24(a)"

It is also consistent with NUREG-1520, Section 5.4.3.4.3, which states:

> "At or above the 10 CFR 70.24 mass limits, CAAS coverage should be required in each area in which SNM is handled, stored, or used."

While the current wording of SAR Section 5.3 is consistent with 10 CFR 70.24(a), AES will revise SAR Section 5.3 to use wording consistent with the code.

Associated EREF License Application Revisions:

The following section of the EREF License Application will be revised to incorporate the above response:

SAR Section 5.3, CRITICALITY ACCIDENT ALARM SYSTEM (CAAS) (page 5.3-1)

Commitments:

Revise the EREF License Application to include these markups in Revision 2.

Attachments:

Enclosure 2.2 provides the markup page for SAR Section 5.3.

References:

The following additional public references were utilized in this response:

NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility, March 2002.

NUREG-1851, Safety Evaluation Report for the American Centrifuge Plant in Piketon, Ohio, September 2006.

NRC RAI Number: NCS-6

Text of NRC RAI:

Revise the Emergency Plan to clearly identify the discussion of the CAAS. The Emergency Plan (Section 2.1.1.2) states:

> Refer to Section 6.4.1 for a description of the Criticality Accident Alarm System (CAAS).

Section 6.4.1 of the Emergency Plan does not describe the CAAS. This appears to be a typographic error.

AES Response to NRC RAI:

AES will correct the wording in Emergency Plan Section 2.1.1.2 to refer to Emergency Plan Section 5.3.3 for a description of the Criticality Accident Alarm System (CAAS).

Associated EREF License Application Revisions:

The following section of the EREF License Application will be revised to incorporate the above response:

Emergency Plan Section 2.1.1.2, Nuclear Criticality (page 2.1-2).

Commitments:

Revise the EREF License Application to include these markups in Revision 2.

Attachments:

Enclosure 3.3 provides the markup page for Emergency Plan Section 2.1.1.2.

NRC RAI Number: NCS-7

Text of NRC RAI:

SAR Section 5.3

Describe AES's commitment to the following statements from Section 3.1.5 of the ISA Summary, and incorporate into the SAR:

a. The CAAS will be uniform throughout the facility for the type of radiation detected, the mode of detection, the alarm signal, and the system dependability.

b. The CAAS is . . . designed to remain operational during credible events or conditions, including fire, explosion, corrosive atmosphere, or seismic shock (equivalent to the site-specific design-basis earthquake or the equivalent value specified by the uniform building code).

c. Whenever the CAAS is not functional, compensatory measures, such as limiting access and restricting SNM movement, will be implemented. Should the CAAS coverage be lost and not restored within a specified number of hours, the operations will be rendered safe (by shutdown and quarantine) if necessary. Onsite guidance is provided and is based on process-specific considerations that consider applicable risk trade-off of the duration of reliance on compensatory measures versus the risk associated with process upset in shutdown.

10 CFR 70.24 requires a CAAS be maintained in each area where SNM is handled, used, or stored for facilities authorized to possess greater than a critical mass of SNM.

NUREG-1520, Section 5.4.3.4.3 indicates that commitments similar to those statements listed above are needed to ensure a CAAS is in place that will adequately meet the requirements of 10 CFR 70.24.

AES Response to NRC RAI:

10 CFR 70.24(a) states:

"Each licensee authorized to possess special nuclear material in a quantity exceeding 700 grams of contained uranium-235...shall maintain in each area in which ***such*** [emphasis added] licensed specific nuclear material is handled, used, or stored, a monitoring system meeting the requirements of..."

AES will revise SAR Section 5.3 to state the following consistent with the commitment in ISA Summary Section 3.1.5:

"The facility is provided with a Criticality Accident Alarm System (CAAS) as required by 10 CFR 70.24, (CFR, 2008d). Each area where Special Nuclear Material (SNM) is handled, used, or stored in amounts at or above the 10 CFR 70.24 (CFR, 2008d) mass limits are provided with CAAS coverage. The CAAS will be uniform throughout the facility for the type of radiation detected and alarm signals. Documentation shall be maintained which demonstrate the CAAS meets the requirements of 10 CFR 70.24. Emergency management measures are covered in the facility Emergency Plan.

The CAAS is provided with emergency power and is designed to remain operational during credible events or conditions, including fire, explosion, corrosive atmosphere, or seismic shock (equivalent to the site-specific design-basis earthquake or the equivalent value specified by the uniform building code).

Whenever the CAAS is not functional, compensatory measures, such as limiting access and restricting SNM movement, will be implemented. Should the CAAS coverage be lost and not restored within a specified number of hours or an equivalent level of protection has not been provided (e.g., portable CAAS system), the operations will be rendered safe (by shutdown and quarantine) if necessary. Onsite guidance is provided and is based on process-specific considerations that consider applicable risk trade-off of the duration of reliance on compensatory measures versus the risk associated with process upset in shutdown."

Associated EREF License Application Revisions:

The following section of the EREF License Application will be revised to incorporate the above response:

SAR Section 5.3, CRITICALITY ACCIDENT ALARM SYSTEM (CAAS) (page 5.3-1)

Commitments:

Revise the EREF License Application to include these markups in Revision 2.

Attachments:

Enclosure 2.2 provides the markup page for SAR Section 5.3.

NRC RAI Number: NCS-8

Text of NRC RAI:

State in the SAR that documentation will be maintained that demonstrates the CAAS meets the requirements of 10 CFR 70.24.

10 CFR 70.24 requires a CAAS be maintained in each area where SNM is handled, used, or stored for facilities authorized to possess greater than a critical mass of SNM.

This information is needed to ensure a CAAS is in place that will adequately meet the requirements of 10 CFR 70.24.

AES Response to NRC RAI:

AES will revise SAR Section 5.3 to add a sentence that states that documentation will be maintained that demonstrates the CAAS meets the requirements of 10 CFR 70.24.

Associated EREF License Application Revisions:

The following section of the EREF License Application will be revised to incorporate the above response:

SAR Section 5.3, CRITICALITY ACCIDENT ALARM SYSTEM (CAAS) (page 5.3-1)

Commitments:

Revise the EREF License Application to include this markup in Revision 2.

Attachments:

Enclosure 2.2 provides the markup page for SAR Section 5.3.

NRC RAI Number: NCS-9

Text of NRC RAI:

Commit to only use NCS controls which are capable of preventing a criticality accident, or provide and justify an alternative commitment. The SAR does not clearly state that to meet the performance requirements AREVA will only use NCS controls which can prevent a criticality accident.

10 CFR 70.61(d) requires the use of preventive controls and measures as the primary means of protecting against a nuclear criticality accident.

AES Response to NRC RAI:

All Nuclear Criticality Safety controls are preventative in nature; there are no mitigative controls. The following statement will be inserted into SAR, Section 5.1.1:

All NCS controls are preventative in nature; there are no mitigative NCS controls.

Associated EREF License Application Revisions:

The following section of the EREF License Application will be revised to incorporate the above response:

SAR Section 5.1.1, Management of the Nuclear Criticality Safety (NCS) Program (page 5.1-1).

Commitments:

Revise the EREF License Application to include this markup in Revision 2.

Attachments:

Enclosure 2.2 provides the markup page for SAR Section 5.1.1.

NRC RAI Number: NCS-10

Text of NRC RAI:

The definition of a non-interacting unit does not appear to be practical. Explain how it is determined if a unit is non-interacting when an NCS analysis (NCSA) is not performed.

SAR Section 5.1.2 states:

> A non-interacting unit is defined as a unit that is spaced an approved distance from other units such that the multiplication of the subject unit is not increased.
>
> If a unit is considered interacting, NCSAs are performed.

This definition is not practical since all units can impact the multiplication factor no matter how far apart the units are. It is unclear how a unit could be considered non-interacting if no NCSA has been performed.

10 CFR 70.61(d) requires all nuclear processes to be subcritical under normal and credible abnormal conditions with an approved margin of subcriticality for safety.

AES Response to NRC RAI:

The generic hand calculation was used as a screen to define acceptable NCS limits and configurations for typical configurations. Although hand calculations are an acceptable method for defining NCS limits and restrictions, the results may not be conservative for specific calculations and/or configurations. Therefore, spacing requirements will be determined on a system by system basis.

While the detailed design has not been developed at this time for these systems, this detail will be available for the NRC to conduct the operability readiness and management measures verification review to verify that management measures that ensure compliance with the performance requirements of 10 CFR 70.61 have been implemented and confirm that the facility has been constructed and will be operated safely and in accordance with the requirements of the license.

Similar to the Materials Licenses for the National Enrichment Facility and the USEC Facility, AES expects that the NRC will include a license condition in the Materials License that states:

> "Introduction of UF_6 into any module of the EREF shall not occur until the Commission completes an operational readiness and management measures verification review to verify that management measures that ensure compliance with the performance requirements of 10 CFR 70.61 have been implemented and confirms that the facility has been constructed and will be operated safely and in accordance with the requirements of the license. The licensee shall provide the Commission with 120 days advance notice of its plan to introduce UF_6 in any module of the EREF."

AES will revise SAR Sections 5.1.2 - Interaction to state:

> NCSAs and NCSEs shall consider the potential effects of neutron interaction including interaction effects of in-transit materials. Spacing requirements will be determined on a system by system basis.
>
> Individual unit multiplication, array interaction, and in-transit material interactions are evaluated using the Monte Carlo computer code MONK8A to ensure k_{eff} $(k_{calc} + 3\sigma_{calc}) < 0.95$.

AES will delete the hand calculation discussion from SAR Section 5.2.1.3.4, Vessel Movement Assumption. The spacing requirements for vessel movement will be determined on a system by system basis, including component insertion or extraction from an array.

Associated EREF License Application Revisions:

The following sections of the EREF License Application will be revised to incorporate the above response:

SAR Section 5.1.2 (page 5.1-6)

SAR Section 5.2.1.3.4 (page 5.2-4)

Commitments:

Revise the EREF License Application to include these markups in Revision 2.

Attachments:

Enclosure 2.2 provides the markup pages for SAR Section 5.1.2 and SAR Section 5.2.1.3.4.

NRC RAI Number: FS-1

Text of NRC RAI:

Safety Analysis Report, Section 7.1.4, p. 7.1-2

Provide detailed information on the frequency, scope, and data collected during inspections of water based fire protection systems. A referenced commitment to National Fire Protection Association (NFPA) 25, "Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems," would be sufficient.

The regulation 10 CFR 70.22(a)(8) requires the applicant to provide proposed procedures to protect health and minimize danger to life and property.

The discussion of facility design in Section 7 on fire safety does not discuss a commitment to an industry standard on the inspection of fire protection systems. Section 7.3 of NUREG 1520 states that an applicant should provide commitments pertaining to fire safety management, including inspection, testing, and maintenance.

AES Response to NRC RAI:

The EREF will conform to the requirements of NFPA 25 for inspection, testing, and maintenance of water-based fire protection systems. Commitments made to NFPA standards 13, 14, 20, 22, and 24 in SAR Chapter 7 invoke NFPA 25 by reference for all of the proposed water-based fire protection systems at EREF. A commitment to NFPA 25 was also made in SAR Section 3.3.7 (as listed in SAR Table 3.3-10).

Associated EREF License Application Revisions:

The response to the RAI does not require any changes or additions to be made to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: FS-2

Text of NRC RAI:

Safety Analysis Report, Section 7.3.1, pp. 7.3-1 through 7.3-2

Describe the types of fireproofing intended to be used on the various structural members. Describe the measures that will be employed to confirm the proper application of fireproofing during construction and that the application remains intact over the life of the building.

The regulation 10 CFR 70.22(a)(7) requires the applicant to provide a description of equipment and facilities which will be used by the applicant to protect health and minimize danger to life or property.

In accordance with the International Building Code, Type 1B construction requires that various structural elements have specific fire resistance ratings. This rating may be obtained through various means (constructed coverings, spray applied coatings, etc). The discussion of building construction in Section 7 on fire safety does not discuss how the engineering calculations used to design the fire resistance are field verified, nor does it discuss how those structural elements protected are inspected over the life of the building to verify that the required rating is the maintained.

AES Response to NRC RAI:

AES may use one or a combination of different materials to achieve the required fire-resistance ratings for structural steel and other structural members. Where spray-applied or troweled fireproofing is applied to structural steel to achieve fire-resistance rating, the application and inspection process will conform to the special inspection requirements of International Building Code, 2006 edition, Section 1704.10, *Sprayed fire-resistant material* or Section 1704.11, *Mastic and intumescent fire-resistant coatings,* as applicable.

AES will develop, as part of its fire protection program, criteria for periodic inspection of fireproofing over the life of the facility. This would be expected to be primarily visual inspection for fireproofing integrity with repair criteria for materials found to be damaged or degraded (i.e., cracking, flaking, separation, etc.). Where fireproofing is part of an IROFS fire barrier, any associated inspection procedure will conform to Eagle Rock QAPD requirements for the assigned quality level.

While the detailed design has not been developed at this time for these components, this detail will be available for the NRC to conduct the operability readiness and management measures verification review to verify that management measures that ensure compliance with the performance requirements of 10 CFR 70.61 have been implemented and confirm that the facility has been constructed and will be operated safely and in accordance with the requirements of the license.

Similar to the Materials Licenses for the National Enrichment Facility and the USEC Facility, AES expects that the NRC will include a license condition in the Materials License that states:

> "Introduction of UF_6 into any module of the EREF shall not occur until the Commission completes an operational readiness and management measures verification review to verify that management measures that ensure compliance with the performance requirements of 10 CFR 70.61 have been implemented and confirms that the facility has been constructed and will be operated safely and in accordance with the requirements of the license. The licensee shall provide the Commission with 120 days advance notice of its plan to introduce UF_6 in any module of the EREF."

Associated EREF License Application Revisions:

The response to the RAI does not require any changes or additions to be made to the EREF License Application.

Commitments:

None

Attachments:

None

References:

International Building Code, 2006 edition

NRC RAI Number: FS-3

Text of NRC RAI:

Safety Analysis Report, Section 7.4, pp. 7.4-1

Section 7.4 states that a combustible silicone oil-based heat transfer media is used by the UF6 cold traps. Provide the combustion characteristics of this oil (flash point, fire point, heat of combustion, etc). In addition, confirm that all lubricating oil is self-contained in the various pumps, fans, centrifuge drives, etc (i.e., no lube oil systems).

The regulation 10 CFR 70.22(a)(7) requires the applicant to provide a description of equipment and facilities which will be used by the applicant to protect health and minimize danger to life or property. In addition, the regulation 10 CFR 70.62(c)(1)(iii) requires that the integrated safety analysis identifies facility hazards that could effect the safety of licensed materials and thus present an increased radiological risk. Additionally, 10 CFR 70.65(b)(3) requires a description of each process analyzed in the ISA in sufficient detail to understand the theory of operation.

AES Response to NRC RAI:

EREF may use one or more different brands of silicone oil as heat transfer media. Final oil(s) selection will be made based on final equipment selection. The following materials were proposed in initial design and evaluated as part of the Integrated Safety Analysis:

Kryo 60 (Lauda)

- Flash point >62°C (>144°F); Fire Point >110°C (230°F)

DW Therm (Huber)

- Flash point >101°C (>214°F)

The key point for use of silicone oil with respect to fire safety is that the oil for each heater/chiller system remain at a maximum temperature below the oil flash point temperature (with margin). High temperature switch cutout controls for individual units are set below the flash point. These attributes were confirmed in initial design and will be reconfirmed when final equipment and heat transfer media (oils) are selected. The other requested characteristics (fire point, heat of combustion, etc.) of individual oils will be available from product literature and/or material safety data sheets (MSDS).

All lubricating oil is self-contained within equipment. Lubricating oil used in UF_6 process pumps (i.e., vacuum pumps) and in centrifuge drives is perfluoropolyether (PFPE) and is non-combustible. There are no lubricating oil circulating or supply systems in UF_6 process structures.

Associated EREF License Application Revisions:

The response to the RAI does not require any changes or additions to be made to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: FS-4

Text of NRC RAI:

Safety Analysis Report, Section 7.5.1, pp. 7.5-1 to 7.5-5

Provide information on the type of tanks used for fire protection water supply. What material is the tank constructed out of? Does the design include any seismic considerations?

The regulation 10 CFR 70.22(a)(7) requires the applicant to provide a description of equipment and facilities that will be used by the applicant to protect health and minimize danger to life or property.

NFPA 22, "Standard for Water Tanks for Private Fire Protection," provides general guidance for various tanks constructed of various materials. Not all tanks are required by NFPA 22 to have seismic loading factored into their design, therefore a commitment to NFPA 22 does not adequately explain the design of the water supply system. Section 7.3 of NUREG 1520 states that an applicant should provide commitments pertaining to fire protection systems, including water supplies.

AES Response to NRC RAI:

The fire protection water supply tanks will be welded carbon steel gravity suction tanks with internal surfaces coated with epoxy. The external surfaces of the tanks will be coated with an epoxy primer and covered with external insulation. The fire protection water supply storage tanks shall adhere to the general requirements imposed by NFPA 22, 2008 Edition with specific requirements related to tank design, erection and testing to be in accordance with AWWA D100, 2005 Edition "Standard for Welded Steel Tanks for Water Storage" including seismic design considerations.

Associated EREF License Application Revisions:

The response to the RAI does not require any changes or additions to be made to the EREF License Application.

Commitments:

None

Attachments:

None

References:

The following additional public reference was utilized in this response:

AWWA D100, Standard for Welded Steel Tanks for Water Storage, American Water Works Association, 2005.

NRC RAI Number: FS-5

Text of NRC RAI:

Safety Analysis Report, Section 7.5.1, pp. 7.5-1 to 7.5-5

Provide clarification on the occupant notification provided by the fire alarm system. Are both audible and visual notification provided to all occupants? How are hearing impaired individuals notified? Are there any areas of potentially high ambient noise levels that may require alternate notification techniques? Are all portions of the fire alarm system designed and installed in accordance with NFPA 72, "National Fire Alarm Code?"

The regulation 10 CFR 70.22(a)(7) requires the applicant to provide a description of equipment and facilities that will be used by the applicant to protect health and minimize danger to life or property. Additionally, 10 CFR 70.64(b)(1) requires the design to be based on defense-in-depth practices.

Section 7.5.1.7 of the submittal, only states that the fire alarm control panel is installed in accordance with NFPA 72. More detail is needed on the fire alarm system design to determine its robustness as a defense-in-depth measure.

AES Response to NRC RAI:

Fire alarm notification appliances will be provided in all normally or periodically occupied structures at the EREF that are provided with fire detection and suppression systems. Such appliances will have audible and visual warning capability to ensure occupants are aware of any detection alarm or suppression system activation. Remote or isolated facilities and outside areas (e.g., cylinder storage pads, trailer area, etc.) that are not provided with fire alarm or suppression systems will be provided with audible appliances (i.e., public address) and/or radio notification to ensure site-wide emergency announcements can be heard in accordance with the EREF Emergency Plan.

Due to the need for aural comprehension within the process areas of multiple warning systems (e.g., fire, leak detection, CAAS, etc.) hearing-impaired employees will be restricted to work assignments in areas where visual warning appliances are provided.

At present, the area inside the cascade thermal enclosure is known to be a high ambient noise environment. This area is not normally occupied but will be provided with visual strobe fire alarm notification appliances.

All portions of the fire alarm system will be designed and installed in accordance with NFPA 72, *National Fire Alarm Code.* This was identified as a commitment standard in SAR Section 3.3.7 (as listed in Table 3.3-10).

Associated EREF License Application Revisions:

The response to the RAI does not require any changes or additions to be made to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: FS-6

Text of NRC RAI:

Safety Analysis Report, Section 7.5.2, pp. 7.5-5 to 7.5-7

Provide a more detailed description of the number of people trained to participate on the facility fire brigade. Is there a minimum number of trained personnel available for any given shift?

The regulation 10 CFR 70.22(a)(7) requires the applicant to provide a description of equipment and facilities that will be used by the applicant to protect health and minimize danger to life or property. In addition, 10 CFR 70.22(a)(8) requires the applicant to provide proposed procedures to protect health and minimize danger to life and property.

AES Response to NRC RAI:

As noted in SAR Section 7.5.2.1, there will be a minimum of five trained fire brigade members available per shift. This complement will include an individual trained as a criticality safety officer.

Associated EREF License Application Revisions:

The response to the RAI does not require any changes or additions to be made to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: FS-7

Text of NRC RAI:

Safety Analysis Report, Figure 7.5-2

Provide a detailed diagram for each building showing sprinkler system coverage.

The regulation 10 CFR 70.22(a)(7) requires the applicant to provide a description of equipment and facilities that will be used by the applicant to protect health and minimize danger to life or property. Additionally, 10 CFR 70.64(b)(1) requires the design to be based on defense-in-depth practices.

It is noted in the referenced figure that certain buildings require evaluation for moderator control and may have limited or no sprinkler coverage in select areas. Details of that evaluation and a more precise description of sprinkler system coverage are needed to determine the system's robustness as a defense-in-depth measure.

AES Response to NRC RAI:

For all buildings listed in SAR Figure 7.5-2, except for areas specifically excluded on that figure, sprinkler coverage will be area-wide unless criticality safety parameters cannot be met.

Information on the methodology for evaluating the need for moderator control to maintain criticality safety parameters and information on the interaction between sprinkler protection and criticality safety was provided in SAR Sections 5.1.2 and 7.3.8. Additional information on the evaluation methodology is provided in the EREF Fire Hazards Analysis, Appendix A. This document is available for NRC review.

The following plant areas do not have sprinkler coverage in order to meet criticality safety parameters committed to in the SAR.

Separations Building Module:

- UF_6 Handling Area – limited-area sprinkler protection with coverage omitted over the southern portion of the room(s) due to the presence of enriched product system vacuum piping that leads to Product System takeoff stations containing 30B cylinders (non-mass controlled, unfavorable geometry).
- Process Services Corridor, 2nd floor – limited-area sprinkler protection in the corridor at this elevation due to the presence of enriched product system vacuum piping that leads to Product System takeoff stations containing 30B cylinders (non-mass controlled, unfavorable geometry).

In the remaining areas where moderation control is required, it is expected that Nuclear Criticality Safety Analysis (NCSAs) will demonstrate that criticality safety parameters can be met.

While the detailed design has not been developed at this time for these areas, this detail will be available for the NRC to conduct the operability readiness and management measures verification review to verify that management measures that ensure compliance with the performance requirements of 10 CFR 70.61 have been implemented and confirm that the facility has been constructed and will be operated safely and in accordance with the requirements of the license.

Similar to the Materials Licenses for the National Enrichment Facility and the USEC Facility, AES expects that the NRC will include a license condition in the Materials License that states:

> "Introduction of UF_6 into any module of the EREF shall not occur until the Commission completes an operational readiness and management measures verification review to verify that management measures that ensure compliance with the performance requirements of 10 CFR 70.61 have been implemented and confirms that the facility has been constructed and will be operated safely and in accordance with the requirements of the license. The licensee shall provide the Commission with 120 days advance notice of its plan to introduce UF_6 in any module of the EREF."

As noted in the SAR, application of fire hose streams will be controlled by the presence of a criticality safety officer assigned to plant fire brigade/off-site fire department response.

Associated EREF License Application Revisions:

The response to the RAI does not require any changes or additions to be made to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: D-1

Text of NRC RAI:

Section 10.1.6 states that decommissioning will take about 8 years. 10 CFR 70.38(h) requires that decommissioning be completed no later than 24 months following the initiation of decommissioning. 10 CFR 70.38(i) allows the Commission to approve a request for an alternate schedule for completion of decommissioning if the alternative is warranted by consideration of 5 factors specified in 70.38(i)(1)-(i)(5). Request an alternate schedule for decommissioning and provide justification for the longer schedule.

AES Response to NRC RAI:

AES hereby formally requests an alternate schedule for decommissioning. As stated in Section 10.1.6.1 of the Safety Analysis Report (submitted as part of the EREF License Application):

"The four Separation Building Modules will be shutdown in sequence starting with Separations Building Module 1. Since only low radiation levels exist at this facility, decommissioning may begin immediately following the permanent shutdown of the first series of cascades in a Separations Building Module. The decommissioning of a single Separations Building Module 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. Dismantling and decontamination of the equipment in the four Separations Building Modules will be performed in a phased approach such that the decommissioning of all four Separations Building Modules is completed within an eight year time frame."

10 CFR 70.38(i) states that:

"The Commission may approve a request for an alternate schedule for completion of decommissioning of the site or separate building or outdoor area, and license termination if appropriate, if the Commission determines that the alternative is warranted by consideration of the following:

(1) Whether it is technically feasible to complete decommissioning within the allotted 24-month period;

As stated above, the decommissioning of a single Separations Building Module 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. Dismantling and decontamination of the equipment in the four Separations Building Modules (SBMs) will be performed in a phased approach such that the decommissioning of all four SBMs is completed within an eight year time frame.

The time estimate for decommissioning the centrifuges and associated equipment within the SBMs was established by Enrichment Technology Corporation (ETC). They are the supplier for the centrifuge technology, and will be responsible for the decommissioning of the centrifuge related, classified equipment in each of the SBMs. This equipment represents the majority of the equipment requiring decontamination and dismantlement within the SBMs.

The primary reason for the 3-year time period is the quantity of the centrifuges and associated equipment that must be decommissioned and the fact that this equipment is classified. As a result, special precautions and processes must be utilized to reduce this equipment to forms that may be disposed of without revealing any classified information. The three year time period represents a realistic estimate to decommission and declassify this equipment given the processes that must be used and the current state of technology of the equipment that will be used to perform the decommissioning/declassification. It is not feasible to significantly increase the numbers of decommissioning personnel or equipment without appreciably decreasing efficiency and increasing cost. Therefore, significantly reducing the 3-year period for the decommissioning of the centrifuge related equipment within a single SBM and overall eight year period is not technically feasible.

An additional consideration is the physical size of the SBM itself. A single SBM has an enclosed volume of approximately 4,800,000 cubic feet. Given that volume, a significant amount of time (estimated as 1½ years) will be required to perform the applicable radiation surveys and any necessary decontamination. This time estimate is based on similar work performed at decommissioned nuclear power plants using state of the art equipment and processes. A significant decrease in the time required for surveying and decontamination of the SBMs using currently available personnel, equipment, and techniques is not technically feasible at this time.

(2) Whether sufficient waste disposal capacity is available to allow completion of decommissioning within the allotted 24-month period;

There is sufficient waste disposal capacity to allow completion within two years; however, shipment of the projected volume of waste, 270,000 cubic feet via an estimated 1352 separate shipments, in a 24-month period is not practical. See (5) below.

(3) Whether a significant volume reduction in wastes requiring disposal will be achieved by allowing short-lived radionuclides to decay;

There will be no significant volume reduction in wastes requiring disposal achieved by allowing short-lived radionuclides to decay. All decommissioning waste is assumed to be low level, Class A waste. This waste classification and thereby the waste volume will not change as a result of extending the decommissioning period.

(4) Whether a significant reduction in radiation exposure to workers can be achieved by allowing short-lived radionuclides to decay; and

The reduction in radiation exposure to workers achieved by allowing short-lived radionuclides to decay will be insignificant. However, the extended period will allow the optimization of personnel compared with a 24-month decommissioning time frame thereby reducing the man-hours expended to decommission. This will result in a reduced total dose to workers for the SBM decommissioning. The expected reduction has not been quantified at this time.

(5) Other site-specific factors which the Commission may consider appropriate on a case-by-case basis, such as regulatory requirements of other government agencies, lawsuits, ground-water treatment activities, monitored natural ground-water restoration, actions that could result in more environmental harm than deferred cleanup, and other factors beyond the control of the licensee."

If a decommissioning time period of 24 months was imposed, there would be additional impacts, including:

- An estimated 1352 shipments of waste that results in an average of about 3 shipments per working day occurring continuously over a 24-month decommissioning period.
- Impacts on the waste disposal site associated with handling this volume of waste, in addition to their normal waste stream.
- Additional environmental impacts due to the increased personnel on-site.
- Additional costs resulting from a non-optimized decommissioning schedule.

These impacts have not been quantified at this time.

Given that decommissioning of the EREF is not technically feasible to complete within the allotted 24-month period and is not ALARA, AES requests that the NRC approve an alternate decommissioning schedule.

Associated EREF License Application Revisions:

This response does not require a revision to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: D-2

Text of NRC RAI:

Revise the initial radiation survey performed prior to initial operation such that it is adequate to establish background for use as a reference area for the final survey at decommissioning time or provide other explanation. NUREG-1757, Consolidated Decommissioning Guidance, Volume 2 Characterization, Survey, and Determination of Radiological Criteria (Revision 1), contains guidance for performing a background survey. The 10 samples discussed in Environmental Report Sections 3.3 and 3.11 are too few and are located at the site boundary or outside it; none are located within the site itself. They are not sufficient to use for demonstration of compliance with 10 CFR Part 20 Subpart E decommissioning criteria.

AES Response to NRC RAI:

As depicted on Environmental Report Figure 3.3-14A, soil samples 4, 5, 6, 7, and 9 were performed within the footprint of the facility and soil samples 1, 2, 3, 8 and 10 were performed outside of the footprint of the facility, but within the area of the EREF site.

The following additional sample protocol is based on the guidance provided by Section A.3.4 of Appendix A to NUREG-1757. 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 TSB, 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 will be taken to characterize foundation soils prior to UF_6 cylinders arriving on-site.

Associated EREF License Application Revisions:

This RAI response does not require a revision to the EREF License Application.

Commitments:

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 TSB, 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 will be taken to characterize foundation soils prior to UF_6 cylinders arriving on-site.

Attachments:

None

NRC RAI Number: D-3

Text of NRC RAI:

The cost estimate in the SAR for decommissioning of the Separations Building Modules has been factored to address a 6.6 M SWU plant, as follows (the tables use 3 M SWU and 6 M SWU instead of 3.3 and 6.6):

- The length of piping for a 6 M SWU facility is assumed to be twice that for a 3 M SWU facility (Note 5 to Table 10.1-1A).
- The length of ventilation/ductwork for a 6 M SWU facility is assumed to be twice that for the 3 M SWU facility (Note 4 to Table 10.1-1D).
- Areas to be contaminated for a 6 M SWU facility are assumed to be twice the areas for a 3 M SWU facility (Note 5 to Table 10.1-1E).
- The amount of electricity (if used to replace natural gas) required for a 3 M SWU facility is doubled for a 6 M SWU facility (Note 2 to Table 10.1-11).
- Piping length for a 6 M SWU facility is assumed to increase by only 50% (Note 5 to Table 10.1-1E).

Provide an explanation for the different factor for piping length.

The regulations in 10 CFR 70.25 (a)(1) require applicants for a license for a uranium enrichment plant to submit a decommissioning funding plan. Section A.3 of NUREG-1757, Volume 3, outlines the level of detail that the cost estimate should contain.

AES Response to NRC RAI:

A 3.3 million SWU facility is a nominal 3 million SWU facility. Likewise, a 6.6 million SWU facility is a nominal 6 million SWU facility.

During the development of Revision 1 of the EREF License Application, the assumptions regarding the decommissioning funding analysis were reviewed to determine how the assumptions should be modified. For a number of assumptions, including the length of piping in the Separations Buildings (Table 10.1-1A), length of GEVS ventilation/ductwork throughout the Plant (Table 10.1-1D), areas to be contaminated in the Blending, Sampling, and Preparation Building (Table 10.1-1E), and the amount of electricity required (Table 10.1-11), the conservative action was to simply double the value that had been presented in the original application since the 3 million SWU estimate basis (e.g., building size, amount required) doubled.

For the Blending, Sampling, and Preparation Building (BSPB), the 3 million SWU estimate of contaminated piping was based on the square footage of the entire BSPB, not just the contaminated areas. Likewise, the estimate of contaminated piping in the 6 million SWU BSPB provided in Table 10.1-1E was based on the total square footage of the entire building (i.e., linear feet of contaminated piping per square foot). Preliminary designs of the 6 million SWU BSBP showed an increase of about 50% in total square footage compared to the 3 million SWU BSBP; therefore, the amount of contaminated piping within the BSBP was increased by 50%, and not doubled.

Associated EREF License Application Revisions:

This response does not require a revision to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: D-4

Text of NRC RAI:

The SAR does not describe any experience that Enrichment Technology Company (ETC) has in decommissioning. Provide a description of the experience, if any, that ETC may have relevant to planning and conducting decommissioning of the Separation Buildings Modules and incorporate into the SAR.

Section A.3.1 of NUREG-1757, Volume 3, indicates that an applicant should assume that a third party contractor will perform the work. The level of experience a contractor has may reflect how a contractor is able to complete the work at reasonable costs.

AES Response to NRC RAI:

A description of ETC's experience in the decommissioning of the classified portion of a centrifuge type uranium enrichment facility such as the Eagle Rock Enrichment Facility (EREF) is provided below. As can be seen from this description, ETC has the requisite experience to provide accurate cost estimates for this type of work.

Decommissioning and disposal of the plant at the end of its lifetime is the responsibility of the operator. Within this process, ETC is responsible for the declassification of the centrifuges and related classified components. This includes the dismantling of crashed and intact centrifuges into a suitably decommissioned level to avoid proliferation of dual use items.

ETC has decommissioning experience related to the E21 Enrichment Plant (at Capenhurst, UK) for early generations of block-mounted centrifuges and the SP3 Separation Plant at Almelo, Netherlands for early generation centrifuges. ETC has maintained a decommissioning cost model based on costs and methodologies of work previously performed.

Due to the continued high performance of the current generation of gas centrifuges no site has yet needed, or is likely in the short term, to decommission a plant containing centrifuge machines similar to those proposed for the EREF. ETC has experience in taking both intact and crashed centrifuges from these plants for autopsy and from refurbishment campaigns for its customers. Centrifuges have been manually dismantled, decontaminated and declassified by ETC-D in Jülich, Germany, with the resulting parts safely disposed of – this includes both TC12 and later generation development centrifuges. Work has been performed by ETC scaling the manual methodologies used in Jülich into a semi-automated line using a design concept based on that currently deployed on the SP3 decommissioning line. A cost estimate for this concept was included in the decommissioning cost model.

Associated EREF License Application Revisions:

Section 10.1.4 of the EREF SAR will be revised to incorporate the discussion provided in response to this RAI.

Commitments:

The markup of SAR Section 10.1.4 provided in Enclosure 2.2 will be incorporated into Revision 2 of the EREF License Application.

Attachments:

Enclosure 2.2: Markup of SAR Section 10.1.4

NRC RAI Number: D-5

Text of NRC RAI:

In several tables in the SAR, notes to the tables state that "Total dimensions not used in estimating model" (e.g., Note 4 to Table 10.1-1A; Note 1 to Table 10.1-1B; Note 1 to Table 10.1-1C; Note 1 to Table 10.1-1D; Note 1 to Table 10.1-1E; and Note 1 to Table 10.1-1F).

Frequently, these notes are associated with a statement that "1 Lot" of items will be decontaminated. These items, which include walls, floors, ceilings, tanks, valves, hand tools, consumables, scaffolding, and miscellaneous pieces of equipment, do not appear to be consistently the same size or to share other characteristics that make their decommissioning costs potentially similar. Decommissioning/decontaminating "1 lot of tanks," for example, is likely to involve significantly different costs from decommissioning "1 lot of hand tools or consumables." Often, another note explains, with respect to these items, that "Allocation based on European decommissioning experience."

Provide a more detailed explanation of the estimating model functions to provide a cost estimate in these cases, including what that decommissioning experience consists of, when it occurred, whether it was carried out by third parties whose costs are reflected in the estimates. Clarify whether the European decommissioning experience is a reference to ETC activities.

Section A.3 of NUREG-1757, Volume 3, outlines the level of detail that the cost estimate should contain. The labor estimates, material costs, and other factors of the cost estimate should have a clear and reasonable basis.

AES Response to NRC RAI:

As discussed in Section 10.1.4 of the EREF FSAR, the estimate for decommissioning funding is based on a cost estimate prepared by Enrichment Technology Corporation (ETC) for the classified portion of the decommissioning (97%) and on similarities between the EREF and the reference plant for the remainder (3%). The ETC cost estimate was provided at a level of detail consistent with maintaining the classified nature of the information. The term lot was used in the non-proprietary and non-classified version of the SAR where further level of detail would compromise the classified nature of the information. The classified portion of the decommissioning cost estimate dealing with the centrifuges and associated pipework was transmitted directly to the NRC by ETC under an AES cover letter dated December 30, 2008.

The use of the term "Lot" and the level of detail provided for the EREF decommissioning cost estimate is consistent with that used by the reference plant (the National Enrichment Facility) and approved by the NRC in NUREG-1827, the Safety Evaluation Report for the National Enrichment Facility in Lea County, New Mexico.

The "allocation based on European decommissioning experience" refers to the decommissioning of gas centrifuge enrichment facilities in Europe and would include ETC and non-ETC activities. This is explained in greater detail in the response to NRC RAI D-4.

Associated EREF License Application Revisions:

This response does not require a revision to the EREF License Application.

Commitments:

None

Attachments:

None

References:

R. M. Krich (AES) letter to U.S. Nuclear Regulatory Commission, "Submittal of Classified Portions of the Application for a Uranium Enrichment Facility License Under 10 CFR 70, Domestic Licensing of Special Nuclear Material," dated December 30, 2008.

NUREG-1827, Safety Evaluation Report for the National Enrichment Facility in Lea County, New Mexico, U.S. Nuclear Regulatory Commission, published June 2005

NRC RAI Number: D-6

Text of NRC RAI:

In Table 10.1-1C, no dimensions are provided for the Third Floor Maintenance Facility or the Third Floor Decontamination Areas in the Technical Support Building. Total dimensions are provided for other components of that building. Provide estimates of the total dimensions to be decommissioned/decontaminated for the Maintenance and Decontamination areas, or explain why no dimensions are provided.

Section A.3 of NUREG-1757, Volume 3, outlines the level of detail that the cost estimate should contain.

AES Response to NRC RAI:

The floor, wall, and ceiling areas associated with the Third Floor Maintenance Facility and the Third Floor Decontamination Areas in the Technical Support Building are included in the overall floor, wall, and ceiling areas for the TSB. A new note (Note 4) will be added to Table 10.1-1C to show this.

Associated EREF License Application Revisions:

SAR Table 10.1-1C will be revised to include a new Note 4 that discusses that the floor, wall, and ceiling areas of the Third Floor Maintenance Facility and the Third Floor Decontamination Areas are included in the overall floor wall and ceiling areas.

Commitments:

The markup of SAR Table 10.1-1C provided in Enclosure 2.2 will be incorporated into Revision 2 of the EREF License Application.

Attachments:

Enclosure 2.2: Markup of SAR Table 10.1-1C

NRC RAI Number: D-7

Text of NRC RAI:

In Table 10.1-3, Note 4, states that "Specific details of decontamination method not defined at this time." However, detailed estimates of the total hours necessary to conduct decontamination of the various components listed in the table are provided. Provide an explanation of the accurately and completeness of the estimated total hours necessary to complete tasks, given that the details of the method are not yet defined.

Section A.3 of NUREG-1757, Volume 3, outlines the level of detail that the cost estimate should contain.

AES Response to NRC RAI:

The detailed estimates of the total hours necessary to conduct decontamination of the various components listed in Table 10.1-3 are developed based on information from the National Enrichment Facility (i.e., the reference plant), appropriately factored for design differences between the EREF and the reference plant. This is explained in Section 10.1.4 of the EREF SAR. The specific details of the decontamination method are not needed. The general decontamination methods to be used at the EREF are discussed in Section 10.1.6 and are consistent with those to be used at the reference plant.

Note 4 will be changed to read "Specific details of decontamination method not defined at this time. General decontamination methods to be used at EREF are discussed in Section 10.1.6."

Associated EREF License Application Revisions:

SAR Table 10.1-3, Note 4 will be revised to include reference to the general methods of decontamination provided in Section 10.1.6.

Commitments:

The markup of SAR Table 10.1-3 provided in Enclosure 2.2 will be incorporated into Revision 2 of the EREF License Application.

Attachments:

Enclosure 2.2: Markup of Table 10.1-3

NRC RAI Number: D-8

Text of NRC RAI:

Safety Analysis Report, Table 1 0.1-4 and 1 0.1 -6

Notes to Tables 10.1-4 and 10.1-6 state that European experience with decommissioning gas centrifuge uranium enrichment plants has been that there is no resulting radiological contamination of the facility grounds. Provide a more detailed description of the European experience upon which these notes are based.

Section A.3 of NUREG-1757, Volume 3, outlines the level of detail that the cost estimate should contain.

AES Response to NRC RAI:

Previous decommissioning of a diffusion plant, the E21 Centrifuge Plant at Capenhurst and the SP3 Plant at Almelo have all been performed without any resulting contamination of facility grounds and the resulting land reused.

At the end of lifetime of a centrifuge plant, the UF_6 inventory will be removed and the cascade pipework and centrifuges purged with nitrogen. Centrifuge plants run under vacuum with a high degree of cleanliness required for reliable performance; the UF_6 inventory per machine is low and the centrifuges isolated if failure occurs during their lifetime – these all contribute to reducing the amount of UF_6 breakdown deposits within the cascade and centrifuge at the end of life. This reduces the risk of contamination during decommissioning. While pipework will contain some UF_6 breakdown, in general this is low in level and can be managed through the use of controlled access areas, and the use of correct tools and methods during the decommissioning process. The floor area in the cascade hall is sealed; any contamination can easily be detected through surface monitoring and removed through cleaning if necessary, thereby providing reasonable assurance that no contamination will be spread to the facility grounds.

Associated EREF License Application Revisions:

None

Commitments:

None

Attachments:

None

NRC RAI Number: D-9

Text of NRC RAI:

In Table 10.1-10, "Packaging, Shipping, and Disposal of Radioactive Waste," the multiplication of Disposal Volume in ft^3 times unit cost per ft^3 sometimes resulted in different total disposal costs from those found in the table—e.g., 30,512 x $410=$12,509,920 rather than $12,522,000; 218,951 x $220=$48,169,220 rather than $48,193,000).

If the results in the table are subject to rounding, this should be noted in the table, or provide other explanation for the discrepancies.

Section A.3 of NUREG-1757, Volume 3, outlines the level of detail that the cost estimate should contain.

AES Response to NRC RAI:

The discrepancies in the results in Table 10.1-10(a) are the result of rounding. Notes have been added to Table 10.1-10(a) and Table 10.1-10(b) to reflect this.

Associated EREF License Application Revisions:

SAR Table 10.1-10(a) and Table 10.10-10(b) will be revised to include notes that state the values provided for Disposal weight in ft^3 and tons, Unit Cost, and Total Processing Costs reflect rounding.

Commitments:

The markup of SAR Table 10.1-10(a) and Table 10.1-10(b) provided in Enclosure 2.2 will be incorporated into Revision 2 of the EREF License Application.

Attachments:

Enclosure 2.2: Markup of SAR Table 10.1-10(a) and Table 10.1-10(b).

NRC RAI Number: D-10

Text of NRC RAI:

In Table 10.1-15, Note 4 explains that the vacuum processes in the Separations Building result in an absence of contamination of floors, walls, and ceilings. The same Note 4 also states that a conservative allowance is provided for cleaning of floors, walls, and ceilings in the final Decommissioning Facility and the Technical Support Building. Assuming that the vacuum process will not be operational in those buildings, provide a more detailed explanation for the very low estimates for cleaning ceilings, floors, and walls in those buildings.

Section A.3 of NUREG-1757, Volume 3, outlines the level of detail that the cost estimate should contain.

AES Response to NRC RAI:

The decommissioning estimates assume that all floors, walls, ceilings in potentially contaminated areas, including, but not limited to, the Decommissioning Facility and the Technical Support Building, are contaminated. The estimates for cleaning of the floors, walls, and ceilings are consistent with similar work performed for decontamination of floors, walls, and ceilings at decommissioning nuclear power plants. The levels of contamination encountered during the decommissioning of nuclear power plants far exceed the levels anticipated at EREF, therefore the estimates are conservative.

The information in Note 4 regarding vacuum processes is limited to the Separations Building Modules. The normal operation of the gas centrifuges within the Separations Building Modules requires vacuum conditions. A loss of vacuum in the system would cause it to automatically shutdown, thereby minimizing the potential for contamination. Any contamination resulting from normal maintenance and repair will be cleaned at that time. Therefore, the potential for contamination is very low.

Associated EREF License Application Revisions:

Note 4 of SAR Table 10.1-5 is revised to state: All floors, walls, ceilings in potentially contaminated areas are assumed to be contaminated. The estimates for cleaning of the floors, walls, and ceilings are consistent with similar work performed for decontamination of floors, walls, and ceilings at decommissioning nuclear power plants. The levels of contamination encountered during the decommissioning of nuclear power plants far exceed the levels anticipated at EREF, therefore the estimates are conservative.

Commitments:

SAR Table 10.1-5 will be revised to include the markup provided in Enclosure 2.2.

Attachments:

Enclosure 2.2: Markup of SAR Table 10.1-5

NRC RAI Number: D-11

Text of NRC RAI:

In Table 10.1-16, no estimate is included for contingencies. Application of a 25% contingency factor is discussed Section 10.2.2 of the SAR. Revise this table to show a minimum of 25% of the costs for contingencies.

Section A.3.1.2.3 of NUREG-1757, Volume 3, specifies that the cost estimate should apply a contingency factor of 25% to the sum of all estimated decommissioning costs.

AES Response to NRC RAI:

The contingency factor of 25% is not applicable to Table 10.1-16. This table provides the cost of the decommissioning of the first SBM only. Section 10.2.2 applies the 25% contingency to the cost of the total of the decommissioning of the first SBM from Table 10.1-16, the decommissioning of the "Other Buildings" from Table 10.1-14, and the tails produced during that first operational period.

Associated EREF License Application Revisions:

This response does not require a revision to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: MM-1

Text of NRC RAI:

Safety Analysis Report. Section 11.1.4

Chapter 11.1.4, "Change Control," states that each change to the facility or activities of personnel will be evaluated in accordance with the requirements of 10 CFR 70.72.

10 CFR 70.72(c) allows the licensee to make changes without prior NRC approval if other conditions, primarily associated with the facility's safety program established in accordance with 10 CFR 70.62, "Safety Program and Integrated Safety Analysis," are met.

What criteria will be used to evaluate changes to the facility and activities of personnel which are not included in the safety program in order to determine whether prior NRC approval is required? How will this evaluation be documented and at what frequency will these changes be provided to the NRC after implementation?

AES Response to NRC RAI:

Following the receipt of the Materials License, the license basis documents, submitted to the NRC, that comprise the License Application will be maintained in accordance with the regulations defined below. In addition, AES will comply with the documentation and notification frequency requirements contained within the various regulatory requirements and commitments.

License Basis Document	***Regulatory Requirements and Commitments***
Safety Analysis Report	***10 CFR 70.72***
Environmental Report	***10 CFR 70.72 and 10 CFR 51.22***
Integrated Safety Analysis Summary	***10 CFR 70.72 Note: The ISA is maintained in accordance with 10 CFR 70.62***
Quality Assurance Program Description	***10 CFR 70.72 and QAPD Chapter 19***
Emergency Plan	***10 CFR 40.35(f), 10 CFR 70.72 and 10 CFR 70.32(i)***
Fundamental Nuclear Material Control Plan	***10 CFR 70.72 and 10 CFR 70.32(c)(1)(ii)***
Physical Security Plan	***10 CFR 70.72, 10 CFR 70.32(d) and 10 CFR 70.32(e)***
Standard Practice and Procedure Plan for the Protection of Classified Matter	***10 CFR 70.72 and 10 CFR 95.19***

This is consistent with the precedent established by the NRC in Material Licenses SNM-2010 and SNM-2011 for the National Enrichment Facility and the American Centrifuge Plant. License Condition 10 of Material License SNM 2010 states:

> "The licensee shall conduct authorized activities at the NEF in accordance with the statements, representations, and conditions, or as revised in accordance with Section 19 of the Quality Assurance Program Description, 10 CFR 40.35(f), 10 CFR 51.22, 10 CFR 70.32, 10 CFR 70.72 or 10 CFR 95.19 in: a. Application for Material License...b. Safety Analysis Report...c. Environmental Report... d. Physical Security Plan...e. Fundamental Nuclear Material Control Plan...f. Quality Assurance Program Description...g. Emergency Plan...h. Standard Practice Procedure Plan for the Protection of Classified Matter...".

AES will maintain the Safety Analysis Report as a living document in accordance with 10 CFR 70.72. 10 CFR 70.72(a) requires:

> "The licensee shall establish a configuration management system to evaluate, implement, and track ***each change*** to the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel ...assure that the following are addressed prior to implementing any change...". In addition, 10 CFR 70.72(c) states: "The licensee may make changes to the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel, without prior Commission approval, if the change...".

Given that the scope of 10 CFR 70.72 includes any change to the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel, AES believes this includes the information provided within the Safety Analysis Report.

In addition to 10 CFR 70.72, AES has proposed a method for maintaining the QAPD (i.e., Appendix A of the Safety Analysis Report). This method is described in Chapter 19 of the QAPD. Section 19.2 of the QAPD requires any changes that reduce commitments in the approved QAPD will be submitted to the NRC for review and approval prior to implementation.

Associated EREF License Application Revisions:

This response does not require a revision to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: MM-2

Text of NRC RAI:

Safety Analysis Report. Section 11.1.4.1

Chapter 11.1.4.1, "Design Phase," states that prior to issuance of the license, AES will notify the NRC of potential changes that reduce the level of commitments or margin of safety in the design bases of QA level 1 and 2 items and activities. Please clarify that AES will provide these changes to the NRC for review and approval prior to issuance of a license.

10 CFR 70.23(a)(3) and (4) require that the NRC determine that the applicant's proposed equipment and facilities and proposed procedures are adequate to protect health and minimize danger to life or property.

AES Response to NRC RAI:

Prior to the NRC's issuance of the Materials License for the EREF, AES will submit potential changes that reduce the level of commitments or margin of safety in the design bases of QA Level 1 and 2 items and activities to the NRC for review and approval prior to implementation of the change. SAR Section 11.1.4.1 will be modified as depicted in Enclosure 2.2.

Following the NRC's issuance of the Materials License for the EREF, AES will maintain license basis documents as described in the response to NRC RAI MM-1.

Associated EREF License Application Revisions:

SAR Section 11.1.4.1 will be revised as identified in Enclosure 2.2.

Commitments:

The revision to SAR Section 11.1.4.1 contained in Enclosure 2.2 will be incorporated into Revision 2 of the License Application.

Attachments:

Enclosure 2.2: Markup of SAR Section 11.1.4.1

NRC RAI Number: MM-3

Text of NRC RAI:

Chapter 11.1.4.3, "Operations Phase," describes the evaluation of modifications in accordance with 10 CFR 70.72 and states that each modification shall be evaluated for any required changes or additions to the facility's procedures, personnel training, testing program, or regulatory documents. Clarify whether regulatory documents include the ISA, ISA summary, and other safety program information developed in accordance with 10 CFR 70.62.

10 CFR 70.72(a)(6) requires that the licensee evaluate the impacts of changes on the ISA, ISA summary, and other safety program information.

AES Response to NRC RAI:

The regulatory documents referred to in Section 11.1.4.3 of the SAR include the Integrated Safety Analysis (ISA), ISA Summary, Safety Analysis Report, Quality Assurance Program Description, Environmental Report, Physical Security Plan, Emergency Plan, Fundamental Nuclear Material Control Plan, and Standard Practice and Procedure Plan for the Protection of Classified Matter. In addition, ISA Summary Section 3.1.8.1, Process Safety Information, defines process safety information. It states:

[Security-Related withheld in accordance with 10 CFR 2.390]

These documents will be subject to the requirements of 10 CFR 70.72 as established in SAR Section 3.0.3. It states: "The configuration management program is required by 10 CFR 70.72 (CFR, 2008f) and establishes a system to evaluate, implement, and track each change to the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel." In addition, SAR Section 3.0.1 states: "AES has developed procedures and criteria for changing the ISA. This includes implementation of a facility change mechanism that meets the requirements of 10 CFR 70.72 (CFR, 2008f)."

Associated EREF License Application Revisions:

This response does not require a revision to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: MM-4

Text of NRC RAI:

Chapter 11.2.4.4.2, "Special Testing" states that special tests may be conducted at the discretion of the Plant Manager. Please clarify the SAR to describe the requirement to evaluate the impact of the special test in accordance with 10 CFR 70.72 prior to conducting it.

AES Response to NRC RAI:

Special tests are considered to be a facility change or change in process safety information that may alter the parameters of an accident sequence. As described in SAR Section 3.0.2, these tests must be reviewed by the ISA method(s) as described in the ISA Summary. In addition, special tests will be reviewed in accordance with 10 CFR 70.72(a) and (c). SAR Section 11.2.4.4.2 will be modified to reflect the above discussion.

Associated EREF License Application Revisions:

SAR Section 11.2.4.4.2 of the SAR will be revised to state:

Special tests are considered to be a facility change or change in process safety information that may alter the parameters of an accident sequence. As described in SAR Section 3.0.2, these tests must be reviewed by the ISA method(s) as described in the ISA Summary. In addition, special tests will be reviewed in accordance with 10 CFR 70.72(a) and (c).

Commitments:

The change to SAR Section 11.2.4.4.2 identified in Enclosure 2.2 will be incorporated into Revision 2 of the License Application.

Attachments:

Enclosure 2.2: Markup of SAR Section 11.2.4.4.2.

NRC RAI Number: MM-5

Text of NRC RAI:

Please describe the relationship of the first paragraph in Chapter 11.4, "Procedures Development and Implementation" to this section of the SAR. This paragraph describes requirements for independent verification as being consistent with the guidance of ANSI/ANS 3.2-1994 and appears to be out of place in the SAR.

AES Response to NRC RAI:

The paragraph in SAR Section 11.4 that refers to ANSI/ANS 3.2-1994 will be eliminated. The specific guidance provided by ANSI/ANS 3.2-1994 will be added to SAR Sections 11.4.1 and 11.4.3.

SAR Section 11.4.1 will be revised to denote that independent reviews will be conducted by personnel not having direct responsibility for the work function under review.

SAR Section 11.4.3 will be revised to state: "In addition, procedural guidance exists to define when verification of significant steps is required."

Associated EREF License Application Revisions:

SAR Sections 11.4, 11.4.1, and 11.4.3 will be modified as depicted in Enclosure 2.2.

Commitments:

The changes to SAR Sections 11.4, 11.4.1, and 11.4.3 depicted in Enclosure 2.2 will be incorporated into Revision 2 of the EREF License Application.

Attachments:

Enclosure 2.2: Markup of SAR Sections 11.4, 11.4.1 and 11.4.3.

NRC RAI Number: MM-6

Text of NRC RAI:

Please clarify the periodicity of procedure reviews in Chapter 11.4 "Procedures Development and Implementation" to assure their continued accuracy and usefulness. NUREG-1520 recommend that operating procedures be reviewed at least every 5 years and emergency procedures be reviewed annually.

10 CFR 70.62(d) requires that management measures, which include certain procedures, shall ensure that IROFS are implemented and maintained such that they are available and reliable to perform their function when needed.

AES Response to NRC RAI:

SAR Section 11.4 will be revised to state:

"At a minimum all operating procedures are reviewed every 5 years and emergency procedures are reviewed every year."

Associated EREF License Application Revisions:

SAR Section 11.4 will be modified as depicted in Enclosure 2.2.

Commitments:

The change to SAR Section 11.4 depicted in Enclosure 2.2 will be incorporated into Revision 2 of the EREF License Application.

Attachments:

Enclosure 2.2: Markup of SAR Section 11.4.

NRC RAI Number: MM-7

Text of NRC RAI:

Please revise Chapter 11.4.3 "Procedures" of the SAR to clarify that temporary changes to procedures are reviewed in accordance with 10.CFR 70.72 prior to implementation and that the temporary procedures will have an approved duration as required by 10.CFR 70.72(a)(5).

AES Response to NRC RAI:

Procedure changes, including temporary changes, are subject to the requirements of 10 CFR 70.72. Besides from being a regulatory requirement, this requirement is addressed via the multiple references to 10 CFR 70.72 in the SAR.

- Section 3.0.3 of the SAR states: "The configuration management program is required by 10 CFR 70.72 (CFR, 2008f) and establishes a system to evaluate, implement, and track each change to the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel."
- SAR Section 11.4.3 states: "Temporary changes to procedures are documented, reviewed and approved with the process described in Section 11.4.4, Changes to Procedures, within 14 days of implementation." This statement will be revised to eliminate "within 14 days of implementation," because the requirements of 10 CFR 70.72 must be met prior to implementation. In addition, a reference to the requirement of 10 CFR 70.72(a)(5) will be added to defined the approved duration for the temporary change.
- SAR Section 11.4.4 states that an evaluation shall be performed in accordance with 10 CFR 70.72 for changes to procedures.

Associated EREF License Application Revisions:

SAR Section 11.4.3 will be revised to state: "Temporary changes to procedures are documented, reviewed and approved with the process described in Section 11.4.4, Changes to Procedures. In addition, the approved duration (e.g., expiration date) for the temporary change will be identified on the temporary procedure change in accordance with 10 CFR 70.72(a)(5)."

Commitments:

The change to the SAR identified in Enclosure 2.2 will be incorporated into Revision 2 of the License Application.

Attachments:

Enclosure 2.2: Markup of SAR Section 11.4.3.

NRC RAI Number: MM-8

Text of NRC RAI:

Chapter 11.5, "Audits and Assessments" states that the audit and assessment program applies to quality assurance and that the Quality Assurance Department shall be responsible for audits. The SAR further states, on page 11.5-46 that personnel performing audits and assessments do not report to the production organization and have no direct responsibility for the function or area being assessed. Please clarify whether the personnel performing audits of quality assurance report to the Quality Assurance Department.

AES Response to NRC RAI:

Audits of areas of the QA Program for which the EREF QA Organization has direct responsibility for implementation will be performed by an independent third party Certified Lead Auditor that is either contracted or obtained from one of our affiliate companies to an appropriate level of independence from the EREF QA Organization.

In addition to these independent audits, the EREF QA Organization will perform self-assessments of their areas of responsibility.

Associated EREF License Application Revisions:

This response does not require a revision to the EREF license application.

Commitments:

None

Attachments:

None

NRC RAI Number: MM-9

Text of NRC RAI:

Chapter 11.5.4, "Qualifications and Responsibilities for Audits and Assessments" describes the requirements for certification of lead auditors. Please describe the requirements for maintenance of proficiency and recertification.

10 CFR 70.62(d) requires that management measures, which include quality assurance elements such as audits and assessments, shall ensure that IROFS are designed, implemented and maintained such that they are available and reliable to perform their function when needed.

AES Response to NRC RAI:

SAR Section 11.5.4 states: "Certification of auditors and lead auditors is 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. A lead auditor must also have participated 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 audit must be a nuclear-related QA audit or audit equivalent within the year prior to certification."

The above requirement has been directly incorporated into our procedure entitled Quality Assurance Lead Auditor Training and Certification. In addition, the procedure contains requirements regarding maintenance of proficiency and re-certification for Lead Auditors. The Lead Auditors are required to maintain their proficiency through one or more of the following: 1) Regular and active participation in the audit process; 2) Review and study of codes, standards, procedures, instructions, and other documents related to QA programs and program auditing; and 3) Participation in training programs. The QA Manager will review the qualifications of each Lead Auditor on an annual basis to determine if retraining or requalification is required. If a Lead Auditor fails to maintain their proficiency for a period of two years or more, the Lead Auditor shall require requalification.

Associated EREF License Application Revisions:

This response does not require a revision to the EREF license application.

Commitments:

None

Attachments:

None

NRC RAI Number: MM-10

Text of NRC RAI:

Chapter [11.7] "Records" describes requirements for when a single records storage facility is used (i.e., the facility should be reviewed for adequacy by someone competent in fire protection and fire extinguishing). The QAPD, item 17.9 states that the requirements of ASME NQA-1-1994, Supplement 17S-1, Section 4.4 will be applied. Please clarify the SAR to make it consistent with the QAPD.

AES Response to NRC RAI:

ASME NQA-1 1994, Supplement 17S-1, Section 4.4.1, Single Storage Facility states: "The construction details shall be reviewed for adequacy of protection of contents by a person who is competent in the technical field of fire protection and fire extinguishing." The statement in SAR Section 11.7 is consistent with this requirement. In addition, SAR Section 11.7 states: "Appendix A, Section 17, Quality Assurance Records, of this chapter provides additional details regarding records management requirements." Thus, the SAR is consistent with the QAPD, including the commitment for hard copy or microfilm storage facilities to meet the requirements of ASME NQA-1-1994, Supplement 17S-1, Section 4.4.

Associated EREF License Application Revisions:

This response does not require a revision to the EREF License Application.

Commitments:

None

Attachments:

None

NRC RAI Number: QA-1

Text of NRC RAI:

Item 6.3, Document Control, states that temporary procedure changes that do not change the intent of procedures can be made at the work location by responsible management. This is inconsistent with the statement in Chapter 11.4.3 of the SAR that says that temporary changes to procedures must be approved by two members of management, at least one of whom is a Production Manager. Please clarify the QAPD to be consistent with the SAR. Additionally, clarify that procedure changes, including temporary changes, are subject to the requirements of 10 CFR 70.72.

AES Response to NRC RAI:

Item 6.3 of the QAPD will be revised to be consistent with the SAR Section 11.4.3 requirement that temporary procedure changes be approved by two members of management, at least one of whom is a Production Manager.

Procedure changes, including temporary changes, are subject to the requirements of 10 CFR 70.72. Besides from being a regulatory requirement, this requirement is addressed via the multiple references to 10 CFR 70.72 in the SAR.

- Section 3.0.3 of the SAR states: "The configuration management program is required by 10 CFR 70.72 (CFR, 2008f) and establishes a system to evaluate, implement, and track each change to the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel."

- Section 11.1.4.3 of the SAR states: "During the operations phase, changes to design will be documented, reviewed, and approved prior to implementation. AES will implement a change process that fully implements the provisions of 10 CFR 70.72 (CFR, 2008b)."

- Section 11.1.4.3 of the SAR states: "Each change to the facility or to activities of personnel shall have an evaluation performed in accordance with the requirements of 10 CFR 70.72 (CFR, 2008b), as applicable."

- SAR Section 11.4.3 states: "Temporary changes to procedures are documented, reviewed and approved with the process described in Section 11.4.4, Changes to Procedures, within 14 days of implementation." This statement will be revised to eliminate "within 14 days of implementation," because the requirements of 10 CFR 70.72 must be met prior to implementation.

- SAR Section 11.4.4 states that an evaluation shall be performed in accordance with 10 CFR 70.72 for changes to procedures.

Associated EREF License Application Revisions:

The SAR will be revised to incorporate the following changes:

SAR Section 11.4.3 will be revised to state: "Temporary changes to procedures are documented, reviewed and approved with the process described in Section 11.4.4, Changes to Procedures."

QAPD Section 6.3 will be revised to state that temporary procedure changes must be approved by two members of management, at least one of whom is a Production Manager.

Commitments:

The changes to the SAR identified in Enclosure 2.2 will be incorporated into Revision 2 of the License Application.

Attachments:

Enclosure 2.2: Markups of SAR Section 11.4.3 and QAPD Section 6.3

NRC RAI Number: QA-2

Text of NRC RAI:

Item 7.4.6, Control of Purchased Items and Services, states that acceptance of items includes one or more of the following methods: certificate of conformance, source verification, receiving inspection, post-installation test, and performance history. It further states that for QA level 1 IROFS, if performance history is used, at least one of the other acceptance methods must also be used. Please justify why performance history alone is an adequate method for acceptance of QA level 2 IROFS.

10 CFR 70.62(d) requires that management measures, which include other quality assurance elements such as procurement, shall ensure that IROFS are available and reliable to perform their function when needed.

AES Response to NRC RAI:

QAPD Section 7.4.6 incorrectly refers to supplier qualification and performance history as a method for acceptance of items, including spare and replacement parts. QAPD Section 7.4.6 will be revised to: 1) Eliminate the discussion regarding supplier qualification and performance history; 2) Require for QA Level 1 items, a Certificate of Conformance plus one or more of the other methods is used to establish acceptance of items; and 3) Require for QA Level 2 items, any one or more of the methods established above is used to establish acceptance of items.

In addition, QAPD Section 7.4.1 defines the methods for performance of pre-award evaluation of suppliers to provide items or services in accordance with the requirements of procurement documents. One of the methods is an evaluation of the potential supplier's history of providing an identical or similar product that performs satisfactorily in actual use. This method is revised to identify that at least one other method of supplier evaluation is used, in addition to performance history, to evaluate suppliers of QA Level 1 items.

Associated EREF License Application Revisions:

QAPD Sections 7.4.1 and 7.4.6 will be revised as identified in Enclosure 2.2.

Commitments:

The revision to QAPD Sections 7.4.1 and 7.4.6 contained in Enclosure 2.2 will be incorporated into Revision 2 of the License Application.

Attachments:

Enclosure 2.2: Markup of QAPD Sections 7.4.1 and 7.4.6

NRC RAI Number: QA-3

Text of NRC RAI:

Item 7.6, "Approved Suppliers List," describes the use of an approved suppliers list for vendors for which AES has determined have an acceptable QA program. Please clarify whether the approved suppliers list will state the scope of items and services for which the supplier is approved.

10 CFR 70.62(d) requires that management measures, which include other quality assurance elements such as procurement, shall ensure that IROFS are available and reliable to perform their function when needed.

AES Response to NRC RAI:

AES issued a procedure entitled Quality Assurance Evaluation and Selection of Suppliers that defines the requirements for the content of the Approved Supplier's List (ASL). Each ASL listing is required to define the product or service scope of supply that the supplier is approved to provide, and define any approval restrictions and conditions.

Associated EREF License Application Revisions:

This response does not require a revision to the EREF License Application.

Commitments:

None

Attachments:

None

Markups of Safety Analysis Report

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."

1.2.2 Financial Information

AES estimates the total cost of the EREF to be approximately $4.1 billion (in 2007 dollars), excluding escalation, contingency, interest, tails disposition, decommissioning, and any replacement equipment required during the life of the facility. Insert A

~~There are financial qualifications to be met before a license can be issued. AES acknowledges the use of the following Commission-approved criteria as described in Policy Issues Associated with the Licensing of a Uranium Facility; Issue 3, Financial Qualifications (LES, 2002) in determining if the project is financially feasible:~~

~~Construction of the facility shall not commence before funding is fully committed. Of this full funding (equity and debt), the applicant must have in place before constructing the associated capacity: (a) a minimum of equity contributions of 30% of project costs from the parents and (b) firm commitments ensuring funds for the remaining project costs.~~

~~AES shall not proceed with the project unless it has in place long-term enrichment contracts (i.e., five years) with prices sufficient to cover both construction and operation costs, including a return on investment, for the entire term of the contracts.~~

AES shall in accordance with 10 CFR 140.13b, (CFR, 2008b), prior to and throughout operation, have and maintain nuclear liability insurance in the amount of up to $300 million to cover liability claims arising out of any occurrence within the United States, causing, within or outside the United States, bodily injury, sickness, disease, or death, or loss of or damage to property, or loss of use of property, arising out of or resulting from the radioactive, toxic, explosive, or other hazardous properties of chemical compounds containing source or special nuclear material.

The amounts of nuclear energy liability insurance required may be furnished and maintained in the form of:

An effective facility form (non-indemnified facility) policy of nuclear energy liability insurance from American Nuclear Insurers and/or Mutual Atomic Energy Liability underwriters; or

Such other type of nuclear energy liability insurance as the Commission may approve; or

A combination of the foregoing.

If the form of liability insurance will be other than an effective facility form (non-indemnified facility) policy of nuclear energy liability insurance from American Nuclear Insurers and/or Mutual Atomic Energy Liability Underwriters, such form will be provided to the Nuclear Regulatory Commission by AES. The effective date of this insurance will be no later than the date that AES takes possession of licensed nuclear material. 1a

Insert A

Pursuant to 10 CFR 70.23(a)(5), AES is required to demonstrate that it is financially qualified to carry out the activities proposed in its application. AES proposes to satisfy this obligation 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). That approach is as follows:

- 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.

- Operation of the EREF shall not commence until AES has in place either: (1) long term contracts lasting five years or more that provide sufficient funding for the estimated cost of operating the facility for the five year period; (2) documentation of the availability of one or more alternative sources of funds that provide sufficient funding for the estimated cost of operating the facility for five years; or (3) some combination of (1) and (2).

By letter dated December 22, 2008, American Nuclear Insurers documented its expectation to provide nuclear liability insurance for the EREF at the maximum policy limit of $300M by the time AES takes possession of source or special nuclear material. AES will provide proof of liability insurance of a type and in the amounts to cover liability claims required by 10 CFR 140.13b prior to taking possession of source or special nuclear material.

Information indicating how reasonable assurance will be provided that funds will be available to decommission the facility as required by 10 CFR 70.22(a)(9) (CFR, 2008c), 10 CFR 70.25 (CFR, 2008d), and 10 CFR 40.36 (CFR, 2008e) is described in detail in Chapter 10, Decommissioning.

1.2.3 Type, Quantity and Form of Licensed Material

Insert A - SAR Section 1.2.3

AES proposes to acquire, deliver, receive, possess, produce, use, transfer, and/or store special nuclear material (SNM) meeting the criteria of special nuclear material of low strategic significance as described in 10 CFR 70.4 (CFR, 2008f). Details of the SNM are provided in Table 1.2-1, Type, Quantity, and Form of Licensed Material. ~~It is expected that other source materials and by-product materials will also be used for instrument calibration purposes. These materials will be identified during the design phase and the SAR will be revised, accordingly.~~

1.2.4 Requested Licenses and Authorized Uses

AES is engaged in providing uranium enrichment services to electric utilities for the purpose of manufacturing fuel to be used to produce electricity in commercial nuclear power plants.

This application is for the necessary licenses issued under 10 CFR 70 (CFR, 2008g), 10 CFR 30 (CFR, 2008h) and 10 CFR 40 (CFR, 2008i) to construct, own, use and operate the facilities described herein as an integral part of the uranium enrichment facility. This includes licenses for source, special nuclear material, and byproduct material. The period of time for which the license is requested is 30 years.

Section 1.1, Facility and Process Description, provides a summary description of the enrichment activities that will occur at the EREF.

1.2.5 Special Exemptions of Special Authorizations

In accordance with 10 CFR 40.14 (CFR, 2008j), "Specific exemptions," and 10 CFR 70.17 (CFR, 2008k), "Specific exemptions," AES requests exemptions from certain provisions of 10 CFR 40.36 (CFR, 2008e), "Financial assurance and recordkeeping for decommissioning," paragraph (d), and 10 CFR 70.25 (CFR, 2008d), "Financial assurance and recordkeeping for decommissioning," paragraph (e). Specifically, 10 CFR 40.36(d) (CFR, 2008e) and 10 CFR 70.25(e) (CFR, 2008d) both state in part that "...the decommissioning funding plan must also contain a certification by the licensee that financial assurance for decommissioning has been provided in the amount of the cost estimate for decommissioning...." As stated in Section 10.2.1, "Decommissioning Funding Mechanism," of the SAR since AES intends to sequentially install and operate modules of the enrichment equipment over time, providing financial assurance for decommissioning during the operating life of the EREF at a rate that is in proportion to the decommissioning liability for these facilities as they are phased in satisfies the requirements of this regulation without imposing the financial burden of maintaining the entire financial coverage for facilities and material that are not yet in existence. The same basis applies to decommissioning funding assurance for depleted uranium tails. As also stated in Section 10.2.1 of the SAR, AES proposes to provide financial assurance for the disposition of

1a

Insert A to SAR Section 1.2.3

"Other source materials and by-product materials will also be used for instrument calibration purposes. These 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. Subsequently, the SAR will be revised to incorporate the additional sources."

The transport of ash to the EREF from future Cascade tephra eruptions was considered. The maximum ash thickness that could be deposited at the EREF is less than 8 cm. The load of ash (8 cm) would range from 8.2 lb/ft^2 (40 kg/m^2) (dry) to 20.5 lb/ft^2 (100 kg/m^2) (wet), which is bounded by both the extreme environmental snow load and normal ground snow load for the EREF.

1.3.3.2 Annual Precipitation – Amounts and Forms

Air masses approaching the EREF location must cross over significant mountain ranges prior to their arrival in southeastern Idaho. In doing so, the majority of the moisture contained in these air masses condenses and precipitates over the mountains. As the air masses descend from the mountains, they warm adiabatically and become relatively dry. As a result, annual precipitation in the vicinity of the EREF is quite light. The data indicate that precipitation occurs infrequently (less than 3% of the time) and that precipitation intensity is predominately less than 0.1 in (2.54 millimeters).

The type of precipitation at the EREF location varies with the seasons. Convective showers and thundershowers occur in the summer. Precipitation during the spring and fall can be characterized as showery or as a steadier rainfall. Winter precipitation is typically in the form of snow which can occur anytime from September through May.

Annual average precipitation at Idaho Falls 2 ESE is 360.93 mm (14.21 in). This precipitation falls fairly evenly throughout the year with the exception of the month of May, which exhibits a significant spike in precipitation. The highest recorded monthly precipitation total is 115.82 mm (4.56 in). There have been several months in the 30-year period of record when no precipitation has been recorded.

Annual average precipitation at Idaho Falls 46 W is considerable less than what occurs at Idaho Falls 2 ESE and measures 224.03 mm (8.82 in). The precipitation pattern of these two locations is somewhat similar in that precipitation falls fairly evenly throughout the year with the exception of a precipitation maximum in May. The highest recorded precipitation total at Idaho Falls 46 W is 117.86 mm (4.64 in).

Over the 30-year period of record, precipitation has always fallen at some time during the months of January, May, June, and August. Over the same period of record, there have been at least ten months when no precipitation has been recorded. The highest daily precipitation event recorded over the 48-year period of record is 41.66 mm (1.64 in).

The annual average snowfall for Idaho Falls 2 ESE is 833.12 mm (32.8 in). The highest daily snowfall at this location is 254 mm (10 in). The highest monthly snowfall is 571.5 mm (22.5 in). The highest daily snow depth is 660.4 mm (26 in).

The annual average snowfall for Idaho Falls 46 W is 637.54 mm (25.1 in). The highest daily snowfall at this location is 218.44 mm (8.6 in). The highest monthly snowfall is 566.42 mm (22.3 in) occurring in December 1971. The highest daily snow depth is 762 mm (30 in).

Additional details on rainfall and snowfall are provided in Section 3.6 of the Environmental Report.

The design basis snow load was developed by combining the "building code" snow load with the additional surcharge from an extreme winter precipitation event. This is consistent with the guidance provided by NRC in the Site Analysis Branch Position for Winter Precipitation Loads (NRC, 1975). The ground "building code" snow load for the EREF was determined to be 44.2 lb/ft^2 (216 kg/m^2). This ground snow load will be converted to a roof snow load in accordance with ASCE 7-05 (ASCE, 2006). The extreme winter precipitation event results in a load of 19 lb/ft^2 (93 kg/m^2). This value will be combined with the appropriate building code roof snow load for use as the design basis snow load.

INSERT SAR 1.3.3.2 (attached)

1.3.3.3 Severe Weather

<u>Tornadoes</u>

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NRC RAI Number: GI-2

INSERT SAR 1.3.3.2 (attached)

The ground snow load at 44.2 lb/ft^2 for the EREF was determined as follows:

- Snow depth data (collected from the National Weather Service) at locations within close proximity to the EREF site that had a long period of record with similar climate conditions were used to compute L-moment parameters, were applied to Generalized Extreme Value (GEV) frequency distributions and used to estimate the 50-year return values.

- Snow course data, snow depth and snow water equivalent (SWE) (collected from the Natural Resources Conservation Service) at two locations near the EREF site that had a long period of record with similar climate conditions were used to create a snow depth/density relationship between snow depth and SWE.

- The 50-year snow depth data was converted into snow loads and spatially distributed to obtain the estimate at the EREF site. The ground snow load was estimated from spatial interpolation of the station snow load values.

CFR, 2008r. Title 10, Code of Federal Regulations, Part 25, Access Authorization for Licensed Personnel, 2003.

CFR, 2008s. Title 10, Code of Federal Regulations, Part 95, Security Facility Approval and Safeguarding of National Security Information and Restricted Data, 2003.

Doherty, 1979. Preliminary geological interpretation and lithologic log of the exploratory geothermal test well (INEL-1), Idaho National Engineering Laboratory, eastern Snake River Plain, Idaho. United States Geologic Survey Open-File Report. 79-1248.

FEMA, 1981. Flood Map, Bonneville County, Idaho, Map ID: FM1600270025C, Website: http://msc.fema.gov, Date accessed: August 8, 2007.

Garabedian, 1992. Hydrogeology and Digital Simulation of the Regional Aquifer System, Eastern Snake River Plain, Idaho. U.S. Geol. Surv. Prof. Paper PP-1408-F, U.S. Government Printing Office, Washington, D.C., S.P. Garabedian, 1992.

IBC, 2006. International Building Code, International Code Council, Inc., 2006.

INL, 2008. Idaho National Laboratory Seismic Monitoring, Website: http://www.inl.gov/geosciences/earthquakes.shtml, Date accessed: November 17, 2008.

~~**LES, 2002.** Letter to U.S. Nuclear Regulatory Commission Document Control Desk from Peter L. Lenny, Louisiana Energy Services, April 24, 2002.~~

LES, 2005. Louisiana Energy Services (LES). "National Enrichment Facility Safety Analysis Report," Revision 6, 2005.

Lindholm, 1996. Summary of Snake River Plain Regional Aquifer-System Analysis in Idaho and eastern Oregon, U.S. Geol. Surv. Prof. Paper 1408-A., U.S. Government Printing Office, Washington, DC, G. F. Lindholm.

Marshall, 1973. J.L. Marshall, Lightning Protection, 1973

Nimmo, 2004. Hydraulic and Geochemical Framework of the Idaho National Engineering and Environmental Laboratory Vadose Zone. Vadose Zone Journal 3, J.R. Nimmo, J.P. Rousseau, K.S. Perkins, K.G. Stollenwerk, P.D. Glynn, R.C. Bartholomay, and L.L. Knobel, 2004.

NOAA, 2008a. National Oceanic and Atmospheric Administration, Query Results of the Storm Event Database for Bingham, Bonneville, Butte and Jefferson Counties, ID, National Climatic Data Center, Asheville, NC, July 2008.

NOAA, 2008b. Colorado Lightning Resource Center, National Oceanic and Atmospheric Administration, National Weather Service, 2008.

NRC, 1975. Site Analysis Branch Position for Winter Precipitation Loads, U.S. Nuclear Regulatory Commission, March 24, 1975.

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

NRC, 2003a. Louisiana Energy Services Policy Issues, Letter, R. Pierson, USNRC, to R. Krich, LES, dated March 24, 2003.

NRC, 2003b. Environmental Review Guidance for Licensing Actions Associated with NMSS Programs, NUREG-1748, August 2003.

NRC, 2005. Safety Evaluation Report for the National Enrichment Facility in Lea County, New Mexico, NUREG-1827, U.S. Nuclear Regulatory Commission, June 2005.

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, the Physical Security Plan (PSP), and the Standard Practices Procedure Plan for the Protection of Classified Matter (SPPP).

, and development and implementation of security programs for nuclear material control and accountability, physical protection of the facility, and protection of classified matter.

D. Environmental, Health, Safety and Licensing Manager

The Environmental, Health, Safety and Licensing (EHS&L) Manager reports to the Plant Manager and has the overall responsibility for the development and implementation of programs addressing worker health and safety; environmental protection; and licensing and permitting. The EHS&L Manager is also responsible for maintaining compliance with safeguards; appropriate rules, regulations, and codes; and implementation and control of the Fundamental Nuclear Material Control Plan (FNMCP)~~.~~ This includes EHS&L activities associated with nuclear criticality safety, radiation protection, chemical safety, environmental protection, *emergency preparedness,* ~~and~~ *industrial safety*~~.~~ The EHS&L Manager works with the other facility managers to ensure consistent interpretations of EHS&L requirements, performs independent reviews, and supports facility and operations change control reviews.

This position is independent from other operations management positions at the facility to ensure objective EHS&L audit, review, and control activities. The EHS&L Manager has the authority to order the shut down of operations if they appear to be unsafe or non-compliant with applicable regulatory requirements and must consult with the Plant Manager with respect to restart of shutdown operations after the deficiency, or unsatisfactory condition, has been resolved.

E. Project Manager

The Project Manager reports to the Plant Manager and has overall responsibility for managing the engineering, procurement, construction, and startup of facility modifications and expansion. This includes managing the work and contracts with the Technology Supplier (i.e., ETC).

F. Human Resources Manager

The Human Resource Manager reports to the AES President and has the responsibility for community relations, ensuring adequate staffing, and providing administrative support services to the facility including document control.

G. Operations Manager

The Operations Manager reports to the Plant Manager and has the responsibility of directing the day-to-day operation of the facility. Inherent in this responsibility is the assurance that the operations are conducted safely and in compliance with license conditions. This includes such activities as ensuring the correct and safe operation of UF_6 processes, proper handling of UF_6, and the identification and mitigation of any off normal operating conditions. The Operations Manager is also responsible for the plant maintenance function, which includes activities to assure that Items Relied on for Safety (IROFS) are reliable and available when needed. In the event of the absence of the Plant Manager, the Operations Manager may assume the responsibilities and authorities of the Plant Manager.

H. Uranium Management Manager

The Uranium Management Manager reports to the Plant Manager and has the responsibility for UF_6 cylinder management (including compliance with transportation requirements) and directing the scheduling of enrichment operations to ensure smooth production. This includes activities such as *ensuring proper feed material and maintenance equipment are available for the facility.* In the event of the absence of the Plant Manager, the Uranium Management Manager may assume the responsibilities and authorities of the Plant Manager.

I. Training Manager

If at least three of the above criteria cannot be met, then the FPIN assigned to the IROFS and the associated accident sequence(s) will be re-evaluated and revised, as necessary, consistent with the overall ISA methodology.

- Upon completion of the design of IROFS, the IROFS boundaries will be defined. In defining the boundaries for each IROFS, ISA Summary Appendix A, Guidelines for Development of Boundary Definitions for IROFS and Attributes of Safe-by-Design Components, will be used. These guidelines require the identification of each support system and component necessary to ensure the IROFS is capable of performing its specified safety function.
- IROFS will be designed, constructed, tested and maintained to QA Levels in accordance with the QAPD. IROFS will comply with design requirements established by the ISA and the applicable codes and standards (current approved version at the time of design). IROFS components and their designs will be of proven technology for their intended application. These IROFS components and systems will be qualified to perform their required safety functions under normal and accident conditions, e.g., pressure, temperature, humidity, seismic motion, electromagnetic interference, and radio-frequency interference, as required by the ISA. IROFS components and systems will be qualified using the applicable guidance in Institute of Electrical and Electronics Engineers (IEEE) standard IEEE-323, 1983, "IEEE Standard for Qualifying Class 1 E Equipment for Nuclear Power Generating Stations" (IEEE, a 1983). Furthermore, IROFS components and systems will be designed, procured, installed, tested, and maintained using the applicable guidance in Regulatory Guide 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," Revision 1, dated October 2003 (NRC, 2003c). IROFS systems will be designed and maintained consistent with the reliability assumptions in the ISA. Redundant IROFS systems will be separate and independent from each other. IROFS systems will be designed to be fail-safe. In addition, IROFS systems will be designed such that process control system failures will not affect the ability of the IROFS systems to perform their required safety functions. ~~Plant control systems will not be used to perform IROFS functions~~. Installation of IROFS systems will be in accordance with engineering specifications and manufacturer's recommendations. Required testing and calibration of IROFS will be consistent with the assumptions of the ISA and setpoint calculations, as applicable. For hardware IROFS involving instrumentation which provides automatic prevention or mitigation of events, setpoint calculations are performed in accordance with a setpoint methodology, which is consistent with the applicable guidance provided in Regulatory Guide 1.105, "Setpoints for Safety-Related Instrumentation," Revision 3, dated December 1999 (NRC, 1999).
- For IROFS that use software, firmware, microcode, programmable logic controllers, and/or any digital device, including hardware devices which implement data communication protocols (such as fieldbus devices and Local Area Network controllers), etc., design will adhere to accepted best practices in software and hardware engineering, including software quality assurance controls as discussed in the QAPD throughout the development process and the applicable guidance of the following industry standards and regulatory guides:

  a. American Society of Mechanical Engineers (ASME) NQA-1-1994, Part II, subpart Part 2.7, "Quality Assurance Requirements of Computer Software for Nuclear Facility Applications," as revised by NQA-1a-1995 Addenda of NQA-1-1994 and ASME NQA-1-1994, Part 1, Supplement 11S-2, "Supplementary Requirements of Computer Program Testing." (ASME, 1994a) (ASME, 1995) (ASME, 1994b)

b. Electric Power and Research Institute (EPRI) NP-5652, "Guideline for the Utilization of Commercial Grade Items in Nuclear Safety Grade Applications," June 1988 (EPRI, 1988).

c. EPRI Topical Report (TR) -102323, "Guidelines for Electromagnetic Interference Testing in Power Plants," Revision 1, December 1996 (EPRI, 1996a).

d. EPRI TR-106439, "Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications," October 1996 (EPRI, 1996b).

e. Regulatory Guide 1.152, "Criteria for Digital Computers in Safety Systems in Nuclear Power Plants," Revision ~~1, January 1996 (NRC, 1996).~~ 2, January 2006 (NRC, 2006)

f. Regulatory Guide 1.168, Revision 1, "Verification, Validation, Reviews, and Audits for Digital Software Used in Safety Systems of Nuclear Power Plants," October, 2004 (NRC, 2004b).

g. Regulatory Guide 1.169, "Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants," September 1997 (NRC, 1997a).

h. Regulatory Guide 1.170, "Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants," September 1997 (NRC, 1997b).

i. Regulatory Guide 1.172, "Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants," September 1997 (NRC, 1997c).

j. Regulatory Guide 1.173, "Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems for Nuclear Power Plants," September 1997 (NRC, 1997d).

- For those IROFS requiring operator actions, a human factors engineering review of the human-system interfaces shall be conducted using the applicable guidance in NUREG-0700, "Human-System Interface Design Review Guidelines," Revision 2, dated May 2002 (NRC, 2002a), and NUREG-0711, "Human Factors Engineering Program Review Model," Revision 2, dated February 2004 (NRC, 2004a).
- For IROFS and IROFS with Enhanced Failure Probability Index Numbers (i.e., enhanced IROFS) that require "independent verification" of a safety function, the independent verification shall be independent with respect to personnel and personnel interface. Specifically, a second qualified individual, operating independently (e.g., not at the same time or not at the same location) of the individual assigned the responsibility to perform the required task, shall, as applicable, verify that the required task (i.e., safety function) has been performed correctly (e.g., verify a condition), or re-perform the task (i.e., safety function), and confirm acceptable results before additional action(s) can be taken which potentially negatively impact the safety function of the IROFS. The required task and independent verification shall be implemented by procedure and documented by initials or signatures of the individuals responsible for each task. In addition, the individuals performing the tasks shall be qualified to perform, for the particular system or process (as applicable) involved, the tasks required and shall possess operating knowledge of the particular system

- All buildings and structures, including such items as equipment supports, are designed to withstand the earthquake loads defined in Section 1613 of the IBC (ICC, 2006) which invokes the earthquake design requirements of ASCE 7-05 (ASCE, 2005a). Every structure is designed to resist the total lateral seismic forces applied in the directions which will produce the most critical load effects as delineated in Section 12.5 of ASCE 7-05 (ASCE, 2005a). The seismic analysis shall consist of one of the types permitted by Table 12.6-1 in ASCE 7-05 (ASCE, 2005a), based on the structure's seismic design category, structural system, dynamic properties, and regularity. The permitted analytical procedures include Equivalent Lateral Force Analysis, Modal Response Spectrum Analysis, and Seismic Response History Procedures.

  The provisions in AISC 341-05 (AISC, 2005b) govern the design fabrication, and erection of structural steel members and connections in the seismic load resisting system (SLRS) and splices in columns that are not a part of the SLRS, in buildings and other structures, where the seismic response modification coefficient, R, (as specified in ASCE 7-05 (ASCE, 2005a)) is taken greater than 3, regardless of the seismic design category.

- The Design Basis Earthquake (DBE) for the EREF site will be determined using the methods in ASCE 43-05 (ASCE, 2005b). The peak accelerations will be determined during detail design. The design spectra will be based on the building construction type in accordance with Limit State C of ASCE 43-05 (ASCE, 2005b). For licensing purposes, soil amplification factors are based on Soil Class C. This assumption will be verified during final design.

- Normal Snow Loads (S) on roofs and other exposed surfaces for all structures including snow drifts, sliding snow, unbalanced snow, and rain on snow loads, are determined in accordance with the IBC (ICC, 2006), Section 1603 which invokes the snow load design requirements in Chapter 7 of ASCE 7-05 (ASCE, 2005a).

- Extreme Environmental Snow Loads on roofs of buildings listed above is based on a Ground Snow Load (p_g) of 309 kg/m^2 (63.2 lb/ft^2).

- The roof drainage systems (including secondary roof ~~drains~~ drainage path) will be designed such that the amount of rainfall that can collect on the roof does not exceed the ~~normal roof design live load~~ design load for the roof.

- Roofs will be designed so as to not pond water to a depth during the extreme local precipitation that could exceed the Extreme Environmental Rainfall which is equivalent to the 24-hour extreme local precipitation estimate of 112 mm (4.39 in).

- The following features apply to the SBM, TSB, CRSB and BSPB:

  - Since the sloped roof design precludes any significant ponding on the roofs, any leaks into the building through the roof liner would not be significant due to small hydrostatic driving heads of any water on the roof. The layouts in the SBMs, CRSB and BSPB are very open designs which would result in significant spreading out any precipitation leaking into the buildings. The layout in the TSB provides for smaller rooms spread over three floors. The individual rooms are interconnected through many doors. Any leaks into the building through the roof liner would disperse from room to room and floor to floor without any significant ponding in any of the individual rooms.

  - The facility floor levels will be set 0.15 m (6 in) above the finished outside adjacent grade. Finished grading will slope away from buildings preventing any accumulations/ponding of precipitation from roof run-off or sheet flow of storm water against the buildings. At roof access doors, the door threshold is set at least 0.15 m (6 in) above the top of the roofing material.

- Exhaust flow from the potentially contaminated rooms (i.e., Decontamination Workshop, Chemical Trap workshop, Mobile Unit Disassembly & Reassembly Workshop, Valve & Pump Dismantling Workshop and Maintenance Facility) of the TSB is filtered by a pre-filter, HEPA filter, activated carbon filter and HEPA filter and is then released through an exhaust vent. The exhaust flow is continuously monitored for alpha and HF. The exhaust air is periodically sampled. The continuous monitoring and periodic sampling is in accordance with the guidance in Regulatory Guide 4.16 (NRC, 1985).
- The Electrical System design complies with the following codes and standards: Insert
  - IEEE C2-2007, National Electrical Safety Code (IEEE, 2007) a
  - NFPA 70, National Electric Code (NFPA, 2008)
  - NFPA 70E, Standard for Electrical Safety Requirements for Employee workplaces (NFPA, 2004)
- On a loss of electrical power, the systems associated with items relied on for safety (IROFS) will be designed such that the safety function is maintained or the feature fails-safe.
- The potential for hydrogen accumulation and explosion will be evaluated as part of final design. The number of batteries, battery type, and charge rate information is required to determine hydrogen generation potential. Once this information is known, the ability of the room or area housing the batteries to develop an ignitable mixture of hydrogen will be evaluated and will identify appropriate features required to prevent or mitigate the effects of hydrogen ignition.
- The ventilation control of hydrogen gas will be provided in accordance with National Fire Protection Association (NFPA) 70E–2004, Standard for Electrical Safety in the Workplaces, (NFPA, 2004) and the Institute of Electrical and Electronics Engineers (IEEE) C2-2007, National Electrical Safety Code (IEEE, 2007). a
- Based on the current level of design, battery control systems have been identified for use by the 13.8 kV switchgear systems. The control system requirements for the 480/440 V switchgear have not been fully developed. This system will require further definition during detailed design to determine the control power scheme to be utilized.
- The Communication and Alarm Annunciation Systems Design complies with the following Codes and Standards:
  - NFPA 70 – 2008. National Electric Code (NFPA, 2008)
  - NFPA 72 – 2007. National Fire Alarm Code (NPFA, 2007)
  - 29 CFR Part 1910.7. Occupational Safety and Health Standards (CFR, 2008e)
  - IEEE C2 – 2007. National Electric Safety Code (IEEE, 2007) a
- The criticality safety for tanks that are not safe-by design will utilize two independent Items Relied on For Safety (IROFS) for mass control, the two are referred to as "sampled and analyzed," e.g., tank contents are sampled and analyzed before being transferred to another tank or out of the system. The "bookkeeping measures" is a process to calculate the potential mass of uranium in the tank for any batch operation to ensure that no tank holds more than a safe mass of uranium. This calculated mass of uranium is then compared to a mass limit, which is based on the double-batching limit on mass of uranium in a vessel from the criticality safety analyses. The "bookkeeping measures" process is described in further detail below.

Insert for SAR Section 3.3.7 Utility and Support Systems Requirements (page 3.3-14):

- IEEE 80-2007, Guide for Safety in AC Substation Grounding (IEEE, 2000)
- IEEE 81-1983, Guide for Measuring Earth Resistivity, Ground Impedance, and Earth Surface Potential of a Ground System (IEEE, 1983b)
- IEEE 142-2007, Grounding of Industrial and Commercial Power Systems (IEEE, 2007b)

Table 3.3-10 Codes and Standards
(Page 1 of 3)

ACI 318-05, Building Code Requirements for Structural Concrete, 2008.
ACI 349-06, Code Requirements for Nuclear Safety Related Concrete Structures, 2007.
AIChE, Guidelines for Hazard Evaluation Procedures, 2nd Edition, April, 1992.
AISC Manual of Steel Construction, Thirteenth Edition, 2005.
ANSI/AISC 360-05 – Specification for Structural Steel Buildings, 2005.
ANSI N14.1-2001, American National Standard for Nuclear Materials – Uranium Hexafluoride Packaging for Transport, 2001.
ASCE 43-05, Seismic Design Criteria for Structural Systems, and Components in Nuclear Facilities, 2005.
ASCE 7-05, Minimum Design Loads for Building and Other Structures, 2006.
ASME B31, Standards of Pressure Piping, revision in effect at the time of detailed design (The applicable provisions of ASME B31 will govern the material, design, fabrication, examination, testing and inspection for piping.)
ASME, Boiler and Pressure Vessel Code, Section VIII, Division 1, 2007.
ASTM C761-04 – Standard Test Methods for Chemical, Mass Spectrometric, Spectrochemical, Nuclear, and Radiochemical Analysis of Uranium Hexafluoride, 2004.
ASTM E84-08a, "Standard Test Method for Surface Burning Characteristics of Building Materials," 2008.
DOE, 2003. HDBK-1169-2003, Nuclear Air Cleaning Handbook, Department of Energy, 2003.
IEEE C2-2007, National Electrical Safety Code, 2007.
ISO 668: 1995, Series 1 Freight Containers – Classification, Dimension and Ratings, 1995.
NFPA 10, Portable Fire Extinguishers, 2007.
NFPA 101, Life Safety Code, 2006.
NFPA 13, Installation of Sprinkler Systems, 2007.

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Insert 1 for Table 3.3-10

IEEE C37.90, IEEE Standard for Relays and Relay Systems Associated with Electric Power Apparatus, 1989.

IEEE C 37.90.1, IEEE Standard for Surge Withstand Capability (SWC) Tests for Relays and Relay Systems Associated with Electric Power Apparatus, 2002.

IEEE 80, Guide for Safety in AC Substation Grounding, 2000.

IEEE 81, Guide for Measuring Earth Resistivity, Ground Impedance, and Earth Surface Potential of a Ground System, 1983.

IEEE 142, Grounding of Industrial and Commercial Power Systems.

IEEE 450, IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications, 2002.

IEEE 484, IEEE Recommended Practice for Installation Design and Installation of Vented Lead-Acid Batteries for Stationary Applications, 2002.

IEEE 485, IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications, 1997.

IEEE 519, IEEE Recommended Practices and Requirements for Harmonic Control in Electrical Power Systems, 1992.

IEEE 946, IEEE Recommended Practice for the Design of DC Auxiliary Power Systems for Generating Stations, 2004.

ASTM, 2004. Standard Specification for Uranium Hexafluoride Enriched to Less Than 5 % ^{235}U, ASTM C-996-04,American Society for Testing and Materials, 2004.

Bowles, 1996. Foundation Analysis and Design, J.A. Bowles, 1996.

CFR, 2008a. Title 10, Code of Federal Regulations, Section 70.65, Additional content of applications, 2008.

CFR, 2008b. Title 10, Code of Federal Regulations, Part 70, Domestic Licensing of Special Nuclear Material, 2008.

CFR, 2008c. Title 10, Code of Federal Regulations, Section 70.61, Performance requirements, 2008.

CFR, 2008d. Title 10, Code of Federal Regulations, Section 70.62, Safety Program and Integrated Safety Analysis, 2008.

CFR, 2008e. Title 29, Code of Federal Regulations, Section 1910, Occupational Safety and Health Standards, 2008.

CFR, 2008f. Title 10, Code of Federal Regulations, Section 70.72, Facility Changes and Change Process, 2008.

CFR, 2008g. Title 10, Code of Federal Regulations, Section 70.4, Definitions, 2008.

CFR, 2008h. Title 10, Code of Federal Regulations, Section 70.64, Requirements for new facilities or new processes at existing facilities, 2008.

ICC, 2006. International Building Code, Internal Code Council, Inc., 2006.

EGG, 1990. Generic Component Failure Data Base for Light Water and Liquid Sodium Reactor PRAs, EGG-SSRE-8875, EGG, February 1990.

EPRI, 1988. Electric Power Research Institute (EPRI) NP-5652, "Guideline for the Utilization of Commercial Grade Items in Nuclear Safety Grade Applications," June 1988.

EPRI, 1996a. EPRI Topical Report (TR) -102323, "Guidelines for Electromagnetic Interference Testing in Power Plants," Revision 1, December 1996.

EPRI, 1996b. EPRI TR-106439, "Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications," October 1996.

a **IEEE, 1983.** Standard for Qualifying Class 1 E Equipment for Nuclear Power Generating Stations, IEEE-323, Institute of Electrical and Electronics Engineers, 2003. Insert 1

a **IEEE, 2007.** National Electrical Safety Code, IEEE C2-2007, Institute of Electrical and Electronics Engineers, 2007. Insert 2

LES, 2005. Louisiana Energy Services (LES), National Enrichment Facility Safety Analysis Report, Revision 6, 2005.

NAVFAC, 1986a. Foundations and Earth Structures, NAVFAC DM-7.02, Naval Facilities Engineering Command Design Manual, 1986.

NAVFAC, 1986b. Soil Mechanics, NAVFAC DM-7.01, Naval Facilities Engineering Command Design Manual, 1986.

NFPA, 2004. Standard for Electrical Safety Requirements in the Workplace, NFPA 70E, National Fire Protection Association, 2004.

NFPA, 2007. National Fire Alarm Code, NFPA 72, National Fire Protection Association, 2007.

Insert 1 for SAR Page 3.4-2

IEEE,1983b. IEEE 81, Guide for Measuring Earth Resistivity, Ground Impedance, and Earth Surface Potential of a Ground System, Institute of Electrical and Electronics Engineers, 1983.

IEEE, 1989. IEEE C37.90, IEEE Standard for Relays and Relay Systems Associated with Electric Power Apparatus, Institute for Electrical and Electronic Engineers,1989.

IEEE, 1992. IEEE 519, Recommended Practices and Requirements for Harmonic Control in Electrical Power Systems, Institute for Electrical and Electronic Engineers, 1992.

IEEE, 1997. IEEE 485, Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications, Institute for Electrical and Electronic Engineers, 1997.

IEEE, 2000. IEEE 80, Guide for Safety in AC Substation Grounding, Institute of Electrical and Electronics Engineers, 2000.

IEEE, 2002. IEEE C 37.90.1, IEEE Standard for Surge Withstand Capability (SWC) Tests for Relays and Relay Systems Associated with Electric Power Apparatus, 2002.

IEEE, 2002a. IEEE 450, Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications, Institute for Electrical and Electronic Engineers, 2002.

IEEE, 2002b. IEEE 484, Recommended Practice for Installation Design and Installation of Vented Lead-Acid Batteries for Stationary Applications, Institute for Electrical and Electronic Engineers, 2002.

IEEE, 2004. IEEE 946, Recommended Practice for the Design of DC Auxiliary Power Systems for Generating Stations, Institute for Electrical and Electronic Engineers, 2004.

Insert 2 for SAR Page 3.4-2

IEEE, 2007b. IEEE 142, Grounding of Industrial and Commercial Power Systems, Institute of Electrical and Electronics Engineers, 2007.

NFPA, 2008. National Electrical Code, NFPA 70, National Fire Protection Association, 2008.

NRC, 1983. Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications, Final Report, NUREG/CR-1278, U.S. Nuclear Regulatory Commission, August 1983.

NRC, 1985. 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, Regulatory Guide 4.16, Revision 1, U.S. Nuclear Regulatory Commission, December 1985.

~~**NRC, 1996.** Regulatory Guide 1.152, "Criteria for Digital Computers in Safety Systems in Nuclear Power Plants," Revision 1, January 1996.~~

NRC, 1997a. Regulatory Guide 1.169, "Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants," September 1997.

NRC, 1997b. Regulatory Guide 1.170, "Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants," September 1997.

NRC, 1997c. Regulatory Guide 1.172, "Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants," September 1997.

NRC, 1997d. Regulatory Guide 1.173, "Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants," September 1997.

NRC, 1999. Setpoints for Safety-Related Instrumentation, Regulatory Guide 1.105, U.S. Nuclear Regulatory Commission, Revision 3, December 1999.

NRC, 2001a. Integrated Safety Analysis Guidance Document, NUREG-1513, U.S. Nuclear Regulatory Commission, May 2001.

NRC, 2001b. Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units for Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants, Regulatory Guide 1.140, Revision 2, U.S. Nuclear Regulatory Commission, June 2001.

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

NRC, 2003a. Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites, Regulatory Guide 1.198, U.S. Nuclear Regulatory Commission, November, 2003.

NRC, 2003b. Potentially Defective 1-Inch Valves for Uranium Hexafluoride Cylinders, NRC Bulletin 2003-03, U.S. Nuclear Regulatory Commission, August 2003.

NRC, 2003c. Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems, Regulatory Guide 1.180, U.S. Nuclear Regulatory Commission, Revision 1, October 2003.

NRC, 2004a. Human Factors Engineering Program Review Model, NUREG-0711, U.S. Nuclear Regulatory Commission, Revision 2, February 2004.

NRC, 2004b. Regulatory Guide 1.168, "Verification, Validation, Reviews, and Audits for Digital Software Used in Safety Systems of Nuclear Power Plants," Revision 1, February 2004.

NRC, 2005. Safety Evaluation Report for the National Enrichment Facility in Lea County, New Mexico, NUREG-1827, U.S. Nuclear Regulatory Commission, June 2005.

NRC, 2006. Regulatory Guide 1.152, "Criteria for Digital Computers in Safety Systems in Nuclear Power Plants," Revision 2, January 2006.

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The guidance of Regulatory Guide 4.21 will be followed to minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste (NRC, 2008).

4.2 COMMITMENT TO AN ALARA PROGRAM

Section 4.1, Commitment to Radiation Protection Program Implementation, above, states the facility's commitment to the implementation of an ALARA program. The objective of the program is to make every reasonable effort to maintain facility exposures to radiation as far below the dose limits of 10 CFR 20.1201 (CFR, 2008f) as is practical and to maintain radiation exposures to members of the public such that they are not expected to exceed the dose constraints of 10 CFR 20.1101(d) (CFR, 2008d). The design and implementation of the ALARA program is consistent with the guidance provided in Regulatory Guides 8.2 (NRC, 1973a), 8.13 (NRC, 1999a), 8.29 (NRC, 1996), and 8.37 (NRC, 1993g). The operation of the facility is consistent with the guidance provided in Regulatory Guide 8.10 (NRC, 1977).

Annual doses to individual personnel are maintained ALARA. In addition, the annual collective dose to personnel (i.e., the sum of all annual individual doses, expressed in person-Sv or person-rem) is maintained ALARA. The dose equivalent to the embryo/fetus is maintained below the limits of 10 CFR 20.1208 (CFR, 2008g).

The Radiation Protection Program is written and implemented to ensure that it is comprehensive and effective. The written program documents policies that are implemented to ensure the ALARA goal is met. Facility procedures are written so that they incorporate the ALARA philosophy into the routine operations of the facility and ensure that exposures are consistent with 10 CFR 20.1101 (CFR, 2008d) limits. As discussed in Section 4.7, Radiation Surveys and Monitoring Programs Commitments, radiological zones will be established within the facility. The establishment of these zones supports the ALARA commitment in that the zones minimize the spread of contamination and reduce unnecessary exposure of personnel to radiation.

Specific goals of the ALARA program include maintaining occupational exposures as well as environmental releases as far below regulatory limits as is reasonably achievable. The ALARA concept is also incorporated into the design of the facility. The size and number of areas with higher doses rates are minimized consistent with accessibility for performing necessary services in the areas. Areas where facility personnel spend significant amounts of time are designed to maintain the lowest dose rates reasonably achievable.

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. 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/ALARA programs. Copies of the report are submitted to the AES President, Plant Manager, Radiation Safety Committee, and the Safety Review Committee.

The subject matter discussed above is identical to the National Enrichment Facility SAR (LES, 2005) subject matter with the exception that some organizational titles have been changed. The differences between the EREF and NEF organizations reflect AREVA's experience in operating fuel cycle facilities. Although some titles and scope of responsibility have been changed, the functions to be performed remain the same. Refer to Chapter 2.0 for additional information regarding these differences.

The NRC staff previously reviewed the National Enrichment Facility SAR (LES, 2005) application relative to the general guidelines of the occupational radiation protection program and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility

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Retraining of personnel previously trained is performed for radiological, chemical, industrial, and criticality safety at least annually. The retraining program also includes procedure changes and updating and changes in required skills. Changes to training are implemented, when required, due to incidents potentially compromising safety or if changes are made to the facility or processes. Records of training are maintained in accordance with the EREF records management system. Training programs are established in accordance with Section 11.3, Training and Qualifications. The radiation protection sections of the training program are evaluated at least annually. The program content is reviewed to ensure it remains current and adequate to assure worker safety.

The specifics of the Radiation Protection Training are described in the following section.

4.5.1 Radiation Protection Training

Radiation protection training is highlighted to emphasize the high level of importance placed on the radiological safety of plant personnel and the public. In-depth radiation protection training is provided for the various types of job functions (e.g., production operator, radiation protection technician, contractor personnel) commensurate with the radiation safety responsibilities associated with each such position. Visitors to a Restricted Area are trained in the formal training program or are escorted by trained personnel while in the Restricted Area.

Personnel access procedures ensure the completion of formal nuclear safety training prior to permitting unescorted access into the Restricted Areas. Training sessions covering criticality safety, radiation protection and emergency procedures are conducted on a regular basis to accommodate new employees or those requiring retraining. Retraining is conducted when necessary to address changes in policies, procedures, requirements and the ISA.

Specific topics covered in the training program are listed in Chapter 11, Management Measures, Section 11.3.3.1.1. The training provided includes the requirements of 10 CFR 19 (CFR, 2008a).

Individuals attending these sessions must pass an initial examination covering the training contents to assure the understanding and effectiveness of the training. The effectiveness and adequacy of the training program curriculum and instructors are also evaluated by audits performed by operational area personnel responsible for criticality safety and radiation protection.

Since contractor employees may perform diverse tasks in the Restricted Areas or Controlled Areas of the facility, formal training for these employees is designed to address the type of work they perform. In addition to applicable radiation safety topics, training contents may include RWPs, special bioassay sampling, and special precautions for welding, cutting, and grinding. Instructors certified by the Radiation Protection/Chemistry Manager conduct the radiation protection training programs.

The Radiation Protection/Chemistry Manager is responsible for establishing and maintaining the radiation protection training for all personnel, including contractor personnel who may be working at the facility. Records are maintained by the Training Manager for each employee documenting the training date, scope of the training, identity of the trainer(s), any test results and other associated information.

Individuals requiring unescorted access to a Restricted Area receive annual retraining. Contents of the formal radiation protection training program are reviewed and updated as required at least ~~every two years~~ annually by the EHS&L Manager or Radiation Protection/Chemistry Manager to ensure that the programs are current and adequate.

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provided upstream of the filters with high level alarms to inform operators of UF_6 releases in the plant.

Normal operation of the facility will not result in a release of radioactive material that exceeds regulatory limits. Ventilation systems for areas that do not have the potential for contamination are not monitored for radioactivity because radioactive material is not handled or processed in these areas. No emergency ventilation systems are provided for operation when the normal ventilation systems are shut down.

Several measures are in place to ensure effective operation of the ventilation systems. Differential pressure across High Efficiency Particulate Air (HEPA) filters in potentially contaminated ventilation exhaust systems is monitored monthly or automatically monitored and alarmed. Operating procedures specify limits and setpoints on the differential pressure consistent with manufacturers' recommendations. Filters are changed if they fail to function properly or if the differential pressure exceeds the manufacturers' ratings.

Filter inspection, testing, maintenance and change out criteria are specified in written procedures approved by the Operations Manager, or a designated alternate. Change out frequency is based on considerations of filter loading, operating experience, differential pressure data and any UF_6 releases indicated by HF alarms.

Gloveboxes are designed to maintain a negative differential pressure of about 0.623 mbar (0.25 in H_2O). This differential pressure is maintained anytime that the glovebox is in use. If the differential pressure is lost, use of the glovebox is suspended until the required differential pressure is restored.

Air flow rates at exhausted enclosures and close-capture points, when in use, are adequate to preclude escape of airborne uranium and minimize the potential for intake by workers. Air flow rates are checked monthly when in use and after modification of any hood, exhausted enclosure, close-capture point equipment or ventilation system serving these barriers. The various programs that pertain to preventive and corrective maintenance are described in Chapter 11, Sections 11.2.2, Corrective Maintenance and 11.2.3, Preventive Maintenance respectively.

4.6.2 Respiratory Protection Program

The facility uses process and engineering controls to control the concentration of radioactive material in air. However, there may be instances when it is not practical to apply process or other engineering controls. 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 are implemented to maintain the total effective dose equivalent ALARA. In these cases, the ALARA goal is 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
- Other controls, as available and appropriate.

Ventilation and containment systems (serving contaminated and potentially contaminated areas of the facility) will be designed and sized appropriately to reduce airborne concentrations below the occupational, derived air concentration (DAC) values specified in 10 CFR 20, Appendix B, during normal operations.

If an ALARA analysis is performed to determine whether or not respirators should be used, safety factors other than radiological factors may be considered. The impact of respirator use on workers' industrial health and safety is factored into decisions to use respirators.

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4.8.1.1 Bioassay

Internal radiological exposures are evaluated at least annually, or more frequently if conditions warrant, as noted in Section 4.7.7, Evaluation of Doses. Based on the results of air sample monitoring data, bioassays are performed for all personnel who are likely to have had an intake of one milligram of uranium. This is 10% of the 10 mg in a week regulatory limit (10 CFR 20.1201(e) (CFR, 2008f)) for intake of Class D uranium. The bioassay program has a sensitivity of 5 micrograms per liter (5 μg/L) of uranium concentration, assuming that the sample is taken within ten days of the postulated intake and that at least 1.4 L of sample is available from a 24-hour sampling period. Until urinalysis results indicate less than 15 μg/L uranium concentration, workers are restricted from activities that could routinely or accidentally result in internal exposures to soluble uranium.

It might not be possible to achieve a sensitivity of 5 micrograms per liter; if for example, all reasonable attempts to obtain a 1.4 liter 24-hour sample within 10 days fail. In such a case, the sample is analyzed for uranium concentration (if measurable) and the worker's intake is estimated using other available data.

The subject matter discussed above is identical to the National Enrichment Facility SAR (LES, 2005) subject matter. The NRC staff previously reviewed the National Enrichment Facility SAR (LES, 2005) application relative to the general guidelines of the occupational radiation protection program and concluded that the descriptions, specifications or analyses provided an adequate basis for safety review of the facility operations and that the construction and operation of the facility would not pose an undue risk to public health and safety. The specific discussion is in NUREG-1827 (NRC, 2005a).

Continuous air

4.8.1.2 Air Monitoring and Sampling

Alarm setpoints on the continuous air monitors in the airborne radioactivity areas may be used to provide an indication that internal exposures may be approaching the action limit.

Airborne activity in work areas ~~is~~ regularly determined in accordance with written procedures. Continuous air sampling ~~in~~ airborne radioactivity areas may be performed to complement the bioassay program. Using the values specified in 10 CFR 20 Appendix B (CFR, 2008m), if a worker could have inhaled radionuclide concentrations that are likely to exceed 12 DAC-hours in one week (7 days), then bioassay is conducted within 72 hours after the suspected or known exposure. Follow-up bioassay measurements are conducted to determine the committed effective dose equivalent. Until urinalysis results indicate less than 15 micrograms per liter uranium concentration, workers are restricted from activities that could routinely or accidentally result in internal exposures to soluble uranium.

~~Active on-line~~ monitors for airborne alpha emitters are used to measure representative airborne concentrations of radionuclides that may be due to facility operation. On-line monitoring for gross alpha activity is performed assuming all the alpha activity is due to uranium. When airborne activity data is used for dose calculations, the assumption is that all the activity is due to 234U , class D material. The lower limit of detection is either 0.02 milligrams of uranium in the total sample or 3.7 nBq/ml (1E-13 μCi/ml) gross alpha concentration. An action level is established at 1 mg of total uranium likely to be inhaled by a worker in seven days.

When deemed necessary, portable air samplers may be used to collect a sample on filter paper for subsequent analysis in the laboratory.

~~Monitors~~ are permanently located in Restricted Areas. These permanent monitors are operated to collect continuous samples. When air sampling is conducted using continuous air sampling devices, the filters are changed and analyzed at the following frequencies:

INSERT 4.8.1.2-1 (attached)

INSERT 4.8.1.2-2 (attached)

- Weekly and following any indication of release that might lead to airborne concentrations of uranium that are likely to exceed (1) 10% of the values listed in 10 CFR 20.1003 (CFR,

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INSERT 4.8.1.2-1

Continuous air monitors provide indication of the airborne activity levels in the Restricted Areas of the facility. These monitors are designed to detect alpha emitters in the air, which would indicate the potential for uranium contamination. Continuous air monitors

INSERT 4.8.1.2-2

Continuous air monitors in locations classified as Airborne Radioactivity Areas are equipped with alarms. The alarm is activated when airborne radioactivity levels exceed predetermined limits. The limits are set with consideration being given to both toxicity and radioactivity. The volume of air sampled may have to be adjusted to ensure adequate sensitivity with minimum sampling time. The operating history of the facility, changes in technology, changes in room functions and design, and changes in regulations may necessitate adjustment of the monitors.

When deemed necessary, portable air samplers may be used to collect a sample on filter paper for subsequent analysis in the laboratory.

continuous samplers

2008n), or (2) the total uranium action level of one milligram of total uranium inhaled in one week.

- Each Shift, following changes in process equipment or process control, and following detection of any event (e.g., leakage, spillage or blockage of process equipment) that are likely to exceed (1) 10% of the values listed in 10 CFR 20.1003 (CFR, 2008n), or (2) the total uranium action level of one milligram inhaled by a worker in one week.

The representativeness of the workstation air samplers shall be checked annually and when significant process or equipment changes have been made. Facility procedures specify how representativeness is determined.

Plant areas surveyed as described in this section include as a minimum UF_6 processing areas, decontamination areas, waste processing areas and laboratories. Continuous air ~~monitors~~ ~~(e.g., stationary samplers or personnel lapel samplers)~~ may be substituted when appropriate, as when continuous monitoring may not be reasonably achieved.

Action levels are based on trending of data collected during facility operation. Investigations are performed if airborne activity:

a. Exceeds 10% of the values listed in 10 CFR 20.1003 (CFR, 2008n) for Airborne Radioactivity Areas

b. Shows a short-term increase of a factor of 10 over historical data from the previous 12 months.

Corrective actions include investigation of the adverse trend and an evaluation of the need for changes, consistent with the principles of ALARA.

4.8.2 External Exposures

As noted previously, the potential for significant external exposure to personnel under routine operating conditions is less significant than that for internal exposures. This is primarily due to the nature of the radionuclides present in the facility.

Parameters important in determining dose from external exposures are:

- The length of time the worker remains in the radiation field
- The intensity of the radiation field
- The portion of the body receiving the dose.

Historical data from European facilities of similar construction show relatively low doses compared to nuclear power plant doses.

4.8.3 Procedures

Procedures are provided in the following areas to administratively control personnel radiation exposure:

- Operation
- Design
- Maintenance
- Modification

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4.11 ADDITIONAL PROGRAM COMMITMENTS

The following section describes additional program commitments related to the Radiation Protection Program.

4.11.1 Leak Testing Byproduct Material Sources

In addition to the uranium processed at the facility, other sources of radioactivity are used. These sources are small calibration sources used for instrument calibration and response checking. These byproduct material sources may be in solid, liquid, or gaseous form; the sources may be sealed or unsealed. Both types of sources present a small radiation exposure risk to facility workers. Typical byproduct material quantities and uses for a uranium enrichment centrifuge plant are summarized in Table 4.11-1, Typical Quantities of Byproduct Material for a Uranium Enrichment Centrifuge Plant. The byproduct materials for the EREF will be identified during the design phase and the Safety Analysis Report will be revised accordingly. Leak-testing of sources is performed in accordance with the following NRC Branch Technical Positions (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).

4.11.2 Records and Reports

- 10 CFR 30.50 (Reporting Requirements) (CFR, 2009a)
- 10 CFR 40.60 (Reporting Requirements) (CFR, 2009b)

The facility meets the following regulations for the additional program commitments applicable to records and reports:

- 10 CFR 20 Subpart L-Records (CFR, 2008w), Subpart M-Reports (CFR, 2008v)
- Section 70.61 (Performance requirements) (CFR, 2008e)
- Section 70.74 (Additional reporting requirements) (CFR, 2008s).

The facility Records Management program is described in Section 11.7, Records Management. The facility maintains complete records of the Radiation Protection Program for at least the life of the facility.

The facility maintains records of the radiation protection 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 corrective action program referrals, RWPs and planned special exposures.

By procedure, the facility will report to the NRC, within the time specified in 10 CFR 20.2202 (CFR, 2008t) and 10 CFR 70.74 (CFR, 2008s), any event that results in an occupational exposure to radiation exceeding the dose limits in 10 CFR 20 (CFR, 2008b). The facility will prepare and submit to the NRC an annual report of the results of individual monitoring, as required by 10 CFR 20.2206(b) (CFR, 2008u).

As previously noted in this chapter, the EREF will refer to the facility's corrective action program any radiation incident that results in an occupational exposure that exceeds the dose limits in 10 CFR 20 (CFR, 2008f), Appendix B (CFR, 2008m), or is required to be reported per 10 CFR

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CFR, 2008h. Title 10, Code of Federal Regulations, Section 70.22, Contents of applications, 2008.

CFR, 2008i. Title 10, Code of Federal Regulations, Section 19.12, Instructions to workers, 2008.

CFR, 2008j. Title 10, Code of Federal Regulations, Section 20.2110, Form of records, 2008.

CFR, 2008k. Title 10, Code of Federal Regulations, Section 19.13, Notifications and reports to individuals, 2008.

CFR, 2008l. Title 29, Code of Federal Regulations, Part 1910, Occupational Safety and Health Standards, 2008.

CFR, 2008m. Title 10, Code of Federal Regulations, Part 20, Appendix B, Annual Limits on Intakes (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage, 2008.

CFR, 2008n. Title 10, Code of Federal Regulations, Section 20.1003, Definitions, 2008.

CFR, 2008o. Title 10, Code of Federal Regulations, Section 20.1301, Dose limits for individual members of the public, 2008.

CFR, 2008p. Title 40, Code of Federal Regulations, Part 190, Environmental Radiation Protection Standard For Nuclear Power Operations, 2008.

CFR, 2008q. Title 10, Code of Federal Regulation, Section 20.1902, Posting requirements, 2008.

CFR, 2008r. Title 10, Code of Federal Regulations, Section 20.1202, Compliance with requirements for summation of external and internal does, 2008.

CFR, 2008s. Title 10, Code of Federal Regulations, Section 70.74, Additional reporting requirements, 2008.

CFR, 2008t. Title 10, Code of Federal Regulations, Section 20.2202, Notification of incidents, 2008.

CFR, 2008u. Title 10, Code of Federal Regulations, Section 20.2206, Reports of individual monitoring, 2008.

CFR, 2008v. Title 10, Code of Federal Regulations, Part 20, Subpart M-Reports, 2008.

CFR, 2008w. Title 10, Code of Federal Regulations, Part 20, Subpart L-Records, 2008.

CFR, 2008x. Title 10, Code of Federal Regulations, Part 30, Rules of General Applicability to Domestic Licensing of Byproduct Material, 2008.

CFR, 2008y. Title 10, Code of Federal Regulations, Part 40, Domestic Licensing Of Source Material, 2008.

CGA, 2004a. CGA G-7.1, Commodity Specification for Air, Compressed Gas Association, 2004

CGA, 2004b. CGA G-7, Compressed Air for Human Respiration, Compressed Gas Association, 2004.

DOE, 2003. HDBK-1169-2003, Nuclear Air Cleaning Handbook, Department of Energy, 2003.

LES, 2005. "National Enrichment Facility Safety Analysis Report," Rev. 7, Louisiana Energy Services, September 2006.

CFR, 2009a. Title 10, Code of Federal Regulation, Section 30.50, Reporting Requirements, 2009.
CFR, 2009b. Title 10, Code of Federal Regulation, Section 40.60, Reporting Requirements, 2009.

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NRC, 2005a. NUREG- 1827, "Safety Evaluation Report for the National Enrichment Facility in Lea County, New Mexico," Docket No. 70-3103, Louisiana Energy Services, U.S. NRC, June 2005.

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

Urenco, 2003. Health, Safety and Environmental Report, Urenco (Capenhurst) Limited, 2003.

Urenco, 2004. Health, Safety and Environmental Report, Urenco (Capenhurst) Limited, 2004.

Urenco, 2005. Health, Safety and Environmental Report, Urenco (Capenhurst) Limited, 2005.

Urenco, 2006. Health, Safety and Environmental Report, Urenco (Capenhurst) Limited, 2006.

Urenco, 2007. Health, Safety and Environmental Report, Urenco (Capenhurst) Limited, 2007.

NRC, 2008. Regulatory Guide 4.21, Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning, Rev. 0, U. S. Nuclear Regulatory Commission, June 2008.

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5.1 THE NUCLEAR CRITICALITY SAFETY (NCS) PROGRAM

The AREVA EREF, located in Bonneville County, Idaho, will be designed, constructed, and operated such that a nuclear criticality event is prevented, and to meet the regulatory requirements of 10 CFR 70 (CFR, 2008c). Nuclear criticality safety at the facility is assured by designing the facility, systems and components with safety margins such that safe conditions are maintained under normal and abnormal process conditions and any credible accident. Items Relied On For Safety (IROFS) identified to ensure subcriticality are discussed in the EREF Integrated Safety Analysis (ISA) Summary.

5.1.1 Management of the Nuclear Criticality Safety (NCS) Program

The NCS criteria in Section 5.2, Methodologies and Technical Practices, are used for managing criticality safety and include adherence to the double contingency principle as stated in the ANSI/ANS-8.1-1998, Nuclear Criticality Safety In Operations with Fissionable Materials Outside Reactors (ANSI, 1998a). The adopted double contingency principle states "process design should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible." Each process that has accident sequences that could result in an inadvertent nuclear criticality at the EREF meets the double contingency principle. To meet the double contingency principle, the EREF will incorporate into process designs sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

Using these NCS criteria, including the double contingency principle, low enriched uranium enrichment facilities have never had an accidental criticality. The plant will produce no greater than 5.0 $^{w}/_{o}$ enrichment. However, as additional conservatism, the nuclear criticality safety analyses are performed assuming a ^{235}U enrichment of 6.0 $^{w}/_{o}$, except for Contingency Dump System traps which are analyzed assuming a ^{235}U enrichment of 1.5 $^{w}/_{o}$, and include appropriate margins to safety. In accordance with 10 CFR 70.61(d) (CFR, 2008a), the general criticality safety philosophy is to prevent accidental uranium enrichment excesses, provide geometrical safety when practical, provide for moderation controls within the UF_6 processes and impose strict mass limits on containers of aqueous, solvent based, or acid solutions containing uranium. Insert Interaction controls provide for safe movement and storage of components. Plant and equipment features assure prevention of excessive enrichment. The plant is divided into six distinctly separate Assay Units (called Cascade Halls) with no common UF_6 piping. UF_6 blending is done in a physically separate portion of the plant. Process piping, individual centrifuges and chemical traps are safe by limits placed on their diameters. Product cylinders rely upon uranium enrichment, moderation control, and mass limits to protect against the possibility of a criticality event. Each of the liquid effluent collection tanks that hold uranium in solution is mass controlled, as none are geometrically safe. As required by 10 CFR 70.64(a) (CFR, 2008b), by observing the double contingency principle throughout the plant, a criticality accident is reduced. In addition to the double contingency principle, effective management of the NCS Program includes:

- An NCS program to meet the regulatory requirements of 10 CFR 70 (CFR, 2008c) will be developed, implemented, and maintained.
- *Safety parameters and procedures will be established.*
- The NCS program structure, including definition of the responsibilities and authorities of key program personnel will be provided.

NRC RAI Number: NCS-9

Insert

All NCS controls are preventative in nature; there are no mitigative NCS controls.

within a particular system. All assumptions relating to process, equipment, material function, and operation, including credible abnormal conditions, are justified, documented, and independently reviewed. Where possible, passive engineered controls are used to ensure NCS. The determination of the safe values of the major controlling parameters used to control criticality in the facility is described below.

Moderation control is in accordance with ANSI/ANS-8.22-1997, Nuclear Criticality Safety Based on Limiting and Controlling Moderators (ANSI, 1997). However, for the purposes of the NCSA, it is assumed that UF_6 comes in contact with water to produce aqueous solutions of UO_2F_2 as described in Section 5.2.1.3.3, Uranium Accumulation and Moderation Assumption. A uniform aqueous solution of UO_2F_2, and a fixed enrichment are conservatively modeled using MONK8A (SA, 2001) and the JEF2.2 library. Criticality analyses were performed to determine the maximum value of a parameter to yield $k_{eff} = 1$. The criticality analyses were then repeated to determine the maximum value of the parameter to yield a $k_{eff} = 0.95$. Table 5.1-1, Safe Values for Uniform Aqueous Solution of Enriched UO_2F_2, shows both the critical and safe limits for 5.0 $^{w}/_{o}$ and 6.0 $^{w}/_{o}$. The values in Table 5.1-1 are changed from the NEF because the MONK8A (SA, 2001) criticality analyses were performed with a revised correlation for the density of aqueous solutions of UO_2F_2.

Table 5.1-2, Safety Criteria for Buildings/ Systems/Components, lists the safety criteria of Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2, which are used as control parameters to prevent a nuclear criticality event. Although the EREF will be limited to 5.0 $^{w}/_{o}$ enrichment, as additional conservatism, the values in Table 5.1-2, Safety Criteria for Buildings/Systems/ Components, represent the limits based on 6.0 $^{w}/_{o}$ enrichment except for the Contingency Dump System traps which are limited to 1.5 $^{w}/_{o}$ ^{235}U. Insert

~~The values on Table 5.1-1 are chosen to be critically safe when optimum light water moderation exists and reflection is considered within isolated systems.~~ The conservative modeling techniques provide for more conservative values than provided in ANSI/ANS-8.1 (ANSI, 1998a). The product cylinders are only safe under conditions of limited moderation and enrichment. In such cases, both design and operating procedures are used to assure that these limits are not exceeded.

All Separation Plant components, which handle enriched UF_6, other than the Type 30B and 48Y cylinders and the first stage UF_6 pumps are safe by geometry. Centrifuge array criticality is precluded by a probability argument with multiple operational procedure barriers. Total moderator or H/U ratio control as appropriate precludes product cylinder criticality.

In the Technical Support Building (TSB) criticality safety for uranium loaded liquids is ensured by limiting the mass of uranium in any single tank to less than or equal to 12.2 kg U (26.9 lb U). Individual liquid storage bottles are safe by volume. Interaction in storage arrays is accounted for.

Based on the criticality analyses, the control parameters applied to EREF are as follows:

Enrichment

Enrichment is controlled to limit the percent ^{235}U within any process, vessel, or container, except the contingency dump system, to a maximum enrichment of 5 $^{w}/_{o}$. The design of the contingency dump system controls enrichment to a limit of 1.5 $^{w}/_{o}$ ^{235}U. Although EREF is limited to a maximum enrichment of 5 $^{w}/_{o}$, as added conservatism nuclear criticality safety is analyzed using an enrichment of 6 $^{w}/_{o}$ ^{235}U.

Geometry/Volume

NRC RAI Number: NCS-4 SAR Section 5.2.1.3

Insert

The values in Table 5.1-1 are calculated optimum moderation (i.e., various H/U ratios greater than and less than 7 are analyzed) and 30 cm water reflection.

against NPH criteria to ensure their failure will not result in flooding of areas containing enriched uranium above a critical mass.

Fire response to all process areas of the facility (whether by the on-site fire brigade or off-site fire department) requires that one member of the response team be assigned as the criticality safety officer. This individual is responsible to ensure that criticality safety is not compromised for any and all firefighting activities including the deployment of any fire hose streams in areas requiring reflection or moderator control.

Chapter 7, Fire Safety, contains additional discussion on fire system locations and application.

Interaction

Insert 1

NCSAs and NCSEs consider the potential effects of interaction. ~~A non-interacting unit is defined as a unit that is spaced an approved distance from other units such that the multiplication of the subject unit is not increased. The NEF facility SAR included a statement that indicated units may be considered non-interacting when they are separated by more than 60 cm (23.6 inches). The justification for 60 cm is based on a generic hand calculation. Although hand calculations are acceptable methods for defining NCS limits and restriction, the results may or may not be conservative for specific calculations and/or configurations. Spacing requirements will be determined on a system by system basis.~~

Insert 2

~~If a unit is considered interacting, NCSAs are performed.~~ Individual unit multiplication ~~and array interaction~~ are evaluated using the Monte Carlo computer code MONK8A to ensure k_{eff} (k_{calc} + $3\sigma_{calc}$) < 0.95.

Concentration, Density and Neutron Absorbers

EREF does not use mass concentration, density, or neutron absorbers as a criticality control parameter.

5.1.3 Safe Margins Against Criticality

Process operations require establishment of criticality safety limits. The facility UF_6 systems involve mostly gaseous operations. These operations are carried out under reduced atmospheric conditions (vacuum) or at slightly elevated pressures not exceeding three atmospheres. It is highly unlikely that any size changes of process piping, cylinders, cold traps, or chemical traps under these conditions, would lead to a criticality situation because a volume or mass limit may be exceeded.

Significant accumulations of enriched UF_6 reside only in the Product Low Temperature Take-off Stations, Product Liquid Sampling Autoclaves, Product Blending System, or the UF_6 cold traps. All these, except the UF_6 cold traps, contain the UF_6 in 30B and 48Y cylinders. All these significant accumulations are within enclosures protecting them from water ingress. The facility design has minimized the possibility of accidental moderation by eliminating direct water contact with these cylinders of accumulated UF_6. In addition, the facility's stringent procedural controls for enriching the UF_6 assure that it does not become unacceptably hydrogen moderated while in process. The plant's UF_6 systems operating procedures contain safeguards against loss of moderation control (ANSI, 1997). No neutron poisons are relied upon to assure criticality safety.

5.1.4 *Description of Safety Criteria*

Each portion of the plant, system, or component that may possibly contain enriched uranium is designed with criticality safety as an objective. Table 5.1-2, Safety Criteria for Buildings/ Systems/Components, shows how the safety criteria of Table 5.1-1, Safe Values for Uniform

Table 5.1-2 Safety Criteria for Buildings / Systems / Components
(Page 1 of 1)

Buildings/System/Components	Control Mechanism	Safety Criteria
Enrichment	Enrichment	5.0 $^{w}/o$ (6.0 $^{w}/o$ ^{235}U used in NCSAs)
Centrifuges	Diameter	< 22.4 cm (8.8 in)
Product Cylinders (30B)	Moderation	H < 0.92 kg (2.03 lb) – Note 1
Product Cylinders (48Y)	Moderation	H < 1.04 kg (2.29 lb) – Note 1
UF_6 Piping	Diameter	< 22.4 cm (8.8 in)
Chemical Traps	Diameter	< 22.4 cm (8.8 in)
Product Cold Trap	Diameter	< 22.4 cm (8.8 in)
Contingency Dump System Traps	Enrichment	1.5 $^{w}/o$ ^{235}U
Tanks	Mass	~~< 16.5 kg U (36.4 lb U)~~ – Note 2
Feed Cylinders	Enrichment	< 0.72 $^{w}/o$ ^{235}U
Uranium Byproduct Cylinders	Enrichment	< 0.72 $^{w}/o$ ^{235}U
UF_6 Pumps (first stage)	N/A	Safe by explicit calculation
UF_6 Pumps (second stage)	Volume	< 19.3 L (5.1 gal)
Individual Uranic Liquid Containers, e.g., Fomblin Oil Bottle, Laboratory Flask, Mop Bucket	Volume	< 19.3 L (5.1 gal)
Vacuum Cleaners Oil Containers	Volume	< 19.3 L (5.1 gal)

Notes:

1. Assumes outside storage (e.g., exposed to snow, ice, or rain).
2. Determined for double batch safe mass.

< 12.2 kg U (26.9 lb U) –

Most components that form part of the centrifuge plant or are connected to it reflect the assumption that any accumulation of uranium is taken to be in the form of a uranyl fluoride/water mixture at a maximum H/U atomic ratio of 7 (exceptions are discussed in the associated nuclear criticality safety analyses documentation). The ratio is based on the assumption that significant quantities of moderated uranium could only accumulate by reaction between UF_6 and moisture in air leaking into the plant process equipment. Due to the high vacuum requirements of a centrifuge plant, in-leakage is controlled at very low levels and thus the H/U ratio of 7 represents an abnormal condition. The maximum H/U ratio of 7 for the uranyl fluoride-water mixture is derived as follows:

The stoichiometric reaction between UF_6 and water vapor in the presence of excess UF_6 can be represented by the equation:

$$UF_6 + 2H_2O \rightarrow UO_2F_2 + 4HF$$

Due to its hygroscopic nature, the resulting uranyl fluoride is likely to form a hydrate compound. Experimental studies (Lychev, 1990) suggest that solid hydrates of compositions $UO_2F_2 \bullet 1.5\ H_2O$ and $UO_2F_2\ 2\ H_2O$ can form in the presence of water vapor, the former composition being the stable form on exposure to atmosphere.

It is assumed that the hydrate $UO_2F_2 \bullet 1.5\ H_2O$ is formed and, additionally, that the hydrogen fluoride (HF) produced by the UF_6/water vapor reaction is also retained in the uranic breakdown to give an overall reaction represented by:

$$UF_6 + 3.5H_2O \rightarrow UO_2F_2 \bullet 1.5H_2O \bullet 4HF$$

For the NCS calculations, the composition of the breakdown product was simplified to $UO_2F_2 \cdot 3.5H_2O$ that gives the same H/U ratio of 7 as above.

In the case of oils, UF_6 pumps and vacuum pumps use a fully fluorinated perfluorinated polyether (PFPE) type lubricant. Mixtures of UF_6 and PFPE oil would be as conservative a case as the uranyl fluoride/water mixture, since the maximum HF solubility in PFPE is only about $0.1^{w}/_{o}$. Therefore, the uranyl fluoride/water mixture assumption provides additional conservatism in this case.

5.2.1.3.4 Vessel Movement Assumption

The interaction controls placed on movement of vessels containing enriched uranium are specified in the facility procedures. In general, any item in movement (an item being either an individual vessel or a specified batch of vessels) must be maintained at the minimum required edge separation from any other enriched uranium, and that only one item of each type, e.g., one trap and one pump, may be in movement at one time. ~~The NEF facility SAR included a statement that indicated units may be considered non-interacting when they are separated by more than 60 cm (23.6 inches). The justification for 60 cm is based on a generic hand calculation. Although hand calculations are acceptable methods for defining NCS limits and restriction, the results may or may not be conservative for specific calculations and/or configurations.~~ Spacing requirements will be determined on a system by system basis. Insert 3 ~~These spacing restrictions are relaxed for vessels being removed from fixed positions. In this situation, one vessel may approach an adjacent fixed plant vessel/component without spacing restrictions.~~

5.2.1.3.5 Pump Free Volume Assumption

There are two types of pumps used in product and dump systems of the plant:

1a

NRC RAI Number: NCS-10 SAR Section 5.1.2

Insert 1

including interaction effects of in-transit materials. Spacing requirements will be determined on a system by system basis.

Insert 2

, array interaction, and in-transit material interactions

Insert 3

, including component insertion or extraction from an array,

- NCS analyses are performed using acceptable methodologies.
- Methods are validated and used only within demonstrated acceptable ranges.
- The analyses adhere to ANSI/ANS-8.1-1998 (ANSI, 1998a) as it relates to methodologies.
- The validation report statement in Regulatory Guide 3.71 (NRC, 2005) is as follows: EREF has demonstrated (1) the adequacy of the margin of safety for subcriticality by assuring that the margin is large compared to the uncertainty in the calculated value of k_{eff}, (2) that the calculation of k_{eff} is based on a set of variables whose values lie in a range for which the methodology used to determine k_{eff} has been validated, and (3) that trends in the bias support the extension of the methodology to areas outside the area or areas of applicability.
- A specific reference to (including the date and revision number) and summary description of either a manual or a documented, reviewed, and approved validation report for each methodology are included. Any change in the reference manual or validation report will be reported to the NRC by letter.
- The reference manual and documented reviewed validation report will be kept at the facility.
- The reference manual and validation report are incorporated into the configuration management program.
- The NCS analyses are performed in accordance with the methods specified and incorporated in the configuration management program.
- The NCS methodologies and technical practices in NUREG-1520 (NRC, 2002), Section 5.4.3.4, are used to analyze NCS accident sequences in operations and processes. Insert
- The acceptance criteria in NUREG-1520 (NRC, 2002), Section 3.4, as they relate to: identification of NCS accident sequences, consequences of NCS accident sequences, likelihood of NCS accident sequences, and descriptions of IROFS for NCS accident sequences are met.
- NCS controls and controlled parameters to assure that under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety are used.
- As stated in ANSI/ANS-8.1-1998 (ANSI, 1998a), process specifications incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded.
- ANSI/ANS-8.7-1998 (ANSI, 1998b), as it relates to the requirements for subcriticality of operations, the margin of subcriticality for safety, and the selection of controls required by 10 CFR 70.61(d) (CFR, 2008a), is used.
- ANSI/ANS-8.10-1983 (ANSI, 1983), as modified by Regulatory Guide 3.71 (NRC, 2005), as it relates to the determination of consequences of NCS accident sequences, is used.
- If administrative k_{eff} margins for normal and credible abnormal conditions are used, NRC pre-approval of the administrative margins will be sought.
- Subcritical limits for k_{eff} calculations such that: k_{eff} subcritical = 1.0 - bias - margin, where the margin includes adequate allowance for uncertainty in the methodology, data, and bias to assure subcriticality are used.

NRC RAI Number: NCS-1 SAR Section 5.2.1.5

Insert:

except for Item 4 in Section 5.4.3.4.1 and Items 9, 13 and 16 in Section 5.4.3.4.2.

Each area

5.3 CRITICALITY ACCIDENT ALARM SYSTEM (CAAS)

The facility is provided with a Criticality Accident Alarm System (CAAS) as required by 10 CFR 70.24, (CFR, 2008d). ~~Areas~~ where Special Nuclear Material (SNM) is handled, used, or stored in amounts at or above the 10 CFR 70.24 (CFR, 2008d) mass limits are provided with CAAS coverage. Emergency management measures are covered in the facility Emergency Plan.

INSERT 1

INSERT 2

NRC RAI Number: NCS-7 SAR Section 5.3

Insert 1

The CAAS will be uniform throughout the facility for the type of radiation detected and alarm signals. Documentation shall be maintained which demonstrate the CAAS meets the requirements of 10 CFR 70.24.

Insert 2

The CAAS is provided with emergency power and is designed to remain operational during credible events or conditions, including fire, explosion, corrosive atmosphere, or seismic shock (equivalent to the site-specific design-basis earthquake or the equivalent value specified by the uniform building code).

Whenever the CAAS is not functional, compensatory measures, such as limiting access and restricting SNM movement, will be implemented. Should the CAAS coverage be lost and not restored within a specified number of hours or an equivalent level of protection has not been provided (e.g., portable CAAS system), the operations will be rendered safe (by shutdown and quarantine) if necessary. Onsite guidance is provided and is based on process-specific considerations that consider applicable risk trade-off of the duration of reliance on compensatory measures versus the risk associated with process upset in shutdown.

Depleted UF_6 (tails), will be removed from the site prior to and during decommissioning. As described in Section 10.3, the tails will be transported to the Department of Energy (DOE) facilities at Portsmouth, Ohio or Paducah, Kentucky for conversion and disposal in accordance with regulatory requirements. Radioactive wastes will be disposed of in licensed low-level radioactive waste disposal sites. Hazardous wastes will be treated or disposed of in licensed hazardous waste facilities. Neither tails conversion, nor disposal of radioactive or hazardous material will occur at the plant site, but at licensed facilities located elsewhere.

Following decommissioning, no part of the facilities or site will remain restricted to any specific type of use.

AES has compared the EREF to the National Enrichment Facility (NEF) in Lea County, New Mexico and fully expects that the decommissioning costs for the EREF are comparable to the decommissioning costs for the NEF, accounting for facility enrichment capacity and minor differences in infrastructure. The supplier of the centrifuges and associated equipment for the EREF i(ETC) has supplied a cost estimate to decommission the centrifuges and associated classified equipment of a 3.3M SWU facility. Costs and quantities associated with decommissioning of a 3.3M SWU facility have been increased by factors based on the particular item to account for the increase in facility capacity to 6.6M SWU. The factors are provided in the appropriate tables associated with this chapter. Decommissioning and decontamination of these components, which is classified, represents approximately 97% of the costs for decommissioning of the EREF. Insert A

The remaining structures, systems, and components (SSCs) to be decommissioned account for approximately 3% of the total cost for decommissioning the facility. AES has developed the costs for decommissioning of these SSCs assuming that the costs are approximately the same as for the NEF, accounting for facility enrichment capacity and minor differences in infrastructure. This is based on the following:

- The overall design and quantities of the SSCs at the two facilities (EREF and NEF) that are to be decommissioned are similar when differences in capacity and infrastructure are taken into account.
- The practices and procedures that will be used to decommission and decontaminate the SSCs at EREF will be similar to those to be used at NEF.

Therefore, the decommissioning and decontamination quantities and costs developed for the NEF for non-classified structures, systems, and components are applicable to the EREF on both overall and unit bases, taking into account differences in capacity and minor differences in facility infrastructure. Where differences do exist, for example more or less floor area, the NEF costs are adjusted as appropriate for the conditions at the EREF.

NRC requested that LES provide a comparison of NEF decommissioning unit costs with the unit costs provided in NUREG/CR-6477 (NRC, 2002b). LES provided the comparison (LES, 2005) and determined that the NEF unit costs and the NUREG unit costs were comparable. Since the EREF decommissioning unit costs are based on NEF decommissioning costs, it can be concluded that the EREF decommissioning unit costs are also comparable to the unit costs computed from NUREG/CR-6477 (NRC, 2002b). Refer to Table 10.1-15 for this unit cost comparison.

Disposal costs for the low level radioactive waste (LLRW) generated during decommissioning may differ between the EREF and the NEF. This is a result of assuming disposal at different LLRW disposal facilities. NEF waste is assumed to be disposed of at the Envirocare Facility in Clive, Utah. However, the state of Idaho is a member of the Northwest Interstate Compact (NWIC) on Low Level Radioactive Waste. Therefore, for the purposes of this analysis, LLRW 1a

Insert After 5th para., Section 10.1.4

ETC has decommissioning experience related to the E21 Enrichment Plant (at Capenhurst, UK) for early generations of block-mounted centrifuges and the SP3 Separation Plant at Almelo, Netherlands for early generation centrifuges. ETC has maintained a decommissioning cost model based on costs and methodologies of work previously performed.

Due to the continued high performance of the current generation of gas centrifuges no site has yet needed, or is likely in the short term, to decommission a plant containing centrifuge machines similar to those proposed for the EREF. ETC has experience in taking both intact and crashed centrifuges from these plants for autopsy and from refurbishment campaigns for its customers. Centrifuges have been manually dismantled, decontaminated and declassified by ETC-D in Jülich, Germany, with the resulting parts safely disposed of – this includes both TC12 and later generation development centrifuges. Work has been performed by ETC scaling the manual methodologies used in Jülich into a semi-automated line using a design concept based on that currently deployed on the SP3 decommissioning line. A cost estimate for this concept was included in the decommissioning cost model.

Table 10.1-1C Number and Dimensions of Facility Components (Page 1 of 1)

Technical Support Building			
Component	**Number of Components**	**Dimensions of Components**	**Total Dimensions**
Glove Boxes	None	NA	NA
Fume Hoods	18	Standard laboratory fume hoods, approx 6.5 - 8 feet (2 to 2.5 m) high x 5 feet (1.5 m) wide	(Note 1)
Lab Benches	25	Various sizes of lab and workshop benches ranging from 6.5 -13 feet (2 to 4 m) long by 2.5 feet wide (0.75 m)	(Note 1)
Sinks	12	Standard laboratory sinks and hand wash basins	(Note 1)
Drains	12	Standard Laboratory type drains	(Note 1)
Floors	~~(Note 3)~~	Floor area covers all Workshops and Labs in the Technical Services Bldg that may be exposed to contamination	70,440 ft^2 (6,544 m^2)
Walls (Note 3, 4)	~~(Note 3)~~	Wall area covers all Workshops and Labs in the Technical Services Bldg that may be exposed to contamination	146,704 ft^2 (13,629 m^2)
Ceilings	~~(Note 3)~~	Ceiling area covers all Workshops and Labs in the Technical Services Bldg that may be exposed to contamination	70,440 ft^2 (6,544 m^2)
Ventilation/Ductwork	(Note 3)	Various pieces of equipment including, filter banks, extractor fans, vent stack, dampers and approx 1,200 feet (366 m) of large and small ductwork	1,200 ft (366 m)
Hot Cells	None	NA	NA
Equipment/Materials	57	Various pieces of equipment including, mass spectrometers, hydraulic lift tables, cleaning cabinets	(Note 1)
Soil Plots	None	NA	NA
Storage Tanks	16	Waste oil storage tank (53 gal) (201 l) and Liquid Effluent Collection and Treatment System Tanks	NA
Storage Areas	2	Storage area for product removal, dirty pumps	(Note 1)
Radwaste Areas	None	NA	NA
Scrap Recovery Areas	None	NA	NA
Maintenance Shop	4 ~~1~~	Third Floor Maintenance Facility	(Note 4) ~~(Note 1)~~
Equipment Decontamination Areas	~~1~~	Third Floor Decontamination Workshop	~~(Note 1)~~
Other	1 Lot (Note 2)	Hand tools and consumables that become contaminated while carving out dismantling/ decontamination work, unmeasured work and scaffolding	(Note 1)

Notes:

1. Total dimensions not used in estimating model.
2. Allocation based on European decommissioning experience
3. Total Dimensions provided
4. Floor, wall, and ceiling areas of Third Floor Maintenance Facility and Third Floor Decontamination Workshop are included in overall totals.

Table 10.1-3 Decontamination or Dismantling of Radioactive Components - (Man-Hours)
(Page 1 of 1)

Other Buildings (Note 1)					
Component	Decon Method (Note 4)	Craftsman	Supervision (Note 2)	Project Management	HP&S/Chem (Note 3)
Glove Boxes		0	0	0	0
Fume Cupboards		382	76	65	81
Lab Benches		397	78	67	83
Sinks		124	24	21	26
Drains		125	24	21	26
Floors		2,184	435	375	459
Walls		2,126	423	363	448
Ceilings		928	186	159	196
Ventilation/Ductwork/ Piping		20,217	4,042	3,455	4,250
Hot Cells		0	0	0	0
Equipment/Materials		1,877	376	321	394
Soil Plots		0	0	0	0
Storage Tanks		37	8	5	8
Storage Areas		135	27	23	28
Radwaste Areas		0	0	0	0
Scrap Recovery Areas		0	0	0	0
Maintenance Shop		0	0	0	0
Equipment Decontamination Areas		0	0	0	0
Other		2,342	468	400	492
Total Hours		30,873	6,168	5,274	6,491

Notes:

1. Includes the Decontamination Facility, Technical Support Building, Gaseous Effluent Ventilation System throughout Plant, Blending, Sampling, and Preparation Building, and Centrifuge Test and Post Mortem Facilities.
2. Supervision at 20%.
3. Supply ongoing monitoring and analysis service for dismantling teams.
4. Specific details of decontamination method not defined at this time.

General decontamination methods to be used at EREF are discussed in Section 10.1.6.

Table 10.1-10 Packaging, Shipping, and Disposal of Radioactive Wastes (Excluding Labor Costs)
(Page 1 of 2)

(a) Waste Disposal Costs (includes packaging and shipping costs)

Materials	Disposal Volume ft^3 / (m^3)		Unit Cost ($/ft^3)	# of Containers (Note 2)	Total Disposal Costs ($000)
Other Buildings	(Note 4)		(Note 4)		(Note 4)
Miscellaneous low level waste	5,716	162	410	97	$2,346
Separation Modules:					
Solidified Liquid Wastes	30,512	864	410	519	$12,522
Centrifuge Components, Piping and Other Parts	15,044	426	410	256	$6,174
Aluminum (Note 3)	218,951	6,200	220	1,063	$48,193
TOTAL (Note 1)	270,223	7,652	-	1,935	$69,235

Notes

1. A revenue cap is imposed on the company that operates the US Ecology disposal facility for the Northwest Compact. On reaching this cap, facilities that dispose naturally occurring radioactive materials such as the EREF may be refunded a portion of their disposal costs. The projected costs do not include an allowance for potential refunds and are therefore conservative.
2. Assumes waste is shipped in Sea Land and B-25 containers and either direct buried or buried in the B-25 containers
3. Aluminum Waste is composed of smelted classified equipment (centrifuges).
4. The values provided for Disposal Volume in ft3, Unit Cost, and Total Disposal Costs reflect rounding.

Table 10.1-10 Packaging, Shipping, and Disposal of Radioactive Wastes (Excluding Labor Costs)

(Page 2 of 2)

(b) Processing Costs

Materials	(Note 2) Disposal Weight tons / (Mt)		(Note 2) Unit Cost ($/lb)	(Note 2) Total Processing Costs ($000)
Aluminum	19,401	17,600	0.218	$8,510
Other materials	589	534	4.18	$4,924
TOTAL	19,989	18,134	-	$13,434

Note:

1. Processing costs represent those costs required to declassify the classified equipment

2. The values provided for Disposal Volume in tons, Unit Cost, and Total Disposal Costs reflect rounding.

Table 10.1-15 Unit Cost Comparison
(Page 2 of 3)

Notes:

1. <u>Lab benches / Sinks / Fume Hoods/ Tools / Equipment / Materials</u>

   Good radiological management procedures will be observed throughout operations within the Separation Building, Technical Support Building (TSB) and the final Decommissioning Facility consistent with AES commitments to maintain occupational doses and doses to members of the public as low as reasonably achievable (ALARA). Consequently contamination occurring on the working surfaces of lab benches / sinks / tools / fume hoods will be monitored, cleaned and maintained in good order through the day-to-day working operation. Therefore, at decommissioning, it is not anticipated that additional decontamination of these items will be required. The items will be dismantled, volume reduced, radiologically characterized and shipped to a licensed disposal facility. For the sinks in the final Decommissioning Facility, at the end of decommissioning, these sinks will be cleaned, volume reduced and shipped to a licensed disposal facility.

   Any contaminated tools, for which it proves not to be cost effective to maintain clean during operations, will be replaced with new tools during operations. Consequently, at close of operations only one set of tools will be required to be decontaminated and shipped to a licensed disposal facility.

2. <u>Ventilation Ductwork</u>

   Experience has shown ventilation ductwork to be only lightly contaminated. As such, the ductwork will be dismantled, volume reduced, radiologically characterized and shipped to a licensed disposal facility.

3. <u>Drains</u>

   There are no process drains in the EREF Separations Building. In the TSB, there are drains from all rooms where operations or processes of a potentially contaminated nature are undertaken to a liquid effluent collection and treatment room. These drains will be removed, decontaminated, volume reduced and shipped to a licensed disposal facility.

4. <u>Floors, Walls, Ceilings and Storage Areas</u>

   Experience from European decommissioning of Separations Buildings has shown that there is no contamination on walls, ceilings and floors in the buildings at the end of their life. This has been confirmed by radiological characterization at the end of operations and at the end of building strip out prior to demolition. This lack of contamination results from the proven contained nature of the vacuum processes and good operational practices, including implementation of the ALARA program throughout the entire facility, which support maintenance of a clean facility throughout the operational life.

   ~~For the TSB and final Decommissioning Facility, an allowance has been conservatively provided in the cost estimate for cleaning of areas within the TSB and the floors, walls, and ceilings in the final Decommissioning Facility.~~

   All floors, walls, ceilings in potentially contaminated areas are assumed to be contaminated. The estimates for cleaning of the floors, walls, and ceilings are consistent with similar work performed for decontamination of floors, walls, and ceilings at decommissioning nuclear power plants. The levels of contamination encountered during the decommissioning of nuclear power plants far exceed the levels anticipated at EREF, therefore, the estimates are conservative.

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- Audit and assessment reports
- Emergency operating procedures
- Emergency response plans
- System modification documents
- Assessment reports
- Engineering documents including analyses, specifications, technical reports, and drawings. These items are documented in approved procedures.

11.1.4 Change Control

Procedures control changes to the technical baseline. The process includes an appropriate Level of technical, management, and safety review and approval prior to implementation. During the design phase of the project, the method of controlling changes is the design control process described in the QA Program. This process includes the conduct of interdisciplinary reviews that constitute a primary mechanism for ensuring consistency of the design with the design bases. During both construction and operation, appropriate reviews to ensure consistency with the design bases of QA Level 1 and QA Level 2 items and activities and the ISA will ensure that the design is constructed and operated/modified within the limits of the design basis. Additional details are provided below.

11.1.4.1 Design Phase

Changes to the design include a systematic review of the design bases for consistency. In the event of changes to reflect design or operational changes from the established design bases, both the integrated safety analysis and other documents affected by design bases of QA Level 1 and QA Level 2 items and activities including the design requirements document and basis of design documents, as applicable are properly modified, reviewed, and approved prior to implementation. Approved changes are made available to personnel through the document control function discussed previously in this section.

Materials

submit

During design (i.e., prior to issuance of the EREF Materials License), the method of ensuring consistency between documents, including consistency between design changes and the safety assessment, is the interdisciplinary review process. The interdisciplinary reviews ensure design changes either: (1) do not impact the ISA; (2) are accounted for in subsequent changes to the ISA; or (3) are not approved or implemented. Prior to issuance of the License, AES will notify the NRC of potential changes that reduce the level of commitments or margin of safety in the design bases of QA Level 1 and QA Level 2 items and activities.

to the NRC for review and approval prior to implementation of the change

11.1.4.2 Construction Phase

the NRC's

When the project enters the construction phase, changes to documents issued for construction, fabrication, and procurement will be documented, reviewed, approved, and posted against each affected design document. Vendor drawings and data also undergo an interdisciplinary review when necessary to ensure compliance with procurement specifications and drawings, and to incorporate interface requirements into facility documents.

During construction, design changes will continue to be evaluated against the approved design bases. Changes are expected to the design as detailed design progresses and construction begins. A systematic process consistent with the process described above will be used to

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- Work Request # (if applicable)
- Test Frequency
- Plant Cascade #
- Last date test was performed
- Next date test is due.

Special tests are considered to be a facility change or change in process safety information that may alter the parameters of an accident sequence. As described in SAR Section 3.0.2, these tests must be reviewed by the ISA method(s) as described in the ISA Summary. In addition, special tests will be reviewed in accordance with 10 CFR 70.72(a) and (c).

In the event that a test cannot be performed within its required interval due to system or plant unit conditions, the responsible department notifies the on-duty Production Manager and processes the condition in accordance with the Corrective Action program. The responsible department lists the earliest possible date the test could be performed and the latest date along with the required system or unit-mode condition. However, the responsible department will ensure that the test is performed as soon as practical once required conditions are met, regardless of the estimated date given earlier.

Periodic testing and surveillance associated with QA Level 1 and QA Level 2 items and activities are performed in accordance with written procedures.

11.2.4.4.2 Special Testing

Special testing is testing conducted at the facility that is not a facility preoperational test, periodic test, post-modification test, or post-maintenance test. Special testing is of a non-recurring nature and is conducted to determine facility parameters and/or to verify the capability of QA Level 1 and QA Level 2 items to meet performance requirements. Purposes of special testing include, but are not necessarily limited to, the following:

A. Acquisition of particular data for special analysis

B. Determination of information relating to facility incidents

C. Verification that required corrective actions reasonably produce expected results and do not adversely affect the safety of operations

D. 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 determination that a certain plant activity is a Special Test is intended to exclude those plant activities which are routine surveillances, normal operational evolutions, and activities for which there is previous experience in the conduct and performance of the activity. At the discretion of the Plant Manager, a test may be conducted as a special test. In making this determination, facility management includes the following evaluations of characteristics of the activity:

A. Does the activity involve an unusual operational configuration for which there is no previous experience?

B. Does the activity have the propensity, if improperly conducted, to significantly affect primary plant parameters?

C. Does the activity involve seldom-performed evolutions, meeting one of the above criteria, in which the time elapsed since the previous conduct of the activity renders prior experience not useful?

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11.4 PROCEDURES DEVELOPMENT AND IMPLEMENTATION

~~The requirements for independent verification are consistent with the applicable guidance provided in ANSI/ANS-3.2-1994, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," (ANSI, 1994).~~

Activities involving licensed materials or QA Level 1 and QA Level 2 items and activities are conducted in accordance with approved procedures. Before initial enrichment activities occur at the facility, procedures are made available to the NRC for their inspection. As noted throughout this document, procedures are used to control activities in order to ensure the activities are carried out in a safe manner and in accordance with regulatory requirements.

Generally, four types of plant procedures are used to control activities: operating procedures, administrative procedures, maintenance procedures, and emergency procedures.

Operating procedures, developed for workstation and Control Room operators, are used to directly control process operations. Operating procedures include:

- Purpose of the activity
- Regulations, polices, and guidelines governing the procedure
- Type of procedure
- Steps for each operating process phase:
    - Initial startup
    - Normal operations
    - Temporary operations
    - Emergency shutdown
    - Emergency operations
    - Normal shutdown
    - Startup following an emergency or extended downtime.
- Hazards and safety considerations
- Operating limits
- Precautions necessary to prevent exposure to hazardous chemicals (resulting from operations with Special Nuclear Material (SNM)) or to licensed SNM.
- Measures to be taken if contact or exposure occurs
- IROFS associated with the process and their functions
- The timeframe for which the procedure is valid.

Applicable safety limits and IROFS are clearly identified in the procedures. AES will incorporate methodology for identifying, developing, approving, implementing, and controlling operating procedures. Identifying needed procedures will include consideration of ISA results. The method will ensure that, as a minimum:

- Operating limits and IROFS are specified in the procedure
- Procedures include required actions for off-normal conditions of operation, as well as normal operations

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inadvertent departure from a procedure could cause an inadvertent criticality. Nuclear criticality safety postings at the EREF are established that identify administrative controls applicable and appropriate to the activity or area in question. Nuclear criticality safety procedures and postings are controlled by procedure to ensure that they are maintained current.

Periodic reviews will be performed on procedures to assure their continued accuracy and usefulness. In addition, applicable procedures will be reviewed after unusual incidents, such as an accident, unexpected transient, significant operator error, or equipment malfunction, or after any modification to a system, and procedures will be revised as needed.

At a minimum all operating procedures are reviewed every 5 years and emergency procedures are reviewed every year.

11.4.1 Preparation of Procedures

Each procedure is assigned to a member of the facility staff or contractor for development. Initial procedure drafts are reviewed by other appropriate members of the facility staff, by personnel from the supplier of centrifuges (ETC), and other vendors, as appropriate for inclusion and correctness of technical information, including formulas, set points, and acceptance criteria and includes either a walkdown of the procedure in the field or a tabletop walkthrough. Procedures that are written for the operation of QA Level 1 and QA Level 2 items shall be subjected to an independent review. The designated approver shall determine whether or not any additional, cross-disciplinary review is required. The Plant Manager or designee shall approve all procedures. If the procedure involves QA directly, the QA Manager must approve the procedure.

performed by personnel not having direct responsibility for the work function under review.

11.4.2 Administrative Procedures

Facility administrative procedures are written by each department as necessary to control activities that support process operations, including management measures. Listed below are several areas for which administrative procedures are written, including principle features:

A. Operator's authority and responsibility: The operator is given the authority to manipulate controls which directly or indirectly affect the enrichment process, including a shut down of the process if deemed necessary by the Production Manager. The operators are also assigned the responsibility for knowing the limits and set points associated with safety-related equipment and systems as specified in designated operating procedures.

B. Activities affecting facility operation or operating indications: All facility maintenance personnel performing support functions (e.g., maintenance, testing) which may affect unit operation or Control Room indications are required to notify the Control Room Operator and/or Production Manager, as appropriate, prior to initiating such action.

C. Manipulation of facility control: Only operators are permitted to manipulate the facility controls, except for operator trainees under the direction of a qualified operator.

D. Relief of Duties: This procedure provides a detailed checklist of applicable items for shift turnover.

E. Equipment control: Equipment control is maintained and documented through the use of tags, labels, stamps, status logs or other suitable means.

F. Master surveillance testing schedule: A master surveillance testing schedule is documented to ensure that required testing is performed and evaluated on a timely basis. Surveillance testing is scheduled such that the safety of the facility is not dependent on the performance of a structure, system or component which has not been tested within its specified testing interval. The master surveillance testing schedule 1a identifies surveillance and testing requirements, applicable procedures, and required test frequency. Assignment of responsibility for these requirements is also indicated.

G. A Control Room Operations Logbook is maintained. This logbook contains significant events during each shift such as enrichment changes, alarms received, or abnormal operational conditions.

H. Fire Protection Procedures: Fire protection procedures are written to address such topics as training of the fire brigade, reporting of fires, and control of fire stops. The Safety, Security and Emergency Preparedness Manager has responsibility for fire protection procedures in general, with the facility's maintenance section having responsibility for certain fire protection procedures such as control of repairs to facility fire stops.

The administrative control of maintenance is maintained as follows:

A. In order to assure safe, reliable, and efficient operation, a comprehensive maintenance program for the facility's QA Level 1 and QA Level 2 items is established.

B. Personnel performing maintenance activities are qualified in accordance with applicable codes and standards and procedures.

C. Maintenance is performed in accordance with written procedures that conform to applicable codes, standards, specifications, and other appropriate criteria.

D. Maintenance is scheduled so as not to jeopardize facility operation or the safety of facility personnel.

E. Maintenance histories are maintained on facility QA Level 1 and QA Level 2 items.

The administrative control of facility modifications is discussed in Section 2.3.1, Configuration Management.

11.4.3 Procedures

Activities involving licensed materials or QA Level 1 and QA Level 2 items and activities are conducted in accordance with approved procedures. These procedures are intended to provide a pre-planned method of conducting operations of systems in order to eliminate errors due to on-the-spot analysis and judgments.

Procedures are sufficiently detailed that qualified individuals can perform the required functions without direct supervision. However, written procedures cannot address all contingencies and operating conditions. Therefore, they contain a degree of flexibility appropriate to the activities being performed. Procedural guidance exists to identify the manner in which procedures are to be implemented. For example, routine procedural actions may not require the procedure to be present during implementation of the actions, while complex jobs or checking with numerous sequences may require valve alignment checks, approved operator aids, or in-hand procedures that are referenced directly when the job is conducted. In addition, procedural guidance exists to define when verification of significant steps is required.

Examples of operating activities are:

- Evacuation and Preparatory Work Before Run Up of a Cascade
- Run Up of a Cascade
- Run Down of a Cascade
- Calibration of Pressure Transmitter

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- Taking UF_6 Samples of a Cascade
- Installation of UF_6 Cylinders in Feed/Take-off Stations and Preparation for Operation
- Removal of UF_6 Cylinder from Feed/Take-off Stations
- Installation of UF_6 Cylinders in Take-off Stations
- UF_6 Gas Sampling in Take-off Lines
- UF_6 Sampling in Product Liquid Sampling Autoclaves
- Emptying of Cold Trap
- Exchange of Chemical Traps in Vent Systems.

Plant specific procedures for abnormal events are written for the facility. These procedures are based on a sequence of observations and actions, with emphasis placed on operator responses to indications in the Control Room. When immediate operator actions are required to prevent or mitigate the consequences of an abnormal situation, procedures require that those actions be implemented at the earliest possible time, even if full knowledge of the abnormal situation is not yet available. The actions outlined in abnormal event procedures are based on a conservative course of action to be followed by the operating crew.

Typical abnormal event procedures include:

- Power Failure
- Loss of Heat Tracing
- Damaged UF_6 Cylinder Repairs
- Annunciator alarms (procedures to include alarm set points, probable causes, automatic actions, immediate manual actions, supplementary actions and applicable references).

Temporary changes to procedures are issued for operating activities that are of a nonrecurring nature. Temporary changes to procedures are used when revision of an operating or other permanent procedure is not practical. Temporary changes to procedures shall not involve a change to the ISA and shall not alter the intent of the original procedure. Examples of uses of temporary changes to procedures are:

- To direct operating activities during special testing or maintenance
- To provide guidance in unusual situations not within the scope of normal procedures
- To ensure orderly and uniform operations for short periods of time when the facility, a unit, a cascade, a structure, a system or a component is performing in a manner not addressed by existing procedures or has been modified in such a manner that portions of existing procedures do not apply.

The temporary changes to procedures are approved by two members of the facility management staff, at least one of whom is a Production Manager. Temporary changes to procedures are documented, reviewed and approved with the process described in Section 11.4.4, Changes to Procedures~~, within 14 days of implementation~~.

In addition, the approved duration (e.g., expiration date) for the temporary change will be identified on the temporary procedure change in accordance with 10 CFR 70.72(a)(5).

Maintenance of facility structures, systems and components is performed in accordance with written procedures, documented instructions, checklists, or drawings appropriate to the circumstances (for example, skills normally possessed by qualified maintenance personnel may not require detailed step-by-step delineation in a written procedure) that conform to applicable codes, standards, specifications, and other appropriate criteria.

6.0 DOCUMENT CONTROL

6.1 Documents and changes to documents that prescribe or specify quality requirements or activities affecting the availability and/or reliability of IROFS or credited attributes of safe-by-design components are controlled in a manner that assures the use of correct documents. Such documents, including changes thereto, are reviewed for adequacy and approved for release by authorized personnel.

6.2 Procedures and instructions assure that documents are prepared; reviewed for adequacy, correctness, and completeness by a qualified individual; approved for release by authorized personnel; distributed to the location where the activity is performed prior to commencing work; and used in performing the activity. Obsolete or superseded documents are removed or appropriately identified. Procedures identify documents to be controlled; responsibility for preparing, reviewing, approving and issuing documents to be used; and require the establishment of current and updated distribution lists. Procedures also require the creation and maintenance of a controlled document index to track and control approved revision levels of those documents.

6.3 Changes to documents other than minor changes are reviewed for adequacy, correctness and completeness, prior to approval and issuance. Major changes are reviewed and approved by the same organization that performed the original review and approval unless other organizations are specifically designated. ~~Temporary procedure changes that do not change the intent of procedures may be made at the work location by responsible management.~~ Temporary changes to procedures are approved by two members of the facility management staff, at least one of whom is a Production Manager. The applicable procedure controls the process, documentation and approval of the temporary changes.

6.4 Minor changes to documents, such as inconsequential editorial corrections, may be made to documents without being subject to the review and approval of the requirements specified above. The applicable procedure defines the organizational positions authorized and criteria acceptable for making minor changes.

For QA Level 1 items, at least one other method of supplier evaluation is used in addition to performance history.

similar product that performs satisfactorily in actual use. The supplier's history will reflect current capability. This evaluation will examine the potential supplier's current Quality Program and Implementing Procedures along with the associated Quality Records as supported by qualitative and quantitative information that can be objectively evaluated.

- Depending on the part or service involved, a supplier QA program meeting the applicable requirements of accepted industry regulations or standards such as, but not limited to, NQA-1, ISO 9001, American National Standards Institute (ANSI) Z540-1, 10 CFR Part 50, Appendix B, or 10 CFR 830.120, may be acceptable. When actions that demonstrate the implementation of the QA program have commenced, the potential supplier's technical: and quality capability is determined by a direct evaluation of the supplier's personnel, and implementation of the supplier's quality assurance program. Supplier audits are conducted in accordance with Section 18.2 of this QAPD.
- For Calibration Services if the supplier has a valid Certificate of Accreditation issued by the National Voluntary Laboratory Accreditation Program (NVLAP) of the National Institute of Standards and Technology (NIST).
- The potential supplier maintains and implements a NRC approved QA program. When using this method, an initial implementation audit will be performed in accordance with Section 18.2 of this QAPD.
- The supplier maintains a valid ASME Code certification for the item or service being provided. When using this method, an initial implementation audit will be performed in accordance with Section 18.2 of this QAPD.

7.4.2 Suppliers with acceptable technical, quality and commercial qualifications are placed on the ASL maintained by the QA organization. Retention on the list is based on performance.

7.4.3 Measures are established to interface with the supplier and to verify supplier's performance, as necessary. The purchaser's verification activities; however, do not relieve the supplier of responsibility for verification of quality achievement. The measures include:

- Establishing an adequate understanding between AES and the supplier on the provisions and specifications of the procurement documents;
- Requirements for the supplier to identify the methods and processes to be used by the supplier in fulfilling the requirements of the procurement;
- Reviewing the supplier documents generated or processed during activities fulfilling procurement requirements;
- Identifying and processing necessary change information;
- Establishing methods for exchange of information with the supplier; and
- Establishing the extent of source surveillance and inspection activities for subtier suppliers.

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- ~~Supplier qualification and performance history. For QA Level 1 items, at least: one of the other methods of acceptance is used in addition to performance history.~~

7.4.7 Documented evidence of acceptability must be complete prior to placing an item in service. Controls are established for conditional release, such as for post-installation testing.

7.4.8 Acceptance of services is based on one or more of the following methods:

- Technical verification of data produced;
- Surveillance and/or audit of the activity; and
- Review of objective evidence for conformance to procurement document requirements.

For QA Level 1 items, a Certificate of Conformance plus one or more of the other methods, established above, is used to establish acceptance of items.

For QA Level 2 items, any one or more of the methods, established above, is used to establish acceptance of items.

Acceptance of services includes review of contractor deliverables (including documentation and records), determination of acceptability for AES use, completion of acceptance testing, completion of start-up testing, turnover, etc.

Supplier nonconformances are processed in accordance with Section 15.0 of this QAPD. Supplier nonconformances consist of one or more of the following:

- Violation of technical or material requirement of AES-supplied documents;
- Violation of requirement of purchaser-approved supplier documents.

Supplier nonconformances may be identified either by AES or by the supplier. For a supplier identified nonconformance, the supplier shall include a recommended disposition and technical justification for the identified condition. Nonconforming items are not released for use until the nonconforming condition is reviewed and accepted by Engineering and the implementation of the disposition is verified, except under conditional release provisions. Records of supplier nonconformance are maintained.

Procurement of QA Level 1 and QA Level 2 items and services by Commercial Grade Dedication

The methods to procure commercially available items and services will be performed in accordance with approved procedures. The criteria and methods for identifying the critical characteristics utilized for acceptance are established and are subject to design control measures in accordance with Section 3 of this QAPD. The critical characteristics, which once selected to be verified, provide reasonable assurance that the item or service provided meets specified requirements. In selecting the critical characteristics, the impact of the activities associated with the item or service on the safety function of plant equipment is considered.

7.5.2 Commercial grade items are identified in procurement documents by

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3.0 ASSUMPTIONS

3.1 KEY ASSUMPTIONS

A key assumption is any assumption or limitation that must be verified prior to using the results and/or conclusions of a calculation for a safety-related task. There are no key assumptions in the present calculation.

3.2 JUSTIFIED ASSUMPTIONS

None.

3.3 MODELING SIMPLIFICATIONS AND CONSERVATISM

The EREF NCSAs use several conservative assumptions in the modeling. These conservatisms are as follows.

For most components that form part of the centrifuge plant or are connected to it, any accumulation of uranium is taken to be in the form of a uranyl fluoride / water mixture at a maximum H/U atomic ratio of 7 (exceptions are product cylinders, vacuum pumps and UF_6 sample bottles.). This is based on the assumption that significant quantities of moderated uranium could accumulate by reaction between UF_6 and moisture in air leaking into the plant. Due to the high vacuum requirements of a centrifuge plant, in-leakage is controlled at very low levels and thus the condition assumed above represents an abnormal condition. The H/U ratio of 7 assumption is conservative and the H/U ratio is not expected to be higher than 7. Higher H/U ratios due to excessive air in-leakage are precluded since the condition would cause a loss of vacuum which in turn would cause the affected centrifuges to crash and the enrichment process to stop. In case of oils, UF_6 pumps and vacuum pumps use a fully fluorinated PFPE (perfluorinated polyether) type lubricant. Mixtures of UF_6 and PFPE oil would be a less pessimistic case than the uranyl fluoride / water mixture considered since maximum hydrogen fluoride (HF) solubility in PFPE is only ~ 0.1% by weight [9]. Insert 2

A uranyl fluoride / water system is the worst combination of materials that can occur in an ETC supplied centrifuge enrichment facility with regard to nuclear criticality safety. In addition, uranium compounds with alumina (Al_2O_3), PFPE oil or active carbon are less reactive than a uranyl fluoride / water system. Alumina and PFPE oil systems are less reactive because they contain no hydrogen to act as a moderating material, and active carbon systems are less reactive because carbon/graphite is a less efficient moderator than hydrogen. In addition, the uranyl fluoride / water system is considered to be worse than any normal non-moderated system. Therefore, the uranyl fluoride / water system is the only system that needs to be included in the benchmark. Additional compounds are used in the benchmark experiments. The justification for using these additional compounds is discussed in Section 4.2.

With exception of the product cylinders, where moderation is used as a control, either optimum moderation or worst case H/U ratio is assumed when performing NCSAs.

Insert 1

RAI ISA-12 Response

Insert 1

, while cold traps use a silicone based oil for a heat transfer medium.

Insert 2

Silicone oil is not included as a potential moderator or reflector because it is bounded by the water reflector considered in the centrifuge plant in the criticality analysis. The hydrogen content is less in silicone based oil than in water. Therefore, the moderator or reflector capabilities of silicone based oil need not be considered in the model.

Markups of Environmental Report

APPENDIX D.1

MAXIMUM ASH THICKNESS DEPOSITED AT THE EREF FROM A CASCADE RANGE ERUPTION

Maximum ash thickness deposited at the EREF from a Cascade Range Eruption

The maximum ash thickness that could be deposited at the EREF from future Cascade tephra eruptions is less than 8 cm, assuming a hypothetical eruption from the nearest Cascade volcano, a maximum credible eruptive volume and explosivity, and the dispersal of ash directly toward the EREF. Therefore, the maximum ash thickness that could be deposited on building roofs at the EREF from future Cascade tephra eruptions is less than 8 cm.

Potential roof loads from a Cascade Range Eruption

Blong (1981, 1984) discusses the effects of ash loading on structures. Snow loading can be used as an analog when making calculations and comparisons. Assume an 8 cm thickness of ash (upper bound; maximum credible ash fall at the site). Assume it has the dry- and wet-ash densities of the Mount St Helens May 18 1980 ash, those being 0.5 g/cc dry and 1.25g /cc wet (Shipley and Sarna-Wojcicki, 1983). The load of the ash (8 cm) would range from 4 g/cm^2 (dry) to 10 g/cm^2 (wet).

Determination of maximum ash thickness deposited at the EREF from a Cascade Range Eruption

The Volcanism Working Group (VWG) (1990; Chapter 5) reviewed published literature, conducted interviews of field researchers in eastern Idaho, and compiled information related to air-fall ash potential at the Idaho National Laboratory (INL), a site immediately west of the proposed Eagle Rock Enrichment Facility. The VWG addressed the 21 most likely sites for ash-fall producing eruptions in the western United States, including all of the Cascade Range volcanoes. The U.S. Geological Survey subsequently published hazard assessments for the five active Cascade volcanoes in Washington, including Mt Adams (Scott, et al., 1995), Mt Baker (Gardner, et al., 1995), Glacier Peak (Waitt, et al., 1995), Mt Rainier (Hoblitt, et al., 1995), Mount St Helens (Wolfe and Pierson, 1995), and the Medicine Lake volcano in northern California (Donnelly-Nolan, et al., 2007). These investigations provide additional detail on eruption probabilities and the distribution of near-vent volcanic products. However, they do not alter the fundamental conclusions of the VWG (1990), which found that prehistoric Cascade tephra deposits on the Eastern Snake River Plain (ESRP) do not exceed 5 cm in thickness. Apparently, the Cascade volcanoes are sufficiently far from the INL (675 to 790 km) or located such that the prevailing westerly winds aloft did not disperse ash directly toward the INL. It was further concluded that the maximum ash thickness that could be deposited at the INL from future Cascade tephra eruptions is less than 8 cm, assuming a hypothetical eruption from the nearest Cascade volcano, a maximum credible eruptive volume and explosivity, and the dispersal of ash directly toward the INL. Because of the proximity of the EREF to INL, the maximum ash thickness that could be deposited at the EREF from future Cascade tephra eruptions is also estimated to be less than 8 cm.

The supporting technical basis for these conclusions is summarized as follows.

1. Empirical curves of compacted air-fall ash thickness vs. distance for the largest late Pleistocene and Holocene Cascade eruptions (Miller, 1989, Figure 3; Hoblitt, et al., 1987, Figure 3-1) include the Mazama (Crater Lake, ca. 8,800 years B.P., 40 km^3); Glacier Peak G (ca. 11,000 – 12,000 years B.P.); Mt St Helens Yn (3,300 – 4,000 years B.P., 3 km^3); and Mt St Helens May 18, 1980 (uncompacted, 1 km^3) tephras. Assuming the most adverse conditions of closest proximity (a hypothetical major eruption of Newberry Volcano, Oregon, 675 km from the INL), ash dispersed directly

toward the EREF, and an eruption magnitude of Mt Mazama 8,800 years B.P. (the largest known eruption from a Cascade volcano), the thickness vs. distance data show that about 6 cm of tephra would be deposited at the EREF.

2. The observed thicknesses of 13 late Pleistocene and Holocene compacted air-fall tephras at field localities on or near the ESRP range from 0.5 to 5 cm, and most are less than 2 cm thick (Volcanism Working Group, 1990; Table 7). Blong (1984) suggests that the initial uncompacted thicknesses of such tephras may have been up to twice as great. These deposits include tephras from the largest eruptions of Cascade volcanoes listed in item 2 above. The observed thicknesses of compacted Mazama ash (the largest known Cascade eruption) at five localities on or near the ESRP range from 0.5 to 3 cm.

REFERENCES

Blong, R.J., 1984, Volcanic Hazards – a Sourcebook on the Effects of Eruptions: Academic Press, New York, 424 p.

Blong, R.J., 1981, Some effects of tephra falls on buildings. In: Tephra Studies, Self, S. and Sparks, R.S.J., editors, Reidel Publishing Co., p. 405-420.

Donnelly-Nolan, J.M., Nathenson, M., Champion D.E., Ramsey, D.W., Lowenstern, J.B., and Ewert, J.W., 2007, Volcano hazards assessment for Medicine Lake Volcano, northern California: U.S. Geological Survey Scientific Investigations Report 2007–5174-A, 33 p., 1 plate.

Hoblitt, R.P, Miller, C.D., and Scott, W.E., 1987, Volcanic hazards with regard to siting nuclear-power plants in the Pacific Northwest: U.S. Geological Survey Open-File Report 87-297, 196 p.

Hoblitt, R.P., Wilder, J.S., Driedger, C.L., Scott, K.M., Pringle, P.T., and Vallance, J.W., 1995, Volcano hazards from Mount Rainier, Washington: U.S. Geological Survey Open-File Report 95-273, 10 p., 1 plate.

Miller, C.D., 1989, Potential hazards from future volcanic eruptions in California: U.S. Geological Survey Bulletin 1847, 17 p., 2 plates.

Scott 1995, Volcano hazards in the Mount Adams region, Washington: U.S. Geological Survey Open-File Report 95-492, 11 p., plates.

Shipley, S., and Sarna-Wojcicki, A.M., 1983, Distribution, thickness and mass of tephra from volcanoes – Pacific Northwest United States: Assessment of hazards to nuclear reactors. U.S. Geological Survey Misc Field Studies Map MF-1435, 27 p.

Volcanism Working Group, 1990, Assessment of potential volcanic hazards for New Production Reactor site at the Idaho National Engineering Laboratory: EG&G Informal Report, EGG-NPR-10624, 98 p. [Original document identifier is UCRL-ID-104722, Technical Information Department, Lawrence Livermore National Laboratory, University of California, Livermore, CA 94551.]

Waitt, R.B., Mastin, Larry, and Beget, J.E., 1995, Volcanic-hazard zonation for Glacier Peak Volcano, Washington: U.S. Geological Survey Open-File Report 95-499, 9 p., 2 plates.

Warrick, R.A., Anderson, J., Dowing, T., Lyons, J., Ressler, J., Warrick, M., and Warrick, T., 1981, Four communities under ash, after Mount St Helens: Program on Tech Envir and Man, Monograph 34, Instutute of Behavioral Science, University of Colorado, Boulder, CO, 143 p.

Wolfe, E.W., and Pierson, T.C., 1995, Volcanic-hazard zonation for Mount St Helens, Washington: U.S. Geological Survey Open-File Report 95-497, 12 p., 2 plates.

The site-specific PSHA results have smaller ground motion amplitudes than ground motion determined for the 2008 update of the USGS national hazard maps. USGS PGA estimates are 30% higher at 10^{-3} per year and 40% higher at 10^{-5} per year than amplitudes determined in the site-specific PSHA. The difference in seismic hazard estimates ~~can result~~ resulted from the following ~~possible~~ causes.

- The site-specific PSHA used ground motion models for normal slip fault mechanisms; the USGS ~~possibly~~ used various fault mechanisms, or unspecified fault mechanisms, which predict higher amplitude seismic ground motions.
- The weighted result for the site-specific PSHA includes hazard results for hard rock attenuation models, which leads to lower amplitude seismic ground motions. The USGS 2008 results are for the NEHRP B-C Boundary site condition which is a firm rock condition that results in higher amplitude seismic ground motions relative to hard rock site conditions.
- The site-specific PSHA used a local earthquake frequency model determined for the ESRP; the USGS ~~possibly~~ used a larger background seismicity model for the Basin and Range province and a local cell earthquake activity rate that could exceed the historical earthquake rate (Petersen et al., 2008).

Refer to Appendix G for documentation that supports these conclusions.

Site-specific UHRS are compared to USGS 2008 seismic hazard results on **Figure 27** for annual probability of exceedance of 10^{-3}. The site-specific and USGS 2008 UHRS are comparable at shorter periods from 0.03 to 0.2 seconds (30 Hz to 5 Hz), with USGS results being higher. USGS results become increasingly higher than the site specific results at longer periods. This difference is attributed to usage of hard rock attenuation models in the site-specific PSHA, whereas the USGS 2008 hazard estimates are for a firm bedrock site condition.

Figure 27 provides a comparison of UHRS determined for the INTEC Facility at INL (Payne, 2002) and the site-specific UHRS for exceedance probabilities of 10^{-3} per year and 10^{-4} per year. The INTEC UHRS overlie the site-specific UHRS and have a different spectral shape which is illustrated by a flatter response spectrum. Spectral amplitudes are similar for periods between 0.1 to 0.2 seconds. At lower and higher periods the INTEC UHRS overlie the site-specific UHRS. The difference in response spectra is attributed to the closer proximity of the INTEC facility to the Basin and Range seismic sources and Quaternary Faults located west of INL. The EREF site is located about 48 km (30 mi) east of the INTEC facility at INL. Ground motion models developed for extensional tectonic regimes and used in the site-specific PSHA predict a high rate of amplitude attenuation over relatively short distances for earthquakes associated with normal fault movements.

Appendix G

1.0 Introduction and Summary

This report provides additional information and analyses to support the conclusions in the site-specific probabilistic seismic hazard assessment (PSHA). The analyses demonstrate that the site-specific PSHA results are lower than the new hazard results published by the USGS. Newly released ground motion information by the USGS, made available after the license application submittal, is incorporated in these analyses.

2.0 Background

Probabilistic seismic hazard is determined for a specific site location using a complex analysis of the three basic input models listed below.

1. Geometries of regional seismic sources including area sources and fault sources that describe the geographic locales of seismic activity.
2. Annual earthquake recurrence rates as a function of magnitude for seismic source zones and faults defined in Item 1.
3. Ground motion prediction equations that describe ground motion amplitudes as functions of frequency given occurrence of an earthquake of a specific magnitude and located at a specific distance from a site underlain with a known surface geologic condition (e.g. soil, bedrock, or known V_{S30} value).

The probabilistic seismic hazard assessment is conducted using information that is affected by two principal types of uncertainty. These include *epistemic* uncertainty that results from incomplete knowledge or understanding of geologic and seismologic processes in the specific geographic region being studied, and *aleatory* uncertainty that results from random processes. *Epistemic* uncertainties are accommodated in the probabilistic seismic hazard assessment by using multiple hypotheses to represent the range of credible models for Items 1 through 3, above. *Aleatory* uncertainty, or randomness, in ground motion predictions is accommodated by specifying the median ground motion prediction as well as the standard error (σ "sigma") determined for each ground motion prediction equation. In addition, random uncertainty is accommodated by specifying the number of standard errors (e.g. 1, 2, 3, or unbounded) that will be analyzed in the seismic hazard calculations.

Parametric variations of each of the three basic input models are analyzed to yield a number of individual seismic hazard assessments (i.e. seismic hazard curves), each of which is the result of a specific combination of varied input models. A final hazard assessment can be obtained by using statistical processing of the individual seismic hazard curves to determine mean or median hazard, or hazard at a higher statistical fractile (e.g. 85th percentile). Also, a final hazard assessment can be obtained using a logic-tree system in which each individual parametric variation is assigned a weight and application of the weights yields the final seismic hazard assessment.

Results of the site-specific seismic hazard assessment performed for the EREF site were compared to new seismic hazard estimates released by the US Geological Survey in Open File Report 2008-1128 (Petersen et al., 2008). Information included in OF 2008-1128 (Reference 1) was used to qualitatively describe several reasons why the site-specific hazard results were 30% to 40% lower than new hazard results published by the USGS.

Following are the qualitative reasons that were cited in the site-specific study that were listed as possible causes for the seismic hazard differences.

"The difference in seismic hazard estimates can result from the following possible causes."

- The site-specific PSHA used ground motion models for normal slip fault mechanisms; the USGS possibly used various fault mechanisms, or unspecified fault mechanisms, which predict higher amplitude seismic ground motions.
- The weighted result for the site-specific PSHA includes hazard results for hard bedrock attenuation models, which leads to lower amplitude seismic ground motions. The USGS 2008 results are for the NEHRP B-C Boundary site condition which is a firm bedrock condition that results in higher amplitude seismic ground motions relative to hard bedrock site conditions.
- The site-specific PSHA used a local earthquake frequency model determined for the ESRP; the USGS possibly used a larger background seismicity model for the Basin and Range province and a local cell earthquake activity rate that could exceed the historical earthquake rate (Petersen et al., 2008).

3.0 <u>Verification of Site-Specific Ground Motion Amplitudes</u>

Previously unavailable information has been recently obtained. The new information includes gridded earthquake activity rates that were only recently posted to the USGS web page dedicated to the 2008 update of the National Seismic Hazard Maps and is provided in this response to further support the three bulleted explanations above for the USGS (2008) hazard estimates being higher than that determined in the site-specific probabilistic seismic hazard assessment.

1. <u>Ground Motion Prediction Equations</u>

The site-specific probabilistic seismic hazard study for the EREF site used ground motion prediction equations published by Spudich et al. (1999) and by Boore and Atkinson (2008). Each of these equations was applied to predict ground motion for earthquake activity associated with normal faulting. The EREF site is located within the intermountain west tectonic zone that is characterized by extensional tectonism with seismic activity being the result of crustal movements predominantly on normal faults.

Spudich et al. (1999, p. 1161) conclude that "extensional regime ground motions are systematically smaller than non-extensional regime motions." Their equations are applicable only to normal faulting tectonic settings. Boore and Atkinson (2008) provide equations to predict ground motions based on fault mechanism. Following are comparisons of ground motion amplitudes for normal vs. strike-slip faulting using equations from Boore and Atkinson (2008).

Magnitude, M_W	Distance, km	Fault Type	Peak Ground Acceleration, g	Spectral Acceleration, g, 5 Hz
7.0	100.0	Normal	0.023	0.055
7.0	100.0	Strike-Slip	0.030	0.066

Boore and Atkinson (2008) illustrate that ground motions are increased by 30% for peak ground acceleration and by 20% for spectral acceleration at 5 Hz for strike-slip vs. normal faults for a magnitude 7 earthquake located at a distance of 100 km.

The USGS used three equally-weighted ground motion prediction equations for their 2008 update of hazard maps for the extensional tectonic region of the Western United States. These include Boore and Atkinson (2008), Campbell and Bozorgnia (2008), and Chiou and Youngs (2008). Petersen et al. (2008, Figure 12.) illustrate that each of the three ground motion equations was applied with equal weight of 0.5 for strike-slip and normal fault motion predictions. The USGS (2008) hazard assessment covers a much broader geographic region of the WUS than the site-specific study performed for the EREF site. This larger region was modeled to include ground motion resulting equally from normal and strike-slip fault movements. USGS (2008) hazard estimates thus are higher than those determined in the site-specific study due to the inclusion of ground motions predicted for strike-slip faults, which produce higher ground motions. Results in the table above for the Boore and Atkinson (2008) models illustrate that ground motion could increase by 20% to 30% if ground motion was attributed exclusively to strike-slip faults. Ground motion variations by fault type attributed to the other two USGS (2008) ground motion models were not determined, but are believed to also predict lower ground motions for normal faulting earthquakes relative to strike-slip faulting. It is therefore estimated that the increase in USGS (2008) seismic hazard relative to the EREF site-specific study, attributed to usage of both strike-slip and normal faults, is about 10% to 15% based on examination of predictions by the Boore and Atkinson (2008) models.

2. Site Conditions for Ground Motion Predictions

Appendix G

The site-specific seismic hazard assessment was determined for site conditions in the upper 30 meters of the earth's crust (i.e. V_{S30}) that included 620 m/sec (2,034 ft/sec) for the Spudich et al. (1999) models, and that ranged from 760 m/sec (2,493 ft/sec) to 1,300 m/sec (4,265 ft/sec) for the Boore and Atkinson (2008) ground motion models. This range of site conditions brackets the one site condition used by the USGS (2008) hazard assessment; that being the NEHRP Site Class B-C Boundary characterized by a V_{S30} of 760 m/sec. It was estimated in the site-specific study using shear wave velocity measurements made at the INL that the actual V_{S30} site condition at EREF was closer to the 1,300 m/sec than to the USGS (2008) B-C site condition of 760 m/sec. The USGS converts all ground motion models that make hard bedrock predictions (e.g. Site Class A, V_{S30} of 1,500 m/sec) to the B-C Boundary site condition using the following factors (Petersen et al., 2008, p. 18).

> *"For several of these models, we used frequency-dependent factors to convert from hard rock (NEHRP site class A) to firm rock (NEHRP site class BC). These factors are: 1.52 for peak ground acceleration, 1.74 for 0.1-s, 1.76 for 0.2-s, 1.72 for 0.3-s, 1.58 for 0.5-s, 1.34 for 1.0-s, and 1.20 for 2.0-s spectral acceleration (see Frankel and others, 1996)."*

Ground motion amplitudes at hard rock sites range from about 60% to 80% of amplitudes predicted for Site Class B-C. Seismic hazard results in the site-specific study were weighted as 75% for Site Class B-C and 25% to a hard bedrock site condition just below the Site Class A condition (e.g. 1,300 m/sec vs. 1,500 m/sec). This weighting used in the site-specific study would slightly decrease the hazard estimate relative to the USGS (2008) which is based on only the lower velocity (V_{S30} of 760 m/sec) B-C Boundary Site Class.

3. <u>Earthquake Activity Rates</u>

Differences in the seismic hazard between the EREF site-specific study and the USGS (2008) study were attributed to possible differences in earthquake activity rates determined for the regions surrounding the EREF site. USGS (2008) input files (include 10^a grids) were not available until recently to make a comparison with activity rates applied in the USGS (2008) hazard assessments. The input files are provided in the USGS website:
http://earthquake.usgs.gov/research/hazmaps/products_data/2008/software/

These technical input data files defined the geometries of the USGS (2008) background sources for the Eastern Snake River Plain and the Yellowstone Parabola, and the gridded and smoothed earthquake activity rates for the region. Petersen et al. (2008, p.21) describe the method for determining activity rates for "Uniform Background Zones" which include the Eastern Snake River Plain and the Yellowstone Parabola (see below). The earthquake activity rates are incremental per grid cell per year, for a magnitude increment of 0.1 mag unit (so 10^a is the rate of earthquakes with magnitude between -

0.05 and 0.05) and a grid cell size of 0.1 x 0.1 degree (so the area of the grid cell varies with latitude). A uniform b value of 0.8 was applied for this region.

Uniform Background Zones (Petersen et al. 2008, p.21)

In contrast to the gridded-(smoothed-) seismicity model, regional background zones account for earthquake potential spread uniformly across tectonic or geologic regions with constant geologic or strain characteristics. These zones are designed to provide a hazard floor and account for future random earthquakes in areas with little or no historical seismicity. The earthquake rate for each WUS background zone is determined by counting earthquakes with $M_w \geq 4$ since 1963, computing an annualized rate, and prorating this rate uniformly across the entire zone. As in the 1996 and 2002 maps, we model background seismicity (WUS Model 2) in five non-overlapping regional zones: the Basin and Range Province extended to include the Rio Grande rift, parts of Arizona and New Mexico, western Texas, eastern Washington, and northern Montana and Idaho; the Cascade volcanic province; the Snake River Plain province; the Yellowstone seismicity parabola province; and a region of southeastern California and southwestern Arizona (figs. 14 and 15). These regions are geologically and seismologically distinct, and the reasoning behind the zonation is discussed in detail in the 1996 documentation. As in 1996 and 2002, the regional zone model (Model 2) is implemented in a way that does not penalize areas of high seismicity in order to provide a hazard floor in areas of low seismicity. In each grid cell, the historical gridded-seismicity rate (Model 1) is compared with the floor value from Model 2. If the historical rate exceeds the floor value, the final cell rate simply equals the historical rate. If, however, the floor value exceeds the historical rate, Models 1 and 2 are combined with respective weights 0.67 and 0.33 to calculate the final cell rate. Nowhere is the final cell rate less than the historical rate, and the total modeled seismicity rate in the WUS exceeds the total historical rate by about 16 percent.

The information described above and from OF 2008-1128 (Reference 1) was synthesized into a regional map in order to make a comparison of earthquake activity rates applied in the EREF site-specific study and in the USGS (2008) seismic hazard update for the Western United States.

Figure 1 (Attachment 1) shows a map prepared using the following USGS (2008, 2009) information.

1. USGS (2008) WUS Earthquake Catalog, $M_W > 4$, duplicates, clustered events eliminated
2. USGS (2009) Gridded a-values, file agrd.out.txt
3. USGS (2009) Coordinates for Eastern Snake River Plain
4. USGS (2009) Coordinates for the Yellowstone Parabola

Appendix G

Figure 1 (Attachment 1) includes a contour map calculated from the data file (Item 2.) of USGS (2008) gridded a-values. The agrd.out.txt includes earthquake activity rates for cell sizes of 0.1^{O} Latitude and 0.1^{O} Longitude. The rate of earthquakes of magnitude -0.5 to 0.5 centered at magnitude 0 is determined as 10^{agrd}. The rate of earthquakes at higher magnitudes is determined using the uniform b-value of 0.8 used in USGS (2008) throughout the region shown on Figure SP-8-1. Figure 2 (Attachment 2) shows the gridded a-value locations with annotated a-values. Gridded a-values for the EREF site region range from 0.034 to 0.058 with a value of 0.0406 interpolated for the EREF site location. The grid cell area for the EREF site region is 89.7 sq. km. Figure SP-8-2 illustrates that the gridded a-values range from a low value of 0.015 about 40 km northwest of the EREF site near the western boundary of the ESRP with the Yellowstone Parabola zone to a high value of 0.182 about 40 km southeast of the EREF site near the contact with the eastern branch of the Yellowstone Parabola.

Earthquake activity rates predicted for the Eastern Snake River Plain by the range of USGS (2008) gridded a-values and associated uniform b-value of 0.8 are compared to earthquake activity rates determined for the ESRP in the site-specific hazard study. The calculation procedure used for this comparison is to normalize gridded a-values to the entire area of the ESRP and to compare predicted annual rates of magnitude 4.0 and larger earthquakes, and also for magnitude 5.0 and larger earthquakes. Calculation sheets are provided in Attachment 3.

The following table summarizes calculations of earthquake activity rates for the ESRP provided in Attachment 3.

Model	a-value 33,264 km^2	b-value	Annual Rate Mw ≥ 4.0	Annual Rate Mw ≥ 5.0	Return Period Mw ≥ 4.0	Return Period Mw ≥ 5.0
Site-Specific	2.931	0.945	0.142	0.016	7.1	62.2
USGS (2008) min a-value	2.584	0.80	0.242	0.038	4.1	26.1
USGS (2008) value at EREF	2.61	0.80	0.257	0.041	3.9	24.6
USGS (2008) max a-value	2.751	0.80	0.356	0.056	2.8	17.7

Earthquake activity rates compared in the above table for the ESRP illustrate that USGS (2008) activity rates for magnitude 4.0 and larger earthquakes are greater than predictions made by the site-specific model by factors that range from 1.7 (minimum gridded a-values) to 2.5 (maximum gridded a-values). Differences increase for larger magnitudes due to the less-steep b-value of 0.8 applied for this region by the USGS.

This difference in local activity rates for the seismic zone containing the EREF site is a direct cause for the higher ground motion estimates determined by the USGS (2008) hazard assessment.

Predictions of earthquake activity rates by the site-specific and USGS (2008) recurrence models are compared to the Western US earthquake catalog (Petersen et al., 2008). Figure SP-8-1 illustrates 2 earthquakes of magnitude 4 and larger located within the ESRP. These include the M=5.5 Shoshone earthquake located at the southwest end of the ESRP, and an earthquake of magnitude 4.1 that occurred in 1964 near the eastern boundary with the Yellowstone Parabola zone. Both the site-specific and USGS (2008) recurrence models overestimate the number of earthquakes actually observed in the ESRP. For example, the USGS (2008) models estimate the return period of magnitude 4 and larger earthquakes to be in the range of 3 to 4 years; therefore about 11 to 15 earthquakes of this size would be expected to have occurred in the 45-year duration of the earthquake catalog. The site-specific recurrence model predicts about 6 occurrences of magnitude 4 and larger earthquakes since 1963. Similarly, the USGS (2008) models predict about 4 or 5 occurrences of magnitude 5 and larger earthquakes during a 100-year time frame, compared to 1 to 2 for the site-specific model. The only known earthquake of this size attributed to the ESRP is the 1905 historical Shoshone earthquake with uncertain location, and possibly not located in the ESRP (INL, 2008).

Comparisons shown above indicate conservative representations of seismic activity rates specified for the ESRP zone which is characterized by a very low seismicity rate. The USGS (2008) earthquake activity rate estimates are substantially more conservative than the site-specific model, most likely due to their procedure of combining gridded and smoothed historical activity rates as described in the Uniform Background Zones section of Petersen et al. (2008) presented at the top of this response.

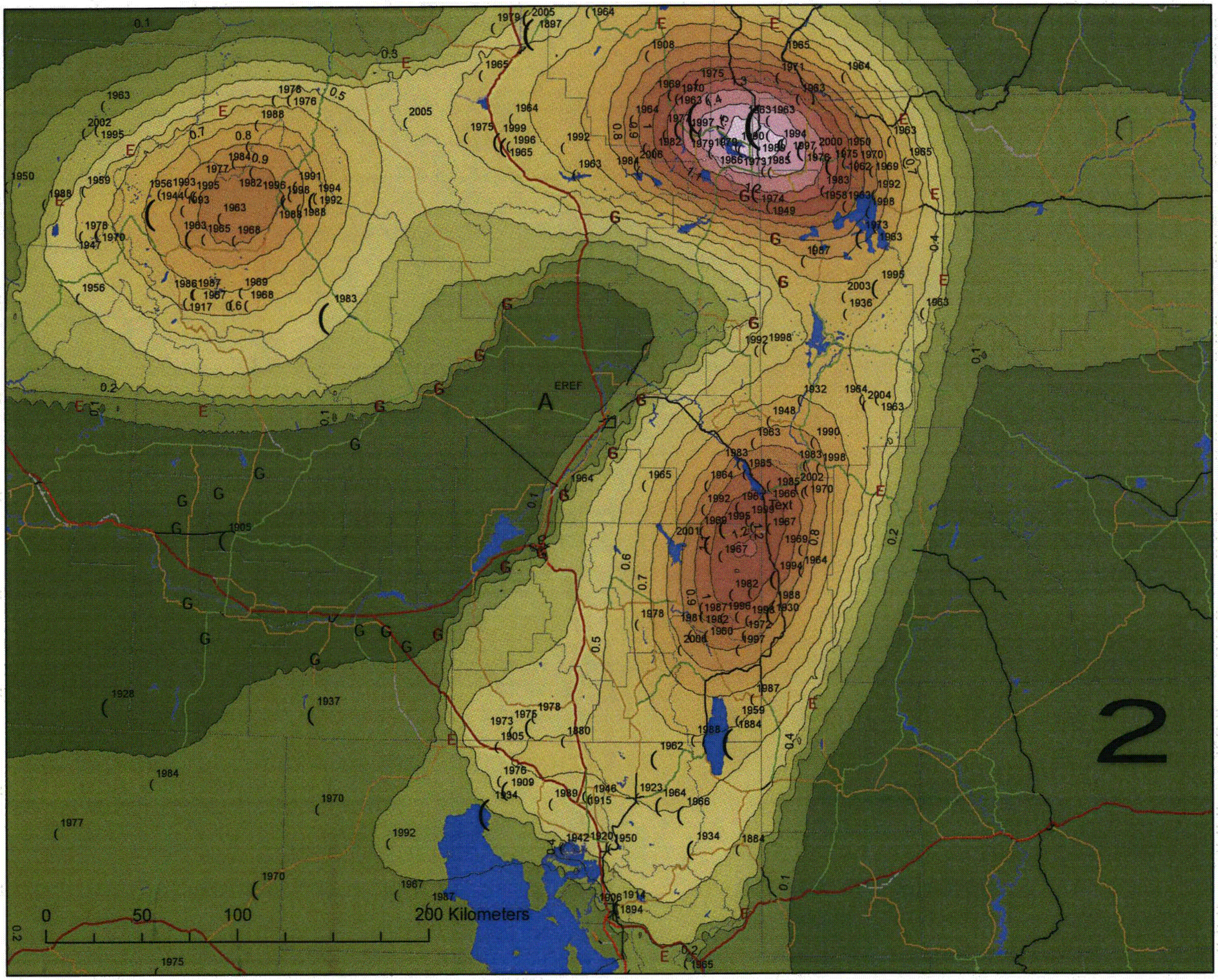
REFERENCES:

1. Petersen, Mark D., Frankel, Arthur D., Harmsen, Stephen C., Mueller, Charles S., Haller, Kathleen M., Wheeler, Russell L., Wesson, Robert L., Zeng, Yuehua, Boyd, Oliver S., Perkins, David M., Luco, Nicolas, Field, Edward H., Wills, Chris J., and Rukstales, Kenneth S., 2008, Documentation for the 2008 Update of the United States National Seismic Hazard Maps: U.S. Geological Survey Open-File Report 2008–1128, 61 p.

Appendix G

ATTACHMENT 1

USGS Compiled Data Contour Map



Legend

A EREF Site

USGS WUS Catalog

Mw

- (4.0 - 5.0
- (5.01 - 6.0
- (6.01 - 7.0
- (7.01 - 8.0

E Yellowstone Parabola Coordinates

G E. Snake River Plain Coordinates

Gridded a-value

USGS (2008)

- 0 - 0.1
- 0.1 - 0.2
- 0.2 - 0.3
- 0.3 - 0.4
- 0.4 - 0.5
- 0.5 - 0.6
- 0.6 - 0.7
- 0.7 - 0.8
- 0.8 - 0.9
- 0.9 - 1
- 1.01 - 1.1
- 1.11 - 1.2
- 1.21 - 1.3
- 1.31 - 1.4
- 1.41 - 1.5
- 1.51 - 1.6

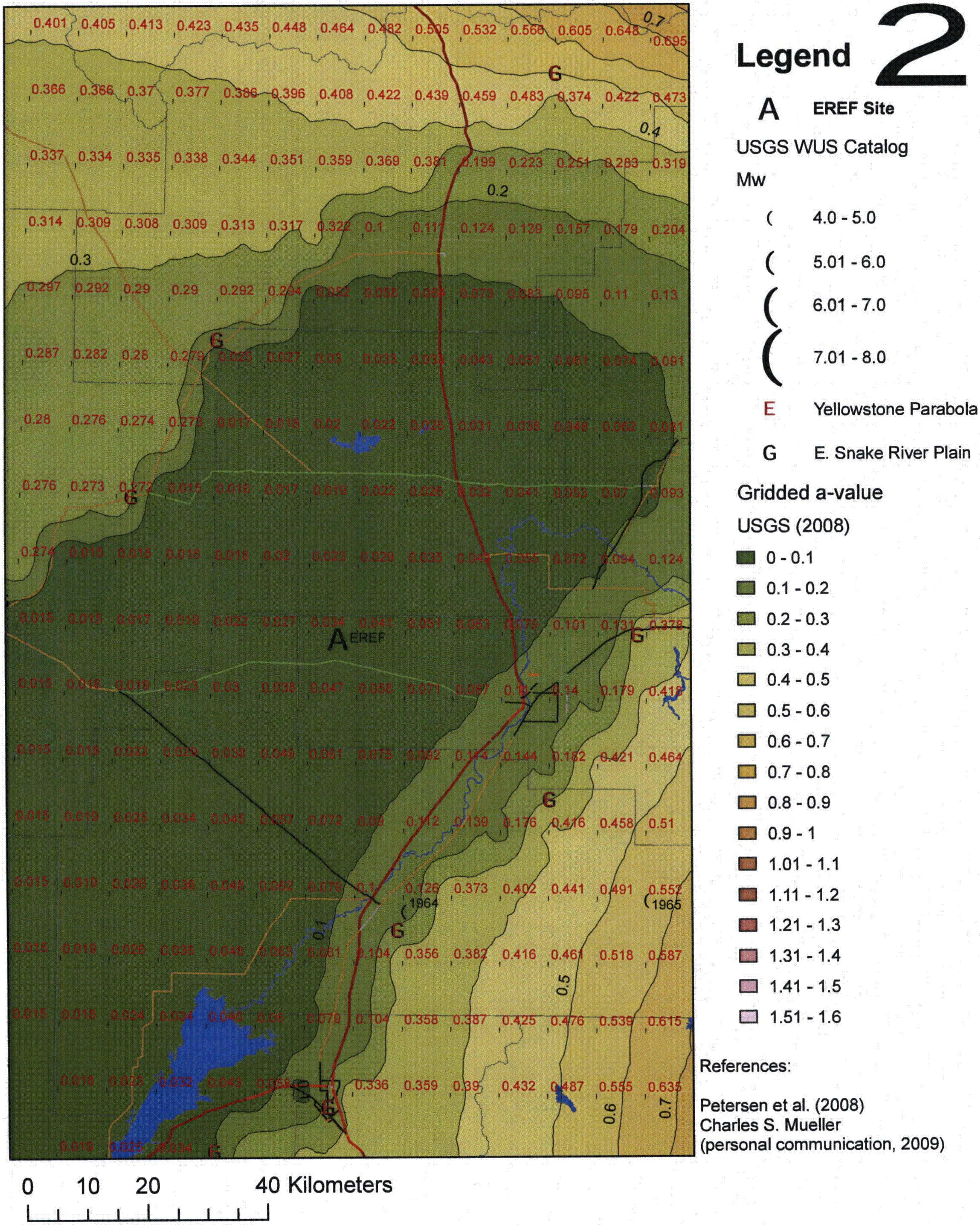
References:

Petersen et al. (2008)
Charles S. Mueller (personal communication, 2009)

Appendix G

ATTACHMENT 2

USGS Compiled Data Grid Map


Legend
2
A EREF Site
USGS WUS Catalog
Mw
4.0 - 5.0
5.01 - 6.0
6.01 - 7.0
7.01 - 8.0
E Yellowstone Parabola
G E. Snake River Plain
Gridded a-value
USGS (2008)
0 - 0.1
0.1 - 0.2
0.2 - 0.3
0.3 - 0.4
0.4 - 0.5
0.5 - 0.6
0.6 - 0.7
0.7 - 0.8
0.8 - 0.9
0.9 - 1
1.01 - 1.1
1.11 - 1.2
1.21 - 1.3
1.31 - 1.4
1.41 - 1.5
1.51 - 1.6
References:
Petersen et al. (2008)
Charles S. Mueller
(personal communication, 2009)
EREF
1964
1965
0 10 20 40 Kilometers

Appendix G

ATTACHMENT 3

ESRP Earthquake Activity Rate Calculations

Site-Specific PSHA earthquake activity rates for the Eastern Snake River Plain

$a1 := 2.931$ Site-specific a-value for ESRP

$b := 0.945$ Site-specific b-value for ESRP

$A1 := 33264$ Area for site-specific a-value

$A2 := 33264$ Total area of ESRP, sq. km.

$Mw_0 := 4.0$

$Mw_1 := 5.0$

$$a2 := \log\left(\frac{10^{a1}\cdot A2}{A1}\right)$$ Eq. 1 to determine a-value for change in area

$a2 = 2.931$ a-value for total area of ESRP

$LogNc := a2 - b\cdot Mw$ Recurrence model for total area of ESRP

$$LogNc = \begin{pmatrix} -0.849 \\ -1.794 \end{pmatrix}$$

$$Nc := 10^{LogNc}$$

$$Nc = \begin{pmatrix} 0.142 \\ 0.016 \end{pmatrix}$$ Annual rates of magnitude 4.0 and larger, and magnitude 5.0 and larger earthquakes

$$RPy := \frac{1.}{Nc}$$

$$RPy = \begin{pmatrix} 7.063 \\ 62.23 \end{pmatrix}$$ Return periods (yrs) of magnitude 4.0 and larger, and magnitude 5.0 and larger earthquakes

USGS (2008) earthquake activity rates for the Eastern Snake River Plain

$a1 := 0.015$ USGS (2008) gridded a-value, minimum for ESRP

$b := 0.8$ USGS (2008) b-value for ESRP

$A1 := 89.7$ Area of 0.1 deg Lon by 0.1 deg Lat grid cell, sq. km.

$A2 := 33264$ Total area of ESRP, sq. km.

$Mw_0 := 4.0$

$Mw_1 := 5.0$

$$a2 := \log\left(\frac{10^{a1} \cdot A2}{A1}\right)$$ Eq. 1 to determine a-value for change in area

$a2 = 2.584$ a-value for total area of ESRP

$LogNc := a2 - b \cdot Mw$ Recurrence model for total area of ESRP

$$LogNc = \begin{pmatrix} -0.616 \\ -1.416 \end{pmatrix}$$

$$Nc := 10^{LogNc}$$

$$Nc = \begin{pmatrix} 0.242 \\ 0.038 \end{pmatrix}$$ Annual rates of magnitude 4.0 and larger, and magnitude 5.0 and larger earthquakes

$$RPy := \frac{1.}{Nc}$$

$$RPy = \begin{pmatrix} 4.129 \\ 26.051 \end{pmatrix}$$ Return periods (yrs) of magnitude 4.0 and larger, and magnitude 5.0 and larger earthquakes

USGS (2008) earthquake activity rates for the Eastern Snake River Plain

$a1 := 0.0406$ USGS (2008) gridded a-value, interpolated value at EREF site

$b := 0.8$ USGS (2008) b-value for ESRP

$A1 := 89.7$ Area of 0.1 deg Lon by 0.1 deg Lat grid cell, sq. km.

$A2 := 33264$ Total area of ESRP, sq. km.

$Mw_0 := 4.0$

$Mw_1 := 5.0$

$$a2 := \log\left(\frac{10^{a1} \cdot A2}{A1}\right)$$ Eq. 1 to determine a-value for change in area

$a2 = 2.61$ a-value for total area of ESRP

$LogNc := a2 - b \cdot Mw$ Recurrence model for total area of ESRP

$$LogNc = \begin{pmatrix} -0.59 \\ -1.39 \end{pmatrix}$$

$$Nc := 10^{LogNc}$$

$$Nc = \begin{pmatrix} 0.257 \\ 0.041 \end{pmatrix}$$ Annual rates of magnitude 4.0 and larger, and magnitude 5.0 and larger earthquakes

$$RPy := \frac{1.}{Nc}$$

$$RPy = \begin{pmatrix} 3.892 \\ 24.559 \end{pmatrix}$$ Return periods (yrs) of magnitude 4.0 and larger, and magnitude 5.0 and larger earthquakes

USGS (2008) earthquake activity rates for the Eastern Snake River Plain

$a1 := 0.182$ USGS (2008) gridded a-value, maximum value at nearest zone boundary

$b := 0.8$ USGS (2008) b-value for ESRP

$A1 := 89.7$ Area of 0.1 deg Lon by 0.1 deg Lat grid cell, sq. km.

$A2 := 33264$ Total area of ESRP, sq. km.

$Mw_0 := 4.0$

$Mw_1 := 5.0$

$$a2 := \log\left(\frac{10^{a1} \cdot A2}{A1}\right)$$ Eq. 1 to determine a-value for change in area

$a2 = 2.751$ a-value for total area of ESRP

$LogNc := a2 - b \cdot Mw$ Recurrence model for total area of ESRP

$$LogNc = \begin{pmatrix} -0.449 \\ -1.249 \end{pmatrix}$$

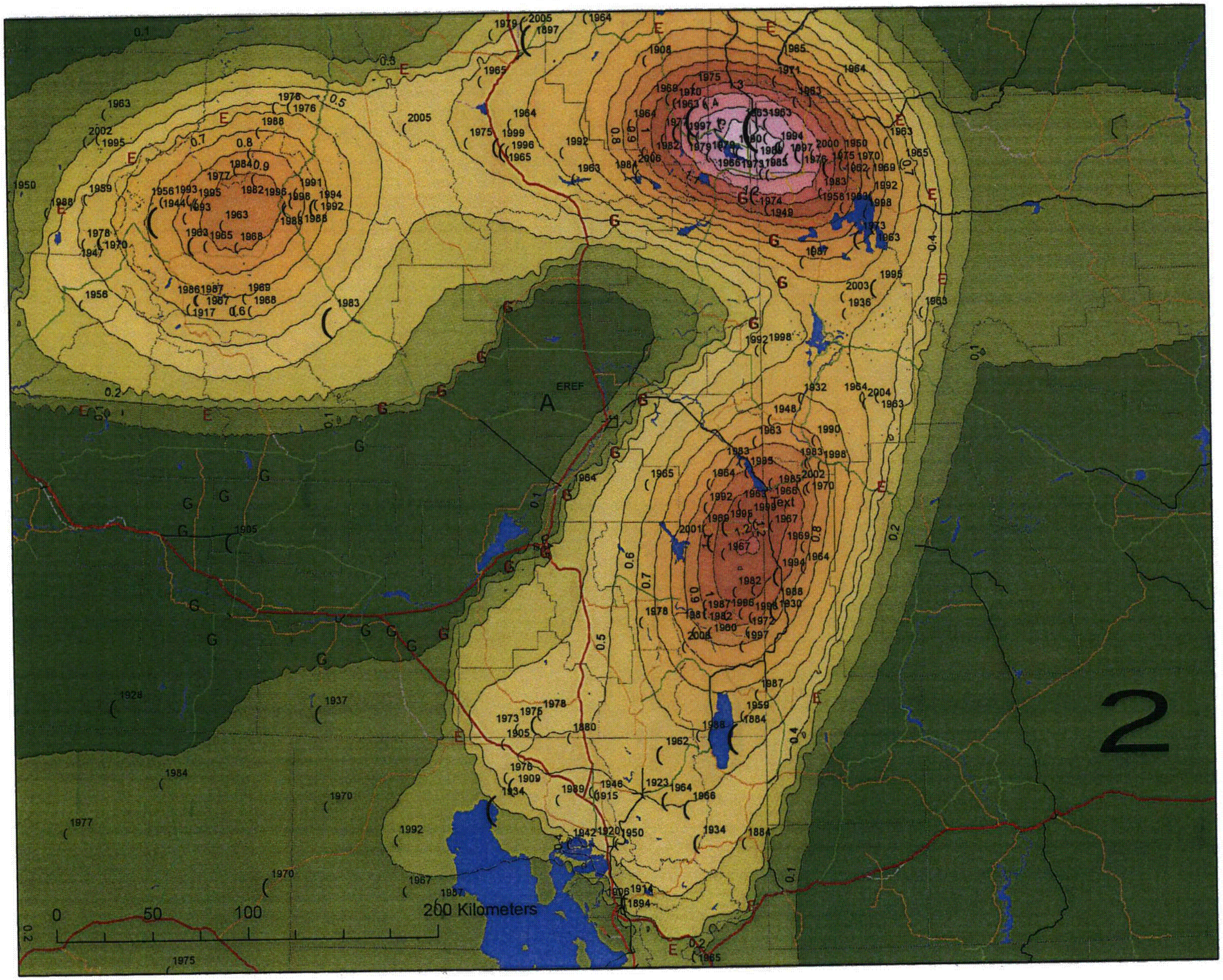
$$Nc := 10^{LogNc}$$

$$Nc = \begin{pmatrix} 0.356 \\ 0.056 \end{pmatrix}$$ Annual rates of magnitude 4.0 and larger, and magnitude 5.0 and larger earthquakes

$$RPy := \frac{1.}{Nc}$$

$$RPy = \begin{pmatrix} 2.811 \\ 17.734 \end{pmatrix}$$ Return periods (yrs) of magnitude 4.0 and larger, and magnitude 5.0 and larger earthquakes

Enclosure 2.4
USGS Compiled Data Contour Map
USGS Compile Data Grid Map
ESRP Earthquake Activity Rate Calculations



Legend

A EREF Site

USGS WUS Catalog

Mw

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E Yellowstone Parabola Coordinates

G E. Snake River Plain Coordinates

Gridded a-value

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- 0.3 - 0.4
- 0.4 - 0.5
- 0.5 - 0.6
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- 0.7 - 0.8
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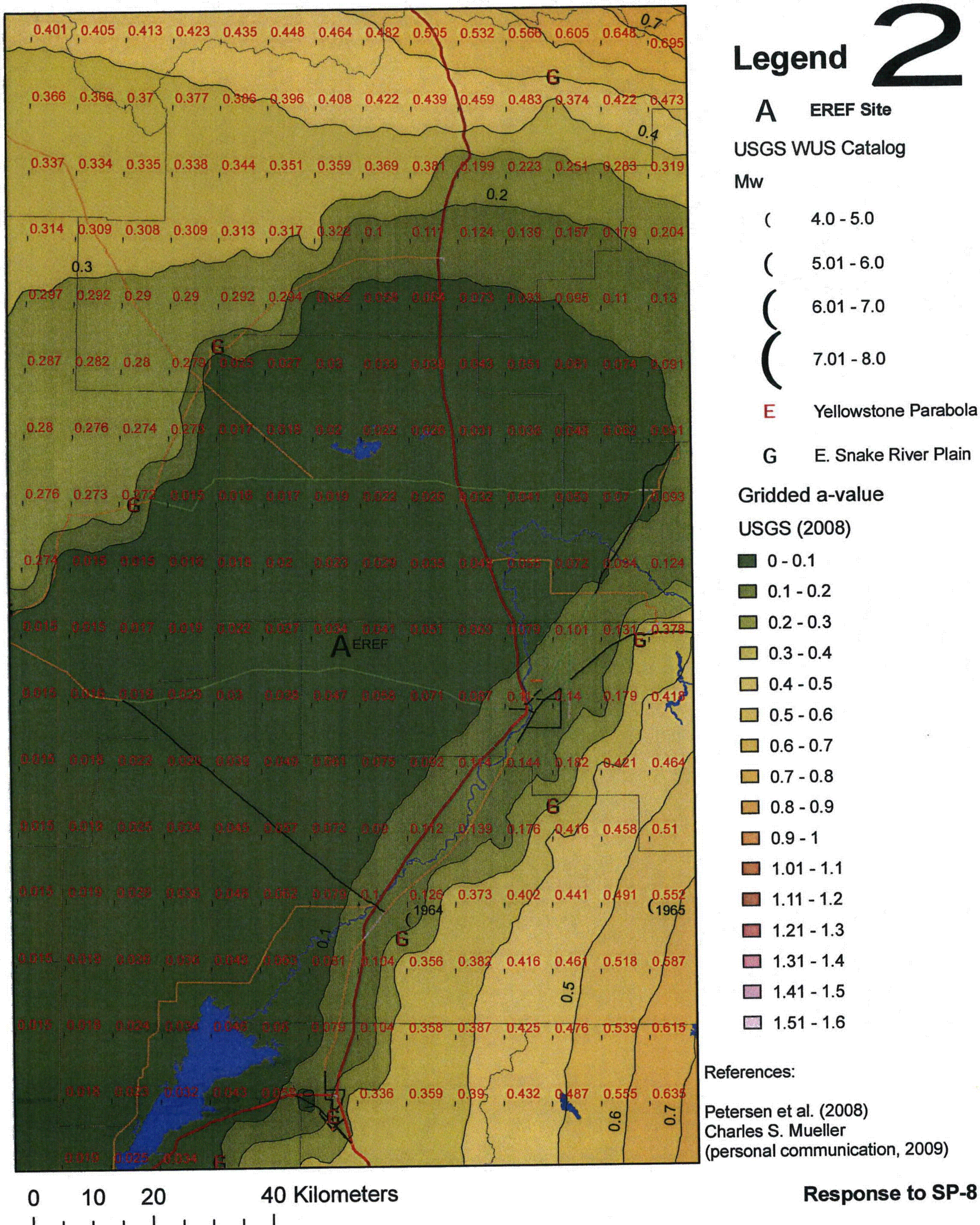
- 1.11 - 1.2
- 1.21 - 1.3
- 1.31 - 1.4
- 1.41 - 1.5
- 1.51 - 1.6

References:

Petersen et al. (2008)
Charles S. Mueller (personal communication, 2009)

Response to SP-8

FIGURE SP-8-1



Response to SP-8

FIGURE SP-8-2

Site-Specific PSHA earthquake activity rates for the Eastern Snake River Plain

$a1 := 2.931$	Site-specific a-value for ESRP
$b := 0.945$	Site-specific b-value for ESRP
$A1 := 33264$	Area for site-specific a-value
$A2 := 33264$	Total area of ESRP, sq. km.

$Mw_0 := 4.0$

$Mw_1 := 5.0$

$$a2 := \log\left(\frac{10^{a1} \cdot A2}{A1}\right)$$ Eq. 1 to determine a-value for change in area

$a2 = 2.931$ a-value for total area of ESRP

$LogNc := a2 - b \cdot \vec{Mw}$ Recurrence model for total area of ESRP

$$LogNc = \begin{pmatrix} -0.849 \\ -1.794 \end{pmatrix}$$

$$Nc := 10^{LogNc}$$

$$Nc = \begin{pmatrix} 0.142 \\ 0.016 \end{pmatrix}$$ Annual rates of magnitude 4.0 and larger, and magnitude 5.0 and larger earthquakes

$$RPy := \frac{1.}{Nc}$$

$$RPy = \begin{pmatrix} 7.063 \\ 62.23 \end{pmatrix}$$ Return periods (yrs) of magnitude 4.0 and larger, and magnitude 5.0 and larger earthquakes

USGS (2008) earthquake activity rates for the Eastern Snake River Plain

$a1 := 0.015$ USGS (2008) gridded a-value, minimum for ESRP

$b := 0.8$ USGS (2008) b-value for ESRP

$A1 := 89.7$ Area of 0.1 deg Lon by 0.1 deg Lat grid cell, sq. km.

$A2 := 33264$ Total area of ESRP, sq. km.

$Mw_0 := 4.0$

$Mw_1 := 5.0$

$$a2 := \log\left(\frac{10^{a1} \cdot A2}{A1}\right)$$ Eq. 1 to determine a-value for change in area

$a2 = 2.584$ a-value for total area of ESRP

$LogNc := a2 - b \cdot Mw$ Recurrence model for total area of ESRP

$$LogNc = \begin{pmatrix} -0.616 \\ -1.416 \end{pmatrix}$$

$$Nc := 10^{LogNc}$$

$$Nc = \begin{pmatrix} 0.242 \\ 0.038 \end{pmatrix}$$ Annual rates of magnitude 4.0 and larger, and magnitude 5.0 and larger earthquakes

$$RPy := \frac{1.}{Nc}$$

$$RPy = \begin{pmatrix} 4.129 \\ 26.051 \end{pmatrix}$$ Return periods (yrs) of magnitude 4.0 and larger, and magnitude 5.0 and larger earthquakes

USGS (2008) earthquake activity rates for the Eastern Snake River Plain

$a1 := 0.0406$ USGS (2008) gridded a-value, interpolated value at EREF site

$b := 0.8$ USGS (2008) b-value for ESRP

$A1 := 89.7$ Area of 0.1 deg Lon by 0.1 deg Lat grid cell, sq. km.

$A2 := 33264$ Total area of ESRP, sq. km.

$Mw_0 := 4.0$

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$$a2 := \log\left(\frac{10^{a1} \cdot A2}{A1}\right)$$ Eq. 1 to determine a-value for change in area

$a2 = 2.61$ a-value for total area of ESRP

$LogNc := a2 - b \cdot Mw$ Recurrence model for total area of ESRP

$$LogNc = \begin{pmatrix} -0.59 \\ -1.39 \end{pmatrix}$$

$$Nc := 10^{LogNc}$$

$$Nc = \begin{pmatrix} 0.257 \\ 0.041 \end{pmatrix}$$ Annual rates of magnitude 4.0 and larger, and magnitude 5.0 and larger earthquakes

$$RPy := \frac{1.}{Nc}$$

$$RPy = \begin{pmatrix} 3.892 \\ 24.559 \end{pmatrix}$$ Return periods (yrs) of magnitude 4.0 and larger, and magnitude 5.0 and larger earthquakes

USGS (2008) earthquake activity rates for the Eastern Snake River Plain

$a1 := 0.182$	USGS (2008) gridded a-value, maximum value at nearest zone boundary
$b := 0.8$	USGS (2008) b-value for ESRP
$A1 := 89.7$	Area of 0.1 deg Lon by 0.1 deg Lat grid cell, sq. km.
$A2 := 33264$	Total area of ESRP, sq. km.

$Mw_0 := 4.0$

$Mw_1 := 5.0$

$$a2 := \log\left(\frac{10^{a1} \cdot A2}{A1}\right)$$ Eq. 1 to determine a-value for change in area

$a2 = 2.751$ a-value for total area of ESRP

$LogNc := a2 - b \cdot Mw$ Recurrence model for total area of ESRP

$$LogNc = \begin{pmatrix} -0.449 \\ -1.249 \end{pmatrix}$$

$$Nc := 10^{LogNc}$$

$$Nc = \begin{pmatrix} 0.356 \\ 0.056 \end{pmatrix}$$ Annual rates of magnitude 4.0 and larger, and magnitude 5.0 and larger earthquakes

$$RPy := \frac{1.}{Nc}$$

$$RPy = \begin{pmatrix} 2.811 \\ 17.734 \end{pmatrix}$$ Return periods (yrs) of magnitude 4.0 and larger, and magnitude 5.0 and larger earthquakes

The human system interface (HSI) design process translates function and task requirements into HSI characteristics and functions. The HSI uses a structured methodology that guides designers in identifying and selecting candidate HSI approaches, defining the detailed design, and performing HSI tests and evaluations. The process and the rationale for the HSI design is documented and controlled under the design control process described in the AES Quality Assurance Program Document (QAPD).

1.0 Human System Interface Design Inputs

The HSI design is developed based on various design inputs. The following HFE program element design inputs will be considered in making design decisions:

- operating experience review (OER),
- functional requirements analysis (FRA) and function allocation (FA),
- task analysis (TA), and
- staffing analysis

Additionally, the HSI design team considers applicable regulatory documents and codes as well as generic HFE standards and industry guidelines (see below).

1.1 Analysis of Personnel Task Requirements

Several analyses, as indicated below, may be performed in the early stages of the design process to identify HSI design requirements.

1.1.1 Operating Experience Review

An OER determines how the strengths and weaknesses of the HSI technology concept impact the effectiveness of the operator when using the technology. The goal of the OER is to compare the analysis of current work practices, operational problems and issues in current designs, and industry experience with candidate technological approaches to system and HSI technology and specific supplier solutions.

1.1.2 Functional Requirement Analysis and Function Allocation

FRA and FA determine which operational functions are to be performed by automatic systems, by plant personnel, or by some combination of the two. The allocation is made based on the FRA after determining what is required to perform the function. FA evolves from FRA and results in allocating functions for the best overall accomplishment for that function.

The results of the FRA and FA are used to identify the personnel role in performance of functions to reveal the task requirements and identify the HSI design implications. These HSI design implications include insight into the information that is to be displayed and how that information is presented. This information is used in the HSI procedure and training design to make sure that adequate task support is available to the operators.

1.1.3 Task Analysis

TA is performed for procedure development and is iterated as the HSI design detail evolves and involves determining the requirements for plant personnel to successfully perform complex real-time control actions that stem from functions assigned to them as a result of the FA design effort. Actions performed by plant personnel to accomplish a common-purpose group of activities or functions are called tasks. TA requirements are a primary consideration in design of the HSI.

1.1.4 Staffing and Qualifications and Job Analysis

Staffing and qualification analysis considers the allocation of assigned operational activities, the impact of those activities on crew member roles and responsibilities, and the impact of changes to operational requirements for the operating crew as a whole.

The results of the evaluation of staffing, qualifications, and integrated work design may impact the HSI design in terms of how operational activities are allocated to crew members, including assignments that make operational activities more efficient or reduce workload, how teamwork is supported, personnel qualifications, and required staffing levels.

1.2 System Requirements

The HSI system requirements will be documented for use throughout the HSI design process. The design control process facilitates the translation of high level requirements to lower level requirements, design inputs to design outputs, and high level design features to lower level subsystem and component design features.

The HSI consists of the controls, alarms, and indications used by the operator for performance of the IROFS safety function.

1.3 Regulatory Requirements and Guidance

The HSIs are designed to address the following regulatory requirements:

- 10 CFR 70.62(d) requires, in part, that "...engineered and administrative controls and control systems that are identified as items relied on for safety pursuant to §70.61(e) of this subpart 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 §70.61 of this subpart".

- 10 CFR 70.64(a) (10) requires that, "The design must provide for inclusion of instrumentation and control systems to monitor and control the behavior of items relied on for safety." Given that the AES-EREF application contains many IROFS that rely on human action, the instrumentation and control systems associated with these IROFS must be designed to adequately support operator task performance.

- NUREG-1513, "Integrated Safety Analysis Guidance Document," identifies that for administrative controls (e.g., certain human actions), "...the man-machine interface for that individual should be carefully designed."

The following references contain industry HFE guidance, which may be considered in the design of the HSIs:

- NUREG/CR-6633, "Advanced Information Systems: Technical Basis and Human Factors Review Guidance," U.S. Nuclear Regulatory Commission, March 2000.
- NUREG/CR-6634, "Computer-Based Procedure Systems: Technical Basis and Human Factors Review Guidance," U.S. Nuclear Regulatory Commission, March
- 2000.
- NUREG/CR-6635, "Soft Controls: Technical Basis and Human Factors Review Guidance," U.S. Nuclear Regulatory Commission, March 2000.
- NUREG/CR-6636, "Maintainability of Digital Systems: Technical Basis and Human Factors Review Guidance," U.S. Nuclear Regulatory Commission, March 2000.

2.0 Concept of Operations

The design of the plant I&C systems utilized to perform an IROFS function and the HSI consider the concept of operations including (1) the physical characteristics and technical abilities of the operating staff, (2) shift staffing and organization, and (3) responsibilities of the operational staff.

A description of the concept of operations and assumptions relative to the staffing, personal characteristics, division of team responsibilities, and other related issues that form the basis for the HSI design will be developed.

The concept of operations is primarily concerned with the operating team. The secondary concern includes system users to be considered in the design of other user interfaces.

3.0 Functional Requirements Specification

Functional requirements for the HSIs will be included in design documents for the HSIs to address the concept of operation, personnel functions and tasks that support their role in the plant as derived from function, task, and staffing/qualifications analyses, and personnel requirements for a safe, comfortable working environment. Requirements will be established for various types of HSIs, e.g., alarms, displays, and controls.

4.0 HSI Concept Design

The EREF will implement a modern I&C design utilizing experience gained at the George Besse II plant. The HSI concepts utilize similar I&C concepts.

5.0 HSI Detailed Design and Integration

A style guide will be developed for use in the design of HSI features, layout, and environment. The content of the Style guide will be derived from (1) the application of generic HFE guidance and (2) guidance developed from design-related analyses and experience. The style guide supports the interpretation and comprehension of design guidance and helps to maintain consistency in the design across the HSIs. The primary topics addressed by the style guide include data presentation, screen-based data presentation, hierarchy, and navigation, presentation and operation of controls, and presentation and interpretation of alarms.

6.0 HSI Tests and Evaluations (Verification and Validation)

Verification and validation (V&V) of the HSI design should be performed so that the as-built HSIs (1) are complete and operable, (2) conform to standard HFE principles and requirements, (3) are free of safety issues and human performance issues, and (4) implement the design accurately in the final design output documentation.

Testing and evaluation should be conducted throughout the HSI development process. Activities such as concept testing, mock-up activities, trade-off evaluations, and performance-based tests may be utilized at various stages of the design.

7.0 HSI Design Documentation

The HSI designs are documented using specific design control process requirements. The various configuration management, design change controls, design verification, and design quality control tools are described in the EREF QAPD.