NATIONAL
ENRICHMENT
FACILITY

10 CFR 30.6
10 CFR 40.5
10 CFR 70.5

May 19, 2004

NEF#04-018

ATTN: Document Control Desk
Director
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Louisiana Energy Services, L. P.
National Enrichment Facility
NRC Docket No. 70-3103

Subject: Response to NRC Request for Additional Information Regarding National Enrichment Facility Safety Analysis Report and Emergency Plan

References:
1. Letter NEF#03-003 dated December 12, 2003, from E. J. Ferland (Louisiana Energy Services, L. P.) to Directors, Office of Nuclear Material Safety and Safeguards and the Division of Facilities and Security (NRC) regarding "Applications for a Material License Under 10 CFR 70, Domestic licensing of special nuclear material, 10 CFR 40, Domestic licensing of source material, and 10 CFR 30, Rules of general applicability to domestic licensing of byproduct material, and for a Facility Clearance Under 10 CFR 95, Facility security clearance and safeguarding of national security information and restricted data"

2. Letter NEF#04-002 dated February 27, 2004, from R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC) regarding "Revision 1 to Applications for a Material License Under 10 CFR 70, "Domestic licensing of special nuclear material," 10 CFR 40, "Domestic licensing of source material," and 10 CFR 30, "Rules of general applicability to domestic licensing of byproduct material"

3. Letter dated April 19, 2004, from T. C. Johnson (NRC) to R. Krich (Louisiana Energy Services) regarding "Request for Additional Information on Louisiana Energy Services Project License Application"

By letter dated December 12, 2003 (Reference 1), E. J. Ferland of Louisiana Energy Services (LES), L. P., submitted to the NRC applications for the licenses necessary to authorize construction and operation of a gas centrifuge uranium enrichment facility. Revision 1 to these applications was submitted to the NRC by letter dated February 27, 2004 (Reference 2). By letter dated April 19, 2004 (Reference 3), the NRC provided the initial technical review of the license application and requested additional information and clarifications be provided within 30 days (i.e., by May 19, 2004).

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The Reference 3 letter includes the NRC Request for Additional Information (RAI) covering the National Enrichment Facility (NEF) Safety Analysis Report (SAR) and Emergency Plan. This letter transmits the LES responses to these requests. As noted and referenced in various responses, certain requests are responded to under separate letter due to the nature of the reply (e.g., proprietary) or desire for earlier transmittal.

Attachment 1 to this letter provides the RAIs with the associated LES response.

Attachment 2 to this letter provides Tables referenced in various RAI responses.

Attachment 3 to this letter provides Figures referenced in various RAI responses.

Attachment 4 to this letter provides LES Procedure DP-ISA-1.1, "IROFS Boundary Definition," which is referenced in various RAI responses.

If you have any questions, please contact me at 630-657-2813.

Respectfully,

Daniel G. Green for

R. M. Krich
Vice President – Licensing, Safety, and Nuclear Engineering

Attachments:

1. LES Response to April 19, 2004 Request for Additional Information
2. LES Response to April 19, 2004 Request for Additional Information: Tables Referenced from Responses
3. LES Response to April 19, 2004 Request for Additional Information: Figures Referenced from Responses
4. LES Response to April 19, 2004 Request for Additional Information: LES Procedure DP-ISA-1.1, "IROFS Boundary Definition"

cc: T.C. Johnson, NRC Project Manager

ATTACHMENT 1

Louisiana Energy Services
Response to April 19, 2004
Request for Additional Information

Louisiana Energy Services
Responses to April 19, 2004
Requests for Additional Information

Chapter 1.0 General Information

GI-1 Section 1.2.1.2, pp. 1.2-1 and 1.2-2

Provide a copy of the LES Partnership Agreement.

Regulations in 10 CFR 70.22(a)(1) require the applicant to provide the corporate name of the applicant and the name of the State where it is incorporated or organized.

The applicant provided general information on the partnership structure. However, information on the financing and partnership control responsibilities needs to be provided.

LES Response

In response to Request for Additional Information (RAI) GI-1, a copy of the Louisiana Energy Services (LES) Partnership Agreement was provided by letter NEF#04-009 dated May 10, 2004, from R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC).

GI-2 Section 1.2.1.2, 1st Para., p. 1.2-1

Provide the name of the LES subsidiary formed for the purpose of purchasing the Lea County Industrial Revenue Bonds and the name of the State where it is incorporated or organized. Also, provide a copy of the Industrial Revenue Bond agreement with Lea County.

Regulations in 10 CFR 70.22(a)(1) require the applicant to provide the corporate name of the applicant and the name of the State where it is incorporated or organized.

LES indicated that it has a wholly-owned subsidiary for the purpose of purchasing Industrial Revenue Bonds issued by Lea County, but did not provide its name or State of incorporation or organization. In addition, the Industrial Revenue Bond agreement needs to be provided to verify the licensing responsibilities of the applicant and Lea County.

LES Response

The response to RAI GI-2 and a copy of the closing papers for the Lea County, New Mexico Industrial Revenue Bond (National Enrichment Facility Project) Series 2004 were provided by letter NEF#04-010 dated May 10, 2004, from R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC).

GI-3 Section 1.2.2, p. 1.2-3

Provide a detailed estimate of the cost to construct the plant.

The regulations in 10 CFR 70.22 provide that where the nature of the proposed activities requires consideration of the applicant's financial qualifications, the Commission may request information with respect to financial qualifications.

The applicant estimated the cost to construct the plant at approximately $1.2 billion in 2002 dollars. The applicant stated that this estimate is the cost to design and construct the facility, and the estimate excluded escalation, a contingency, interest, and any replacement equipment that may be needed during the life of the plant. The application did not provide a detailed basis that supported the $1.2 billion estimate. Because the NRC must make a finding in accordance with 10 CFR 70.23(a)(5), regarding whether the applicant appears to be financially qualified to engage in the purposed activities, the staff will need to review the supporting basis for the $1.2 billion estimate. The validity of the estimated cost, with its supporting assumptions is a key factor in determining if the applicant is financially qualified. LES should submit a detailed estimate of the cost to construct the plant.

LES Response

The detailed estimate of the cost to construct the facility is considered to be proprietary and will be submitted once clearance is obtained from the owner of the information.

GI-4 Section 1.2.2, p. 1.2-4

Provide the amount of public liability insurance to be provided and the basis for the amount proposed.

Under the regulations in 10 CFR 140.13b, a licensee of a uranium enrichment facility must have and maintain liability insurance.

In the application, the applicant indicated that when the plant is ready for operation it will obtain liability insurance coverage to closely approximate the $300 million limit. The applicant needs to provide the amount of coverage it will obtain to meet the requirements in 10 CFR 140.13b. In addition, the regulations require that the liability insurance be obtained prior to issuing the license, not prior to operations.

LES Response

In the next revision to the Safety Analysis Report (SAR), LES will clarify the 10 CFR 140.13b requirements discussed in Section 1.2.2.

LES is currently in possession of a $1,000,000 stand-by liability policy through American Nuclear Insurers (ANI). A letter to this effect from ANI was included with the submittal of the license application. The basis for future liability insurance will be based on a risk assessment, performed by ANI, on the engineering and planned operational capacity of the plant. This process will begin during the project engineering and construction phase. The ANI process also requires a site visit and inspection of the facility as the NEF approaches operation and production. ANI currently anticipates that at the time of operation it can write a limit for their current maximum capacity of $300,000,000; this amount may vary depending on the result of the risk assessment.

10 CFR 140.13b requires liability insurance of the type and in the amounts to cover liability claims arising out of any occurrence within the United States, causing, within or outside the United States, bodily injury, sickness, disease, death, 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. Until such time as LES takes possession of source or special nuclear material, the effects described in 10 CFR 140.13b involving source or special nuclear material are not possible. Therefore, the $1,000,000 standby liability policy, in addition to appropriate construction coverage, is considered to be sufficient for the construction phase. As described above, ANI will conduct a site visit and inspection of the facility prior to operation and revise the limit for liability insurance, depending on the result of the risk assessment. As required by 10 CFR 140.15, LES 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.

GI-5 Section 1.2.3, p. 1.2-4

Provide possession limits for all proposed licensed material in terms of total quantity to be possessed.

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

The applicant in Table 1.2-1 provides information on the types of material proposed for use in average annual quantities. The applicant should provide total quantities of licensed material to be possessed. In addition, any other licensed material, including potential sources of contamination in UF_6, such as ^{99}Tc, and any calibration sources proposed to be used should be identified and quantity limits proposed.

LES Response

Text removed under 10 CFR 2.390.

SAR Table 4.11-1 provides a listing of typical byproduct materials and quantities that may be used at the National Enrichment Facility (NEF) for instrument calibration and/or instrument checks. As reflected in SAR Section 1.2.3, the source and byproduct materials to be used for instrument calibration will be identified during final design. At that time, the types, forms, and quantities of these materials will be provided to the Nuclear Regulatory Commission (NRC) for inclusion in the license.

To preclude potential sources of contamination of UF_6, such as ^{99}Tc, LES will require UF_6 suppliers to provide Commercial Natural UF_6 in accordance with ASTM C 787-96, "Standard Specification for Uranium Hexafluoride for Enrichment." In addition, cylinder suppliers will be required to preclude use of cylinders that, in the past, have contained reprocessed UF_6, unless they have been decontaminated. Periodic audits of suppliers will be performed to provide assurance that these requirements are satisfied. Therefore, LES is not proposing to possess these types of materials.

Chapter 2.0 Organization and Administration

OA-1 Section 2.1.1, p. 2.1-1 and Figures 2-1 and 2-2

Clarify the organization charts for the design and construction organization and the operating organization.

10 CFR 70.22(a)(6) requires the technical qualifications, including training and experience, of the applicant and staff to engage in the proposed activities.

Figure 2.1-1 provides a diagram of the proposed organization for design and construction. Figure 2.1-2 provides a diagram of the organization for operations. However, the references in Figure 2.1-1 to Figure 2.1-2 are confusing and appear to duplicate some positions (e.g., Health, Safety, and Environment Manager).

LES Response

During the design and construction phase the Health, Safety, & Environment (HS&E) Manager position will report directly to the LES President (as shown in Figure 2.1-1). During the full capacity operating phase, the HS&E Manager position will become part of the organization reporting to the Plant Manager (as shown in Figure 2.1-2). This position is intentionally duplicated to provide significant continued focus on the health, safety, and environment goals during design and construction when the operating organization is not yet developed and implemented. When full capacity operations begin, it is appropriate for that position to assume the appropriate operating role.

Certain other positions will experience similar transitions from involvement in NEF design and construction and during full capacity operations assume either a direct NEF organizational duty or remain strictly corporate organization responsibilities. For example, certain engineering, projects, and communications functions specific to the NEF will transfer to part of the operating organization, while similar corporate functions would remain part of the Corporate Organization for other LES responsibilities.

OA-2 Sections 2.2.1, p. 2.2-2 & p. 2.2-4; 2.2.4, pgs. 2.2-9 - 2.2-10; 5.1.5,pgs. 5.1-4 - 5.1-5; and Emergency Plan Section 4.1, p. 4.1-2 & p. 4.1-4

Clarify the positions and responsibilities of the Health, Safety, and Environmental Manager, Criticality Manager, Criticality Safety Engineer, and Nuclear Criticality Engineer.

10 CFR 70.22(a)(6) requires the technical qualifications, including training and experience, of the applicant and staff to engage in the proposed activities.

Sections 2.2.1 and 2.2.4 identify the operating organization and personnel qualification requirements, including those for the Health, Safety, and Environment Manager and Criticality Safety Engineer. Section 5.1.5 identifies relevant Nuclear Criticality Safety staff, including a Nuclear Criticality Manager and a Nuclear Criticality Engineer that are not described in Section 2.2.1 and 2.2.4. Also, the responsibilities described in Chapter 2.0, Chapter 5.0, and the Emergency Plan for the same positions are different.

LES Response

SAR Chapter 2.0, the Emergency Plan Chapter 4.0, and the Fundamental Nuclear Material Control Plan (FNMCP) Chapter 1.0, responsibilities consistently describe the appropriate responsibilities for the HS&E Manager and criticality safety engineer. In the next revision to SAR Section 5.1.5, to enhance organizational clarification, the reference to "criticality safety manager" will be replaced with "HS&E Manager" and responsibilities of the "criticality safety staff" will be clarified as responsibilities of the HS&E Manager. Also, refer to response to OA-4 for additional commitments and clarifications related to the organization.

OA-3 Sections 2.2.1, p. 2.2-2; and 2.2.4, p. 2.2-4

Provide the qualifications of individuals that may be designated to (1) review and approve changes to the facility or activities of personnel that require NRC approval prior to making the change, in place of the Health, Safety and Environment Manager and (2) review and approve changes to the facility or to operations that involve chemical, radiation hazard, or criticality considerations prior to making the change, in place of the Health, Safety, and Environmental Manager.

10 CFR 70.22(a)(6) requires the technical qualifications, including training and experience, of the applicant and staff to engage in the proposed activities.

Section 2.2.1.E refers to "designees" that have approval authority in place of the Health, Safety, and Environmental Manager. However, there needs to be a discussion of the qualification requirements for these individuals in either Section 2.2.1 or Section 2.2.4.

LES Response

SAR Section 2.2.1.E will be modified in the next revision to delete the discussion related to approving changes to the facility or activities of personnel that require NRC approval. Change control processes and responsibilities are appropriately addressed in Section 11.1.4, Change Control, and are not intended to be detailed in the Chapter 2 overview of responsibilities. Furthermore, the requirements and scope of delegation procedures is generally addressed in Chapter 11, Appendix A, Quality Assurance Program Description, Section 1, under Delegation of Work, which states:

Responsible managers have the authority to delegate tasks to another qualified individual within their organization provided the designated individual possesses the required qualifications and these qualifications are documented. All delegations shall be in writing. The responsible manager retains the ultimate responsibility and accountability for implementing the applicable requirements.

OA-4 Section 2.2.1, 2.2.4, and 5.1.5, General

Clarify which position(s) will be responsible for the NCS management and NCS supervision activities described in ANS-8.19, "Administrative Practices for NCS" and provide this information in either Section 2.2.3 or Section 5.1.5.

10 CFR 70.62(a) requires the establishment and maintenance of a safety program to demonstrate compliance with the performance requirements of 10 CFR 70.61. 10 CFR 70.65(a) requires a description of this safety program to be submitted in the license application.

Sections 2.2.1, 2.2.4, and 5.1.5 need to clearly describe which individual has responsibility for different elements of the NCS program. It is unclear if the Health, Safety, and Environment Manager has responsibilities for both NCS management and NCS supervision as well as what the phrase "administration of NCS reviews" means. It is unclear if the Criticality Safety Engineer has responsibilities for both NCS supervision and NCS staff. Also, it is unclear whether a Criticality Safety Engineer will be onsite during all shift operations, and, if not, whether a Criticality Safety Engineer will be able to effectively respond to emergency conditions.

LES Response

In the next revision to SAR Chapter 2, LES will specifically state a commitment to the Management Responsibilities, Supervisory Responsibilities, and Nuclear Criticality Safety Staff Responsibilities provided in Sections 4, 5, and 6 of ANS-8.19-1996, "Administrative Practices for Nuclear Criticality Safety." Specifically:

- The reporting responsibility for the "Key Management Positions" that are not specifically identified as "managers" (e.g., Criticality Safety Engineer) will be clarified to reflect that reporting may be through designated supervisors. The SAR Section 2.2.1(R) description for the Criticality Safety Engineer is consistent with ANS-8.19-1996, Section 5 description of staff responsibilities. No intent to convey supervisory responsibilities is presented in this description. The managers identified in the SAR Chapter 2 "Key Management Positions" currently reflect the responsibility to assure appropriate conduct of activities under their authority, which would include assigning, as necessary, adequate supervisory personnel. LES does not intend to reflect specific discussion of Supervisory positions and titles within the SAR. The organizational structure at this level, and appropriate responsibilities and commensurate authority, will be reflected in site-specific guidance documents approved by the Plant Manager.

- The Section 2.2.4(E) reference to "administration of nuclear criticality safety reviews" will be revised to "administration of nuclear criticality safety evaluations and analyses." (Note clarification of these terms is being addressed in response to RAI ISA-2.)

While ANS-8.19-1996, or NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," does not address recommendations for the Criticality

Safety Engineer to be onsite during all shift operations, and/or whether a Criticality Safety Engineer will be able to effectively respond to emergency conditions, LES does intend to maintain qualified criticality safety engineer(s) available, with appropriate ability to be contacted and to respond to any routine request or emergency condition. This commitment will be addressed in the next revision to Chapter 2 of the SAR. It is also noted that the LES Emergency Plan, Section 4.4.2(g), provides for an HS&E Coordinator with expertise in criticality safety as part of the activated emergency response organization.

Chapter 3.0 Integrated Safety Analysis Summary

ISA-1 Chapter 3.0, General

Clarify the separation between the Integrated Safety Analysis (ISA) Summary, which does not need to be incorporated into the license, and the programmatic commitments related to the ISA and ISA Summary that are required to be in the application.

10 CFR 70.62(a) requires the establishment and maintenance of a safety program to demonstrate compliance with the performance requirements of 10 CFR 70.61.
10 CFR 70.65(a) requires a description of this safety program to be submitted in the license application. 10 CFR 70.65(b) requires the ISA Summary to be submitted with the license application, but shall not be incorporated into the license.

It is unclear what part of Chapter 3.0 are the programmatic commitments for the ISA and ISA Summary as well as descriptions of how to meet those commitments that are needed to meet 10 CFR 70.62(a) and 70.65(a) versus what part of Chapter 3.0 is the ISA Summary that is needed to meet 10 CFR 70.65(b).

LES Response

In NEF SAR Section 3.0, it is clear that each Section of Chapter 3 contains material (in part or in whole) that would be required to meet 10 CFR 70.65(b) contents for the ISA Summary. Additionally, in the next revision to the SAR, LES will revise Chapter 3 to also relocate additional detail from other SAR chapters into Chapter 3, to meet 10 CFR 70.65(b)(4) required content of the ISA Summary. As such, all requirements necessary for ISA Summary content will be adequately addressed completely within Chapter 3. Specifically, description of the Criticality Accident Alarm Systems (CAAS) (refer to response to RAI ISA-6) will be moved from SAR Chapter 5 to Chapter 3, and a description of the management measures will be added to Chapter 3 in response to various RAIs regarding clarification of management measures applied to items relied on for safety (IROFS) (e.g., ISA-43, ISA-48, ISA-58, NCS-6).

However, LES has not segregated the portions of Chapter 3 that would be required solely for the 10 CFR 70.65(a) required content of "a description of the applicant's safety program" or portions that may not be required for the minimum content of either 10 CFR 70.65(a) or (b) for the following reasons.

NUREG-1520 provides the following in its Executive Summary:

> *... because this SRP describes the scope, level of detail, and acceptance criteria for reviews, it serves as regulatory guidance for applicants who need to determine what information to present in a license application and related documents. ...*

Each nuclear fuel cycle facility license application should contain a safety program description that addresses all of the topics listed in the table of contents of this SRP, in the same order in which they are presented in this document.

The review criteria from NUREG-1520 Chapter 3 provides for review of major portions of the safety program as well as the ISA Summary. While the review criteria clearly requires confirmation that the ISA Summary meets the content requirements of 10 CFR 70.65(b), there is no reference within NUREG-1520 to defining the portions of the application that explicitly reflect meeting 10 CFR 70.62(a) and 10 CFR 70.65(a).

As is reflected in this RAI, there is also much explicit differentiation made in 10 CFR Part 70 (and the associated Statements of Consideration from the Federal Register Notice dated September 18, 2000), as well as in NUREG-1520, between the "license application" and the docketed submittal of the ISA Summary. However, since they both are required to be submitted, have materials that reside within the same document (i.e., SAR), have identical reporting requirements for changes (in accordance with 10 CFR 70.72(d) (2) and (3)), and reflect submissions that form the basis for NRC review and license approval, the specific requested delineation does not appear to be a technically relevant or regulatory-based issue.

In this RAI, it is noted that the statement "ISA Summary, which does not need to be incorporated into the license" is explicitly contrasted with the statement "the programmatic commitments related to the ISA and ISA Summary that are required to be in the application." In so doing, it could be inferred that there is some intent to include "the programmatic commitments related to the ISA and ISA Summary that are required to be in the application" within the License. This intent is supported by February 24, 2004 issuance of the American Centrifuge Lead Cascade Facility Materials License (SNM-7003; docket No. 70-7003) in which License Condition 10 incorporates the "statements, representations, and conditions" in the various 10 CFR 70.22 application documents and revisions.

LES has concluded that this action is unwarranted and is not in accordance with the revisions made to 10 CFR Part 70 (reference Federal Register Notice dated September 18, 2000), guidance found in NUREG-1520, or the objectives of the Part 70 rule change.

Specific attention is brought to Federal Register Notice dated September 18, 2000, "Background" Section, where Commission objectives are noted: "(5) the allowance for licensees to make certain changes to their safety program and facilities without prior NRC approval." If the NRC intent were to make the statements, representations, and conditions in the entire License Application (with the exception of the ISA Summary) a License Condition for the NEF, then LES would not be able to make any changes the statements, representations, and conditions (e.g., safety program and facilities) in the entire License Application without obtaining prior NRC review and approval of a license amendment.

ISA-2 Sections 3.0, 3.1.1, 3.1.1.1, 3.1.5, 5.1.6, 5.2, 5.3, and Emergency Plan Section 4.1, General

Clarify the different terminology used for NCS documents and provide the purpose, use, content, and relationships between the documents in Chapter 5.0.

10 CFR 70.62(a) requires the establishment and maintenance of a safety program to demonstrate compliance with the performance requirements of 10 CFR 70.61.
10 CFR 70.65(a) requires a description of this safety program to be submitted in the license application.

Throughout the application, there are different terms used for NCS documents, such as: NCS analyses, NCS assessments, NCS determinations, NCS evaluations, criticality safety analyses, criticality safety assessments, criticality safety evaluations, criticality evaluations, criticality assessments, and criticality evaluations. It is unclear if these documents have different purposes, uses, and content. It is unclear how each document relates to the others.

One example is the use of NCS Determinations. Section 3.1.5 describes one purpose (i.e., "NCS Determinations are specialized studies that assure the risk of having a criticality accident is highly unlikely, and that the double contingency principle is satisfied."); Section 5.2.1.3 describes a different purpose (i.e., "The NCS Determinations presented in Section 5.3 provide values of k-eff to conservatively meet the USL."); and, during the site visit to Massachusetts in March 2004, applicant staff indicated a third purpose and applied in a way during the ISA process not described in the application (i.e., NCS Determinations were unclassified summaries of the four basic applicant NCS documents - three bounding NCS evaluations and a critical dimensions document).

LES Response

The next revision to the SAR will provide consistent nomenclature for the documents, and/or associated activities to produce these documents, described above. In general, the following clarifications will be shown in this next revision:

- Criticality Safety "Determinations" is being replaced with Criticality Safety "Analyses" (corresponding SAR Section 5 Titles also appropriately renamed)
- Clarify that there are two unique documents/processes:

  (1) Nuclear Criticality Safety Evaluation (NCSE) – Non-calculation engineering judgments regarding whether existing criticality safety analyses (which are included within the ISA) and ISA Summary event sequences bound an issue being evaluated, or whether new or revised safety analysis and/or revision to the ISA Summary is required

  (2) Nuclear Criticality Safety Analyses – Engineering calculations performed in accordance with SAR Chapter 5 and documented in the ISA, ISA Summary (i.e., SAR Chapter 3), and and/or Chapter 5, as appropriate

- Table 3.7-2 references to criticality safety "evaluations" are revised to be consistent with similar Table 3.7-2 references to criticality safety "requirements"

"Assessments" is utilized in a variety of contexts, such as "hazards assessments," radiological assessments," and various "audits and assessments." Additionally, "criticality assessment" is utilized in Section 3.1.1.1 to describe the initial hazards identification process. To provide enhanced clarity, the phrase "criticality assessment" in this context will be revised to "criticality hazards identification."

ISA-3 Sections 3.1.1, p. 3.1-1 and Chapter 3.0, General

Clarify whether the statement in Section 3.1.1, "The approach used for performing the ISA is consistent with Example Procedure for Accident Sequence Evaluation, Appendix A to Chapter 3.0 of NUREG-1520 (NRC, 2002)" was intended to mean a commitment to follow the example in the NUREG.

10 CFR 70.65(b)(4) requires in the ISA Summary information to demonstrate compliance with the performance requirements of 10 CFR 70.61.

For example, Tables 3.1-7, 3.1-8, 3.1-9, 3.1-10, and 3.1-11 are essentially the same as those in Appendix A to Chapter 3.0 of NUREG-1520, Tables A-4, A-8, A-9, A-10, and A-11, respectively. However, it is unclear if the tables were used properly because the descriptions of the tables in the ISA Summary did not include all the accompanying text from NUREG-1520 that describes how the tables should be used.

LES Response

As stated in SAR Section 3.1.1, the LES approach to performing the ISA was consistent with Example Procedure for Accident Sequence Evaluation, Appendix A to Chapter 3.0 of NUREG-1520. As stated in 3.1.1.1, the HAZOP analysis is consistent with the guidance provided in NUREG-1513, "Integrated Safety Analysis Guidance Document," and NUREG-1520. The intent is to convey only general conformance with methods, processes, and approaches presented in these guidance documents. No specific exception to following the general guidance is taken. NEF-specific processes and engineering judgements of the ISA Team were followed, whereas the example of NUREG-1520, Appendix A was not specific to an actual NEF process. As such, a commitment to follow the example explicitly was not the intent of the statement.

However, one point of potential confusion was identified in reviewing the request to clarify how the example Tables were used. As SAR Table 3.1-7 (which corresponds to NUREG-1520, Appendix A Table A-4) was not specifically utilized in the NEF ISA process, and adequate bases for the ISA method is otherwise presented, SAR Table 3.1-7, and reference to it, will be deleted in the next revision to the SAR.

The NUREG-1520 example procedure also did not utilize Table A-4 -- presenting it as "for illustrative purposes only" and "coarse criteria." The NUREG-1520 example details for the ISA risk index assignment that followed the presentation of Table A-4, used data from Tables A-9, A-10, and A-11 (and not Table A-4). Similarly, NEF SAR Tables 3.1-9, 3.1-10, and 3.1-11 (which correspond to NUREG-1520 Tables A-9, A-10, and A-11) were utilized in performing the NEF ISA.

ISA-4 Section 3.1.1.1, p. 3.1-2

Clarify the differences in the terminology "normal and bounding conditions" and "normal and credible abnormal conditions."

10 CFR 70.61(d) requires that the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical.

In Section 3.1.1.1 the applicant described a process used for evaluating normal and bounding conditions, but there needs to be an explanation of how the "bounding conditions" relate to "credible abnormal conditions" to meet the regulations.

LES Response

The regulatory required analysis for "normal and credible abnormal conditions" was intended. In the next revision to SAR Section 3.1.1.1, the phrase "bounding conditions" will be revised to "credible abnormal conditions."

ISA-5 Sections 3.1.5 and 5.2.1.3, General

Clarify the approach used to meet the performance requirements of 10 CFR 70.61 and the associated regulations for NCS and provide this information in the ISA Summary.

10 CFR 70.62(a) requires the establishment and maintenance of a safety program to demonstrate compliance with the performance requirements of 10 CFR 70.61.
10 CFR 70.65(a) requires a description of this safety program to be submitted in the license application. 10 CFR 70.65(b)(4) requires in the ISA Summary information to demonstrate compliance with the performance requirements of 10 CFR 70.61.

It is unclear what approach was used to meet the performance requirements of 10 CFR 70.61 for NCS. Examples include: (1) Section 3.1.5 indicates that the NCS Determinations were used in some manner and (2) Section 3.8.1 indicates that an alternative process was sometimes used.

LES Response

The approach used to meet the performance requirements of 10 CFR 70.61 is described in SAR Section 3.1.1. (Note that references to "NCS Determinations" are being revised to "NCA Analyses" in response to RAI ISA-2.) In the next revision to the SAR, the RAI-referenced Section 3.1.5 (note that the response to ISA-6 will result in the first two paragraphs of Section 3.1.5 being moved up into Section 3.1.4) will be clarified to be consistent with the SAR Chapter 5 description. Specifically, the existing second paragraph of Section 3.1.5 will be revised to:

> *Nuclear criticality safety is evaluated for the design features of the plant system or component and for the operating practices that relate to maintaining criticality safety. The evaluation of individual systems or components and their interaction with other systems or components containing enriched uranium is performed to assure the criticality safety criteria are met. The nuclear criticality safety analyses in Chapter 5, and the safe values in Table 5.1-1, Safe Values for Uniform Aqueous Solution of Enriched UO_2F_2, provide a basis for the plant design and criticality hazards identifications performed as part of the integrated safety analysis described in Chapter 3, Integrated Safety Analysis Summary.*

Additionally, Section 3.8.1 will be re-written to clarify that only methodology described in SAR Section 3.1.1 was used to demonstrate compliance with the performance requirements in 10 CFR 70.61. No alternative method was employed in the ISA. The responses to RAIs ISA-42, ISA-46, and ISA-62 provide additional detail related to these changes.

ISA-6 Section 3.1.5 and 5.5

Clarify the information about the Criticality Accident Alarm Systems (CAAS).

10 CFR 70.22(a)(7) requires a description of equipment and facilities which will be used by the applicant to protect health and minimize danger to life or property (such as criticality accident alarm systems). 10 CFR 70.62(a) requires the establishment and maintenance of a safety program to demonstrate compliance with the performance requirements of 10 CFR 70.61. 10 CFR 70.65(a) requires a description of this safety program to be submitted in the license application. 10 CFR 70.65(b)(4) requires in the ISA Summary information to demonstrate compliance with the performance requirements of 10 CFR 70.61, including the requirements for criticality monitoring and alarms in 10 CFR 70.24.

It appears that not all the information in Section 3.1.5 is appropriate (i.e., it is not about CAAS). It appears that some of the information in Section 5.5 is appropriate for the ISA Summary (e.g., figures and text regarding the figures). For guidance only, see NUREG-1520, Section 5.4.3.4.3.

LES Response

The initial two paragraphs of 3.1.5, which do not address the CAAS, will be moved to the end of Section 3.1.4, Hazards Analyzed, in the next revision to the SAR. The remaining discussion in Section 3.1.5, which does address CAAS, will be enhanced with appropriate discussion from Section 5.5. Also, in the next revision to the SAR, Section 5.5 will be abbreviated and include a reference to Section 3.1.5 for details. The guidance from NUREG-1520, Section 5.4.3.4.3 will be adequately addressed with the revision. The details shown in Figures 5.5-1 through 5.5-5 will also be removed in the next revision to the SAR.

ISA-7 Section 3.1.7, page 3.1-16

Provide a discussion of how Items Relied on for Safety (IROFS) are protected from environmental conditions and dynamic effects and how the requirements of 10 CFR 70.64(a)(4) are met for individual IROFS. The discussion should consider appropriate industry standards. Also, discuss how non-IROFS will be able to withstand environmental stress caused by environmental and dynamic service conditions under which their failure could prevent satisfactory accomplishment of safety functions by IROFS. Provide information on the facility's essential utility services (if any) and how the design provides for their continued operation.

The regulations, 10 CFR 70.64(a), require that the applicant address the baseline design criteria. Specifically, 10 CFR 70.64(a)(4) requires that the design must provide for adequate protection from environmental conditions and dynamic effects associated with normal operations, maintenance, testing, and postulated accidents that could lead to loss of safety functions. Also 10 CFR 70.64(a)(7) requires that the design must provide for continued operation of essential utility services.

Section 3.1.7.D of the application stated that "Structures, systems, and components that are determined to have safety significance (IROFS) are protected against dynamic effects of missiles and discharging fluids, that may result from natural phenomena, accidents at nearby industrial, military, or transportation facilities, equipment failure, and other similar events and conditions both inside and outside the facility." Since this statement does not indicate how IROFS are protected from environmental conditions and dynamic effects, provide a discussion of how the requirements of 10 CFR 70.64(a)(4) are met for individual IROFS. The discussion should consider appropriate industry standards. Also, discuss how non-IROFS will be able to

withstand environmental stress caused by environmental and dynamic service conditions under which their failure could prevent satisfactory accomplishment of safety functions by IROFS.

Section 3.1.7.G of the Safety Analysis Report stated that "On site utility service systems required to support IROFS shall be provided. Each utility service system required to support IROFS shall provide for the meeting of safety demands under normal and abnormal conditions." Since this statement does not identify the facility's essential utility services (if any) and does not discuss how the design provides for their continued operation, provide this information.

LES Response

SAR Section 3.8.1 states that the 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, as required by the ISA. As such, the IROFS components and systems and components will be designed, procured, constructed, and maintained such that environmental conditions and dynamic effects associated with normal operations, maintenance, testing, and postulated accidents will not lead to a loss of the IROFS safety function. To accomplish this, IROFS components and systems will be qualified using the applicable guidance in Institute of Electrical and Electronics (IEEE) standard IEEE-323, 1983, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." The qualification for each IROFS component and system will demonstrate that the IROFS will perform their safety functions under environmental and dynamic service conditions in which they will be required to function and for the length of time the function is required. In this same manner, non-IROFS components and systems will be qualified to be able to withstand environmental stress caused by environmental and dynamic service conditions under which their failure could prevent satisfactory accomplishment of IROFS safety functions. This approach is consistent with the application of equipment qualification requirements at nuclear power plants.

For the current IROFS and design, no essential utility service systems have been identified that are required to support the IROFS safety function. Upon completion of the design and identification of IROFS, the IROFS boundaries will be defined. In defining the boundaries for each IROFS, LES procedure DP-ISA-1.1, "IROFS Boundary Definition," will be used. (A copy of DP-ISA-1.1 is provided in Attachment 4). This procedure requires the identification of each support system and component necessary to ensure the IROFS is capable of performing its specified safety function. As such, essential utility service systems that are required to support the safety function of the IROFS will fall within the boundary of the IROFS. This procedure also requires identification of the management measures necessary to ensure that the IROFS, including any required support systems and components, availability and reliability are maintained consistent with the assumptions of the ISA (as will be described in SAR Section 3.1.8.3 -- refer to response to RAI ISA-43). If implementation of DP-ISA-1.1 for the final design of the IROFS identifies that essential utility service systems are within the IROFS boundary, then these essential utility service systems will be designed to withstand environmental stresses caused by environmental and dynamic service conditions under which failure could prevent the satisfactory accomplishment of the IROFS safety function. This will be accomplished in the same manner as described above for the IROFS components and systems.

ISA-8 Section 3.1.7-I, p. 3.1-17; 3.8.1, p. 3.8-2; 5.1.1, p. 5.1-1; and 5.7, p. 5.7-1

Clarify the commitments to double contingency principle, double contingency protection, as well as clarify the quote from the ANS-8.1 standard regarding the double contingency principle.

10 CFR 70.62(a) requires the establishment and maintenance of a safety program to demonstrate compliance with the performance requirements of 10 CFR 70.61.
10 CFR 70.65(a) requires a description of this safety program to be submitted in the license application. 10 CFR 70.65(b)(4) requires in the ISA Summary information to demonstrate compliance with the performance requirements of 10 CFR 70.61. 10 CFR 70.65(b)(4) requires in the ISA Summary information to demonstrate compliance with the requirements of 10 CFR 70.64(a)(9).

There needs to be clear and consistent commitments to the double contingency principle and double contingency protection throughout Chapter 3.0 and Chapter 5.0.

Examples of inconsistency include: Section 3.1.7-I states that, "All process and storage systems shall be designed to be maintained subcritical and to ensure that no nuclear criticality accident can occur unless at least two unlikely, independent, and concurrent changes have occurred in the conditions essential to nuclear criticality safety." Section 3.8.1 states that, "For accident sequences postulated to result in nuclear criticality, the double contingency protection requirement is satisfied by IROFS and multiple independent controls on a single process parameter." Section 5.1.1 states that, "The adopted double contingency principle states 'process designs shall incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.'" Section 5.1.1 states that "In the current design, each process that has accident sequences that could result in an inadvertent nuclear criticality at the [facility] will have double contingency protection." Section 5.7 states that, "The double contingency principle will be used in determining NCS controls and IROFS in the design of new facilities or new processes..."During the onsite visit in Massachusetts in March 2004, applicant staff indicated that for the initial design of the facility, the commitment is to the double contingency principle, as described in Section 5.1.1 and then afterwards, the commitment is to the double contingency protection, as described in Section 5.3.16.

Also, the reference to the double contingency principle from the ANS-8.1 standard, "NCS in Operations with Fissionable Materials Outside Reactors" needs to be changed to reflect that it is a different statement from that in the standard.

LES Response

The next revision to the NEF SAR will more clearly reflect the intent for the process design to meet the 10 CFR 70.64(a)(9) regulatory requirement to the double contingency principle (as defined in 10 CFR 70.4) for criticality control, and providing double contingency protection, when practicable. Responses to various other RAIs also relate to clarifying instances where double contingency protection did not appear to be met (e.g., ISA-42); as such, the response here will simply focus on clarifying the general concept and commitments. Specific exceptions (if any) to double contingency protection will be addressed in other RAI responses.

The guidance from NUREG-1520, Section 5.4.3.4.4(7)(c) describes the possibility that "there may be processes where double contingency protection is not practicable" (which is also reflected in NEF SAR Section 5.3.16, "Additional NCS Determinations"). Consistent with the NUREG-1520 review criteria and the existing SAR Chapter 5 commitment, any exceptions to meeting double contingency protection will be described in the ISA Summary (SAR Chapter 3) and justification provided. The justification will assure that there is sufficient redundancy and diversity in controlled parameters such that at least two unlikely and concurrent events, errors, accidents, or equipment malfunctions are necessary before an inadvertent nuclear criticality is possible.

In the next revision to the NEF SAR, the following specific clarifications will be made:

- In general, when appropriate, references to meeting double contingency principle/protection will have added "when practicable."

- Section 3.1.7.I, Criticality Control, the statement referenced in the RAI will be modified consistent with the review criteria from NUREG-1520 to state:

  All process and storage systems should be designed and maintained with sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible. When this is not practicable, the design justification will assure that there is sufficient redundancy and diversity in controlled parameters such that at least two unlikely and concurrent events, errors, accidents, or equipment malfunctions are necessary before an inadvertent nuclear criticality is possible.

- Section 5.1.1 will be corrected to accurately quote American National Standards Institute/American Nuclear Society (ANSI/ANS) ANSI/ANS-8.1-1998, "Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors," definition of double contingency principle (replace "shall" with "should"), and include the discussion of "when practicable" and associated commitment as stated in 3.1.7.I (see above).

- Section 5.7, first bullet page 5.7-1, will be revised to correctly reference NUREG-1520 Section 5.4.3.4.4(7). (NUREG-1520 Section 5.4.3.4.5 contains a typo, incorrectly referencing Section 5.4.3.4.4(9), which had been copied in the original SAR development.)

- Section 5.7, third bullet page 5.7-2, will be revised to more closely reflect the Reporting Requirements of 10 CFR 70, Appendix A, which has criteria based on IROFS loss and degradation, not based on "double contingency" loss and degradation.

<u>ISA-9</u> <u>Section 3.2.6.1, pp. 3.2-23 through 3.2-29</u>

Provide results of investigations conducted to identify any capable faults within a 322-km [200-mi] radius. Page 3.2-23 of the Safety Analysis Report states, "No Quaternary faults are mapped for the site locale. The nearest recent faulting is situated more than 161 km [100 mi] west of the site." It is not clear if any of these faults are capable.

Under 10 CFR 70.62(c)(iv), an applicant is required to address potential accident sequences caused by external events, including natural phenomena. The Standard Review Plan, NUREG-1520, on page 3-12, (1)c, states that characterization of natural phenomena (e.g., tornadoes, hurricanes, floods, and earthquakes) and other external events is needed to assess their impact on facility safety and to assess their likelihood of occurrence.

Based on the information provided, the staff is unable to determine if the faults cited are capable.

LES Response

Locations of recent tectonic faulting within the 322-km (200-mi) radius of the NEF site located in Lea County, New Mexico, were determined through literature research. The publications used to support the conclusions that "no Quaternary faults are mapped for the site locale" and "the nearest recent faulting is situated more than 161 km (100 mi) west of the site," were as follows.

1. Map and data for Quaternary faults and folds in New Mexico, Open-File Report 98-521, Machette, Michael N., Personius, Stephen F., Kelson, Keith I., Dart, Richard L., and Haller, Kathleen M., US Geological Survey, (electronic version, last revision 2000).

2. Earthquake Hazards Program, Quaternary Fault and Fold Database of the United States, US Geological Survey, 2004.

3. Waste Isolation Pilot Plant, WIPP Contact Handled Safety Analysis Report, DOE/WIPP-95-2065, Revision 7, US Department of Energy June 2003.

(Note: Figures ISA-9.1 through ISA-9.4 are provided in Attachment 3.) Figure ISA-9.1, Quaternary Faults in New Mexico, and Figure ISA-9.2, Quaternary Faults in Texas, illustrate traces of Quaternary Faults for New Mexico and adjacent areas of west Texas. The Quaternary geologic time period extends from 1.6 million years ago to the present. Other time sub-divisions within the Quaternary include the Late Quaternary that extends from 130,000 years ago to the present, and the Holocene, which includes the most recent 10,000-year time period.

Shown on Figures ISA-9.1 and ISA-9.2 are 1° Latitude by 2° Longitude geographic blocks. The NEF site is located in the Hobbs geographic block. Geographic blocks containing Quaternary faults are color-coded (i.e., non-gray). Figure ISA-9.3, Quaternary Faults Within 322 km (200 mi) of NEF Site, shows geographic blocks for which Quaternary faults are mapped. All of these geographic blocks are located west of the NEF site. Figure ISA-9.4, Locations of Nearest Faults to the NEF Site, shows the Quaternary fault locations detailed in the "Map and data for Quaternary faults and folds in New Mexico, U.S. Geological Survey (USGS) Open-File Report 98-521." The block containing the site, as well as others due north, south, and east of the NEF site has no documented Quaternary faults. Quaternary faults within 322 km (200 mi) of the site are shown on Figure ISA-9.3 using colored and numbered traces, and are plotted over shaded relief topographic maps. The use of topographic relief maps is highly illustrative, because ground deformations resulting from recent fault movements are usually manifested as prominent linear topographic features.

Figure ISA-9.4 provides a summary of Quaternary fault locations, including fault names obtained from the "Map and data for Quaternary faults and folds in New Mexico, USGS Open-File Report 98-521" and the "Earthquake Hazards Program, Quaternary Fault and Fold Database of the United States." Characteristics of these faults are described below. Following descriptions of these faults, an assessment is made whether the fault is "capable" as defined in the 10 CFR 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants" and Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Motion," dated March 1997.

The following descriptions of Quaternary-Aged Faults provide summaries of results of investigations conducted to identify and describe faults within a 322-km (200-mi) radius of the NEF site. The designation of faults as capable or non-capable is based on these data.

Alamogordo Fault (No. 2054b, 2054c)

This fault is a normal fault dipping down to the west and bounds the west side of the Sacramento Mountains. Average strike of the fault is N9°W. The total fault length is 109.5 km (68 mi).

Nearest distance to the site:	262 km (163 mi)
Most recent prehistoric deformation:	Latest Quaternary (< 15,000 years ago)
Recurrence interval:	two episodes in last 35,000 years
Slip rate category:	less than 0.2 mm/yr.
Capable fault:	Yes

Guadalupe Fault (No. 2058)

This normal fault forms a zone of short but continuous scarps on unconsolidated sediment (alluvial fans and colluvium) shed from the Guadalupe Mountains. The fault is 5.6 km (3.5 mi) long, strikes N5°W and dips to the west.

Nearest distance to the site:	191 km (119 mi)
Most recent prehistoric deformation:	Latest Quaternary (< 15,000 years ago)
Recurrence interval:	two discrete faulting events
Slip rate category:	less than 0.2 mm/yr.
Capable fault:	Yes

Carlsbad Fault

The Carlsbad Fault is shown on Figure ISA-9.4 as the NE-trending fault in the southeast corner of the Carlsbad block. This fault, however, is not included as a Quaternary fault in the "Earthquake Hazards Program, Quaternary Faults and Fold Database of the United States," nor is it shown in the "Map and data for Quaternary faults and folds in New Mexico, Open-File Report 98-521." Based on discussions with the report author (M. Machette), it was revealed that this fault is no longer considered a Quaternary fault. There is no evidence to support Quaternary movement.

Nearest distance to the site:	116 km (72 mi)
Most recent prehistoric deformation:	Older than Quaternary
Recurrence interval:	Not determined
Slip rate category:	Not determined
Capable fault:	No

Unnamed Fault at Base of Guadalupe Mountains (No. 907)

This normal fault is a range-bounding fault at the base of the Guadalupe Mountains. The fault is 9.6 km long, strikes N29°W and dips 74° to the southwest.

Nearest distance to the site:	179 km (111 mi)
Most recent prehistoric deformation:	Quaternary (<1.6 million years)
Recurrence interval:	No detailed studies
Slip rate category:	less than 0.2 mm/yr., by association with other regional faults
Capable fault:	No

East Flat Top Mountain Fault (No. 908)

This normal fault is a down-to-the east fault that bounds the western margin of Salt Basin. The fault is 21.1 km (13.1 mi) long, strikes N8°W and dips to the east.

Nearest distance to the site:	200 km (124 mi)
Most recent prehistoric deformation:	Late Quaternary (<130,000 years)
Recurrence interval:	No detailed studies
Slip rate category:	less than 0.2 mm/yr., by association with other regional faults
Capable fault:	Yes

North Sierra Diablo Fault (No. 909)

This normal fault is a down-to-the north fault that bounds the north margin of the Sierra Diablo and Salt Basin. The fault is 4.3 km (2.7 mi) long, strikes N83°W and dips to the north.

Nearest distance to the site:	193 km (120 mi)
Most recent prehistoric deformation:	Quaternary (<1.6 million years)
Recurrence interval:	No detailed studies
Slip rate category:	less than 0.2 mm/yr., by association with other regional faults
Capable fault:	No: Recent movement in last 35,000 years not demonstrated.

East Sierra Diablo Fault (No. 910)

This normal fault is a down-to-the east fault that bounds the east margin of the Sierra Diablo and western side of Salt Basin. The fault is 32.5 km (20.2 mi) long, strikes N1°W and dips to the east.

Nearest distance to the site:	196 km (122 mi)
Most recent prehistoric deformation:	Late Quaternary (<130,000 years)
Recurrence interval:	80,000 to 160,000 years
Slip rate category:	less than 0.2 mm/yr.
Capable fault:	Yes

West Delaware Mountain Fault Zone (No. 911)

These normal faults are down-to-the-west en echelon faults that bound the west flank of the Delaware Mountains from the eastern (deepest) part of Salt Basin. The fault zone has a cumulative length of 32.1 km (19.9 mi), strikes N30°W and dips to the southwest.

Nearest distance to the site:	185 km (115 mi)
Most recent prehistoric deformation:	Late Quaternary (<130,000 years)
Recurrence interval:	Not determined
Slip rate category:	less than 0.2 mm/yr., based on other faults
Capable fault:	Yes

East Baylor Mountain – Carrizo Mountain Fault (No. 912)

These normal faults are down-to-the-east en echelon faults that separate the southwestern flank of the Baylor Mountains and eastern flanks of the Beach and Carrizo Mountains from the Wild Horse basin. The fault zone has a cumulative length of 40.8 km (25.4 mi), strikes N24°W and dips to the southeast.

Nearest distance to the site:	201 km (125 mi)
Most recent prehistoric deformation:	Mid to Late Quaternary (<750,000 years)
Recurrence interval:	125,000 to 250,000 years
Slip rate category:	less than 0.2 mm/yr.
Capable fault:	Possible, but movement in last 35,000 years is not demonstrated.

The above information supports the conclusions that "no Quaternary faults are mapped for the site locale" and that "the nearest recent faulting is situated more than 161 km (100 mi) west of the site." Some of the Quaternary faults located west of the LES site are "capable." However, none of the capable faults pose a ground deformation hazard to the NEF site due to the distances (> 161 km (100 miles)) from the site, the northerly strike of these faults, and the associated topographic landforms shown in Figure ISA-9.4. The strikes of the assessed capable faults do not project toward the NEF site. Topographic features, like those correlated to the Quaternary faults west of the site, are not present near the NEF site, thus making it an unlikely scenario that unmapped "capable faults" are located nearer than 161 km (100 miles) to the NEF site.

<u>ISA-10 Section 3.2.6.1, pp. 3.2-23 and 3.2-24</u>

Include in Tables 3.2-20 and 3.2-21 of the Safety Analysis Report focal depths and distances to site for all events greater than magnitude 3. Also, include all available magnitude designations (i.e., m_b, M_L, M_s, and M_w).

Under 10 CFR 70.62(c)(iv), an applicant is required to address potential accident sequences caused by external events, including natural phenomena. The Standard Review Plan, NUREG-1520, on page 3-12, (1)c, states that characterization of natural phenomena (e.g., tornadoes, hurricanes, floods, and earthquakes) and other external events is needed sufficient to assess their impact on facility safety and to assess their likelihood of occurrence.

The earthquake focal depths and distances to site for all earthquake events greater than magnitude 3 and the appropriate magnitude designations are needed to appropriately consider the potential effect of earthquake events.

LES Response

Attachment 2 provides Tables ISA-10.1 and ISA-10.2 containing the requested information where available. In the next revision to the SAR, the additional requested information will be included in associated SAR Tables 3.2-20 and 3.2-21 (except certain additional aftershock data (e.g., see 1931 event) that are not reported in SAR Tables 3.2-20 and 3.2-21). Additionally, SAR Table 3.2-21 will be revised to reflect magnitude 3.0 and greater, versus the current listing of magnitude 4.0 or greater.

Discussion of Magnitudes

Magnitude scaling is thoroughly discussed in the New Mexico Bureau of Geology and Mineral Resources, New Mexico Institute of Mining and Technology (NMIMT), 2002, (the data citing this source is shown as "NMTH" and "NMTR"). The following quote is taken from NMIMT:

> **"Earthquake Magnitudes"**
>
> "A major effort was made to have all magnitudes in our catalog based on or tied to a New Mexico duration magnitude scale (Newton et al., 1976; Ake et al., 1983). For determining magnitudes the relation
>
> $$M_d = 2.79 \log \tau_d - 3.63$$
>
> was used, where τ_d is duration in seconds." ...... "The basic data for the NMIMT duration magnitude equation were 64 Wood-Anderson seismograph recordings of earthquakes in New Mexico and bordering areas. Local magnitudes (M_L) calculated from the amplitudes on the Wood-Anderson seismograms (Richter, 1958) were linearly related to the logarithm of τ_d measured on seismograms from the Albuquerque (ALQ) station of the World Wide Seismographic Network (WWSN). Hanks and Kanamori (1979) have demonstrated that local magnitude (M_L) is equivalent to moment magnitude (M)."

Given the above discussion, all events listed in Tables ISA-10.1 and ISA-10.2 originating from the NMIMT catalogs have Moment Magnitude scaling, M, determined from an empirical regional coda magnitude scale.

For the earthquake data from University of Texas at Austin Institute for Geophysics (UTIG), specifics on magnitude scaling are not described, however, the earthquake list uses (M) which typically is used to show (Local) or (Moment) magnitude scales.

The third data source, the composite earthquake list maintained by the Advanced National Seismic System (ANSS), produced a total of 64 events, 40 of which are listed with unspecified magnitude type (un). Magnitude scaling for remaining events includes body-wave magnitudes (m_b), Local Magnitude (M_L) and two entries of coda-wave magnitude (M_C).

Data from NMIMT provides a magnitude adjustment for events co-located by other regional or national seismological organizations. NMIMT determined that the USGS reported a slightly higher (mean difference of 0.185 units) magnitude for 182 events separately analyzed by New Mexico Institute of Mining and Technology and the USGS. Given this reasonably small

deviation in magnitude determination, no adjustments were made to scale the earthquakes obtained from the ANSS Earthquake Hazards Program.

Discussion of Focal Depths

Focal depth information is only available for the events obtained from the ANSS catalog. These focal depths have typical upper crustal depths ranging from 1 to 33 km (0.6 to 20 mi). Focal depths for these events are included in the Tables ISA-10.1 and ISA-10.2. Earthquake data obtained from NMIMT and UTIG do not provide focal depth information. Focal depths are inferred to be upper crustal with depths (1 to 33 km (0.6 to 20 mi)) reported by the USGS in the ANSS composite catalog.

ISA-11 Section 3.2.6.2, pp. 3.2-24 and 3.2-25

Explain how seismic source regions for the site are determined on the basis of the earthquake frequency pattern shown on Figure 3.2-21. Specifically, explain how the spatial density was calculated and provide the appropriate units on the legend of Figure 3.2-21.

Under 10 CFR 70.62(c)(iv), an applicant is required to address potential accident sequences caused by external events, including natural phenomena. The Standard Review Plan, NUREG-1520, on page 3-12, (1)c, states that characterization of natural phenomena (e.g., tornadoes, hurricanes, floods, and earthquakes) and other external events is needed to assess their impact on facility safety and to assess their likelihood of occurrence.

The requested information is needed to correctly read the contours shown on Figure 3.2-21.

LES Response

In the context of performing probabilistic seismic hazard assessments, a seismic source zone is defined as a geographic region that is characterized by a uniform earthquake potential that is distinct from earthquake potentials of surrounding regions. Uniform earthquake potential implies the same maximum magnitude and a uniform earthquake recurrence frequency throughout a seismic source zone. This requires maps that depict regions of equal earthquake recurrence frequency to help delineate seismic source zone boundaries.

Mapping software (e.g., ArcMap GSI, Environmental Systems Research Institute) provides tools that enable distribution and evaluation of point data, such as earthquake epicenters, over a geographic area. One of these tools determines a contour map of data point density. Data point density calculations are performed by first defining (a) a search radius, and (b) a cell size for the output density contour map. Density maps result from summing points located in the search area defined by the search radius and dividing by the corresponding map area. This procedure is conducted iteratively for each defined map cell to produce a density map. For the case of large search radii and small map cells, as was used to develop Figure 3.2-21, the search areas significantly overlap. This overlapping feature results in magnifying the point count at all locations with the purpose of increasing the resolution of the density contours. The earthquake frequency contours shown on Figure 3.2-21 are meant to provide a visual portrayal of areas with similar earthquake counts per area, i.e., earthquake density. The density units themselves are not meant to be absolute but a relative representation of earthquake frequency from one location to another location.

Units of resulting density maps depend on selection of the length of the search radius, the area of output map cells, and units of the selected map projection (e.g. decimal degrees, meters,

kilometers). Resulting units represent total events in the search area, normalized by the search area in map projection units. Therefore, the density contours represent event counts per area, rather than a strict count of epicenters located within a contour line.

The utility of the density map is to analytically study the regional distribution of earthquake epicenters. Resulting density contour maps (e.g. Fig. 3.2-21) illustrate regions of similar earthquake epicenter density, as required for definition of seismic source zones, and provide a sound technical basis for determining the boundaries of regional seismic source zones. Earthquake recurrence frequencies are determined using statistical analyses performed on the sub-catalog of events contained within a defined seismic source. Actual density contour values resulting from the density map calculations are not considered in the statistical determination of earthquake recurrence frequencies.

To clarify the presentation of SAR Figure 3.2-21, the following note will be added to the figure in the next revision to the SAR:

> NOTE: The earthquake frequency contours shown provide a visual portrayal of areas with similar earthquake counts per area, i.e., earthquake density. The density units themselves are not absolute, but a relative representation of earthquake frequency from one location to another location.

ISA-12 Section 3.2.6.4, pp. 3.2-26 through 3.2-28

Discuss the possible effects caused by human activities such as withdrawal of fluid from or addition of fluid to the subsurface on the evaluation of tectonic structures underlying the site and the region surrounding the site. If possible, identify the seismic events related to gas and oil recovery methods in the vicinity of the site, including the magnitudes and locations of these events and the effects on the recurrence models if these events are removed.

Under 10 CFR 70.62(c)(iv), an applicant is required to address potential accident sequences caused by external events, including natural phenomena. The Standard Review Plan, NUREG-1520, on page 3-12, (1)c, states that characterization of natural phenomena (e.g., tornadoes, hurricanes, floods, and earthquakes) and other external events is needed sufficient to assess their impact on facility safety and to assess their likelihood of occurrence.

As stated in Section 3.4.3.2.(1)c of the Standard Review Plan, the applicant should assess which events could occur without adversely impacting safety. The staff requests that the applicant provide a discussion of possible effects caused by human activities such as withdrawal of fluid from or addition of fluid to the subsurface.

LES Response

Regional earthquake recurrence models and probabilistic seismic hazard estimates for the NEF site assumed a tectonic origin for all events in the Central Basin Platform (CBP) sub-region. This assumption was conservative and appropriate given the published uncertainties on discrimination between natural and induced seismic events. Earthquake focal depths, critical for correlation with oil/gas reservoirs, are largely unavailable. Even for the case of the January 2, 1992, magnitude 5 earthquake, focal depths range from 5 km (3.1 mi) (USGS, 2004) to 12 km (7.5 mi) (DOE, 20003). Studies (refer to discussion following reply) conclude that seismological data are insufficient for this moderate earthquake to constrain the depth sufficiently to permit a correlation with local oil/gas producing horizons.

It is currently not possible to definitively differentiate natural tectonic from induced seismic events in the study region, but for such cases of uncertainty, sensitivity analyses done for seismic hazard analysis can provide valuable insights into the impacts of induced earthquakes. The following sensitivity analysis results are provided to show trends in seismic hazard results for assumptions that increasing percentages of earthquakes in the CPB seismic source zone are induced by oil/gas recovery activities.

Two hypotheses are considered in the seismic hazard sensitivity analyses. First is the case is that a fraction of earthquakes of all magnitudes are induced. Second is the case that only smaller magnitude earthquakes (e.g., less than M=3.5) are likely induced while larger events result from tectonic processes. Consistent with Sanford commentary in the Background discussion that follows, that a large fraction of events in the CBP was induced by oil/gas recovery efforts, is modeled by scaling the CBP recurrence model by factors of 0.15, 0.5, and 0.85. These scaling factors are applied to the entire recurrence model such that the predicted frequencies of events for all magnitudes are scaled by these factors. The three scaling factors are used to model the general commentary that a "large fraction" of CPB events are induced. For the second case, the concept that many of the small events could be induced while larger events have tectonic origins is modeled by re-computation of the recurrence model for the CPB following removal of 50% of events with magnitudes less than 3.5. This case results in a recurrence model that predicts relatively fewer small magnitude events, and recurrence rate of larger events of magnitude 5.0 and greater remains unchanged.

Seismic hazard sensitivity results show a significant impact only when a scaling factor of 0.15 is applied to the total recurrence model. For this instance, peak horizontal acceleration (pga) is reduced from about 0.15g to about 0.10g at 1.0 E-4 annual exceedance probability. Application of a scaling factor of 0.50 to the entire model resulted in a pga near 0.13g at 1.0 E-4 annual exceedance probability. Two of the analyses, scaling the entire recurrence model by 0.85, and determination of a new model based on removal of 50% of events smaller than M=3.5, showed little sensitivity. Given uncertainties related to the tectonic vs. induced nature of larger regional events, and high likelihood that many smaller events are induced by ongoing oil/gas recovery activities, results of the last sensitivity analysis (e.g. removal of smaller events only) are preferred. The negligible sensitivity to removal of smaller events emphasizes that seismic hazard in large part is determined by the assessed regional frequency of events with magnitudes larger than 5.0.

Background / Reference Studies

Possible effects of fluid withdrawal from, or injection to, subsurface geologic strata include vibratory ground motion impacts from induced seismic events. Induced seismicity can be of a "triggered" nature wherein seismogenic tectonic features, able to produce natural earthquakes, prematurely release strain energy as a result of fluid addition or removal. Also, induced seismicity can result from the creation of new small fractures that occur during hydrofracturing procedures intended to make oil/gas reservoirs more permeable. Typically, smaller magnitude induced seismicity, ranging from microseismic swarms to largest magnitudes in the range of 3 to 4, is associated with enhanced oil recovery (Canadian Induced Seismicity Research Group, 2004).

Discrimination between induced and natural seismic events usually requires extensive field investigations spanning several years. Grasso and Wittlinger (1990) concluded based on 10-years of continuous seismographic measurements that approximately 800 earthquakes of magnitude 1.0 to 4.2 were induced by gas extraction in a major gas field in Lacq, France. Their conclusion depended on precise three-dimensional location of seismic events using a network of up to 8 closely spaced seismograph stations. All events were located within the area of gas

production and the causative mechanism was determined to be triggering of ruptures above the gas reservoir due to pore pressure reductions associated with gas extraction.

Keller et al. (1987) studied seismic activity over a 46-month period in the Permian Basin Keystone oil field near Kermit, Texas. They conclude that over 1,300 events located during their study exhibited a complex relationship with known geologic structures and oil field activity. Concentrations of small earthquakes (too small to be felt) showed a strong spatial correlation to the War-Wink gas field. Also concluded by Keller et al. is the basement of the CBP is complexly faulted and that a portion of the observed seismicity is associated with a major, deep-seated structural anomaly which created many zones of weakness in Paleozoic and older rocks. In summary, this study demonstrated a strong spatial correlation of microseismicity with the War-Wink gas field, but other regional larger events could be tectonic in nature, and fundamental understanding of these events would require further study.

Sanford et al. (2002, 1993) reported that ongoing research strongly suggests that a large fraction of activity in southeastern New Mexico and adjacent areas of west Texas is induced by production, secondary recovery, or waste injection within this petroleum and natural gas province.

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- Grasso, J.R. and Wittlinger, G., 1990, Ten Years of Seismic Monitoring Over a Gas Field, Bulletin of the Seismological Society of America, Vol. 80, No. 2, pp. 450-473.

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- Sanford, A.R., Balch, R., Jaksha, L., and Delap, S., 1993, Location and Fault Mechanism of the 2 January 1992 Rattlesnake Canyon Earthquake in Southeastern New Mexico, Geophysics Open-File Report 70, January 7, 1993, 9p., New Mexico Institute of Mining and Technology, Socorro NM 87801.

- Sanford, Allan R., Lin, K.W., Tsai, I., and Jaksha, L.H., 2002, Earthquake catalogs for New Mexico and bordering areas: 1869-1998, New Mexico Bureau of Geology and Mineral Resources, NMIMT, Circular 210.

- US Department of Energy, 2003, Waste Isolation Pilot Plant, WIPP Contact Handled Safety Analysis Report, DOE/WIPP-95-2065 Revision 7, June 2003.

- US Geological Survey Earthquake Hazards Program, Advanced National Seismic System, 2004, Composite Earthquake Catalog Search

ISA-13 Section 3.2.6.5, p. 3.2-29

Describe the design safety margins, structural elasticity and conservatism needed to demonstrate that use of a 10,000 year (1.0 E-4) earthquake in the detailed design process can achieve a performance level of less than about 1.0 E-5 for seismic IROFS.

10 CFR 70.64(a)(2), Natural Phenomena Hazards, requires that the design must provide for adequate protection against natural phenomena with consideration of the most severe documented historical events for the site. 10 CFR 70.61(b) requires that the risk of each credible high consequence event must be limited. High consequence events are those internally or externally (i.e., seismic) initiated events that result in specified chemical and/or radiological exposures.

Section 3.2.6.5, Selection of the Design Basis Earthquake, identifies a 10,000 year return earthquake as the design basis earthquake (DBE) to be used in the detailed design process to demonstrate compliance with the overall ISA performance requirements. Confirmatory seismic performance calculations for the seismic IROFS will be performed to demonstrate that use of the DBE will achieve a likelihood of unacceptable performance of less than approximately 1.0 E-5. The difference between the mean annual probabilities for design (1.0 E-4) and performance (1.0 E-5) is achieved through conservatism in the design (factors of safety), elasticity in the structures, and conservatism in the evaluation of the design.

LES Response

SAR Section 3.2.6.5, Selection of the Design Basis Earthquake, notes that Department of Energy (DOE) Standard DOE-STD-1020-94, "Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities," dated April 1994, and American Society of Civil Engineers (ASCE) Standard Seismic Design Criteria, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities and Commentary (draft standard), Nuclear Standards Committee, Dynamic Analysis of Nuclear Structures Subcommittee," dated July 2003, provide details of the approach that will be utilized for the NEF during detailed seismic design to demonstrate compliance with the overall ISA performance requirements. Both the DOE and ASCE standards outline methodologies to demonstrate compliance to a target performance goal of 1.0E-05 annual probability by designing to a seismic hazard of 1.0E-04 annual probability. For NEF, the 1.0E-04 annual probability earthquake has been selected as the design basis earthquake (DBE) as described in SAR Section 3.2.6.5. For NEF, following the DOE and ASCE approaches, this provides a risk reduction ratio of design to target performance of 10 (1.0E-04/1.0E-05). The DOE and ASCE standards address design and evaluation of structures, systems, and components (SSCs). The equivalents of SSCs for the NEF are considered to be the IROFS and the items that may affect the function of IROFS.

The objective of the NEF seismic design approach is to introduce sufficient design safety margins, i.e., conservatism, during the design process to allow for demonstration of compliance to the target performance goal. The DOE and ASCE standards implement this objective using slightly different methodologies with the same end result, i.e., demonstration of compliance to the target performance goal.

In the DOE approach, the deterministic seismic evaluation and acceptance criteria are structured to achieve less than a 10% probability of unacceptable performance for a SSC subjected to the scaled design/evaluation basis earthquake (SDBE). The SDBE is defined in the DOE approach as the product of the DBE times a factor of 1.5 and a scale factor, which is a function of the slope of the seismic hazard curve.

The ASCE approach is based on achieving the target performance goal annual frequencies by incorporating sufficient conservatism in the seismic demand and structural capacity evaluations to achieve both of the following:

- Less than about a 1% probability of unacceptable performance for the DBE ground motion

- Less than a 10% probability of unacceptable performance for a ground motion equal to 150% of the DBE ground motion

The ASCE method is based on achieving both of the above probability goals, which represent two points on the underlying fragility curve. Meeting these two probability goals allows the target performance probabilities to be achieved with less possibility of non-conservatism. The resulting nominal factors of safety against conditional probability of failure are 1.0 and 1.5, respectively, for the above two goals.

The actual seismic design detailed approach for NEF will be based on the DOE and/or ASCE methods and finalized prior to detailed design. The safety margins will be representative of those discussed above and described in more detail in the DOE and ASCE standards.

During the seismic design process, there are many areas where conservatism is introduced. Examples include:

- Limits on inelastic behavior
- Damping
- Analysis methods
- Specification of material strengths
- Determination of structural capacity (code required factors of safety)
- Deflection and drift limits
- Design overstrength

Additional information on these conservative features is provided below.

Limits on inelastic behavior are introduced by utilizing inelastic energy absorption factors. By permitting less inelastic behavior, the margin of safety is effectively increased and the probability of damage is reduced in accordance with the target performance goal. Traditional elastic analysis does not take any credit for inelastic behavior. However, energy absorption in the inelastic range of response of structures and equipment to earthquake motions can be very significant. Elastic analysis approaches are recommended in both the DOE and ASCE methods for initial assessment of the structures and equipment. The elastically computed seismic responses are then reduced in the two methods to account for the inelastic absorption capacity of structures and equipment.

For NEF, earthquake demand on IROFS, and the items that may affect the function of IROFS, will be reduced by inelastic energy absorption. Representative inelastic energy absorption factors are provided in both the DOE and ASCE standards. The inelastic energy absorption factor is defined as the amount that the elastic-computed seismic demand may exceed the capacity of a component without impairing the performance of the component. The inelastic energy absorption factors presented in the DOE and ASCE standards were established to approximately meet the target performance goals for structural behavior of the analyzed structures and equipment. It is recognized that the inherent seismic resistance of a well-designed and constructed structure is usually much greater than that expected based on elastic analysis. This occurs largely because nonlinear behavior is mobilized to limit the imposed forces.

Damping accounts for energy dissipation in the linear range of response of structures and equipment to dynamic loading. Lower damping results in increased seismic forces, introducing conservatism. So as not to be non-conservative, it is important to use damping which is consistent with stress levels reached in the majority of the lateral force resisting system in the

evaluation of input to structure supported components or input for interaction considerations. Both standards provide recommended damping values for structures and equipment appropriate for use in demonstrating compliance to the desired performance targets.

Analysis methods utilizing industry guidelines, as recommended in the DOE and ASCE standards, will introduce conservatisms in the design. High confidence in the earthquake safety of facilities is achieved by including redundancy, ductility, tying elements together to behave as a unit, adequate equipment anchorage, detailing of connections and reinforced concrete elements, and quality of design and materials in the analyses.

Specification of material strengths, in accordance with building code provisions, introduces industry, recognized conservatisms. Similarly, building code provisions provide for conservatisms in the determination of structural capacities. Both the DOE and ASCE approaches specify the utilization of building code provisions for the determination of material strengths and capacities. Model code provisions aid in designing structures and equipment to have sufficient strength and stiffness to resist lateral loads induced by earthquake shaking. Additional conservatism is introduced through use of material strengths in the design process that are less than the actual strength of the material.

The DOE and ASCE standards require limiting deflections and drifts. In many cases limiting deflections and drifts will control the design of structural elements and will be in excess of stress requirements. This will also introduce conservatism in the design. Design overstrength, typically introduced by the designer, will be on the order of 10 percent of the minimum required strength, but may rise as high as 50 to 100 percent in some cases.

Through use of the approach outlined above with its associated safety margins and conservatisms, the NEF IROFS, and the items that may affect the function of IROFS, designed to the 1.0E-04 DBE, will be demonstrated through confirmatory seismic calculations to meet the ISA performance requirements.

ISA-14 Section 3.3.1.2.2.18, p. 3.3-8; Section 3.5.7, pp. 3.5-40 and 3.5-41; Section 3.5.9.2.1, pp. 3.5.44 and 3.5.45; and Section 5.5, p. 5.5-1

Describe the process used to conduct the human factors engineering review of the Control Room, the Communication and Alarm Annunciation System, the Central Control Room, and the Criticality Accident Alarm System as it applies to IROFS requiring operator actions.

The regulations in 10 CFR 70.61 and 70.62 require that an applicant perform an integrated safety analysis of the hazards associated with the proposed facility and demonstrate compliance with the performance requirements in 10 CFR 70.61(b), (c), and (d).

The applicant describes the Control Room in Section 3.3.1.2.2.18, the Central Control System in Section 3.5.9.2.1, the Communication and Alarm Annunciation System in Section 3.5.7, and the Criticality Accident Alarm System in Section 5.5. For those IROFS functions requiring operator actions, the applicant should describe the process used to conduct the human factors engineering review of these areas, and for any other safety - significant human-system interfaces located outside the areas. NUREG-0711, Rev. 1, "Human Factors Engineering Program Review Model," dated 2004, and NUREG-0700, Rev. 2, "Human-System Interface Design Review Guidelines," dated 2004, are sources that can be used to conduct this review, adjusted as appropriate for the facility ISA. Operating experience with these systems at similar Urenco enrichment facilities in Europe may also be used to conduct this review to give NRC

review staff additional confidence that the facility will meet the performance requirements in 10 CFR 70.61.

LES Response

The Urenco Group gas centrifuge plants have been operating and evolving since the early seventies. The best design features of this evolution together with the wealth of operating experience were recently brought together into a Core Plant Design (CPD) by means of a significant exercise known as the Urenco Plant Design Forum (UPDF). The UPDF brought together the design, maintenance, and operating experience from across the Urenco group of companies to derive best practices for all elements of the plant. Incorporated implicitly in the CPD are all the human factors engineering enhancements from over 30 years of experience. The SP5 plant design, and therefore the NEF, is based on this CPD.

Additionally, every element of the design is subject to formal hazard and operability (HAZOP) analysis and design review and operations experienced personnel are mandatory members during such HAZOPs and design reviews.

Urenco design review guidelines include design review topics of functionality, operability, maintenance, layout, and orientation. Urenco engineering design safety principles also address the human factors engineering issues. Specifically these principles state that the design of all interfaces between operating personnel and the plant should follow good human factors and ergonomics practice. These principles also note that analysis of the safety function tasks requires determination of the demands on personnel in order to evaluate the feasibility of the tasks and provide input to the design interfaces. The design of tasks and equipment should be fully compatible with training arrangements for operations personnel, proposing staffing levels, and the development of operating procedures.

ISA-15 Section 3.3.2.2.6.2, p. 3.2-28

The application states, "Rainfall loadings on roofs and other exposed surfaces result from two different events. The first event is normal heavy rainfall having a 100-year return period." Provide information about the rainfall with a 100-year return period, including amount and duration. Also, provide the technical basis on how this 100-year return period rainfall was determined.

Under 10 CFR 70.62(c)(iv), an applicant is required to address potential accident sequences caused by external events, including natural phenomena. The Standard Review Plan, NUREG-1520, on page 3-12, (1)c, states that characterization of natural phenomena (e.g., tornadoes, hurricanes, floods, and earthquakes) and other external events is needed to assess their impact on facility safety and to assess their likelihood of occurrence.

The staff requires the additional information to determine how the rainfall loadings were determined from the two events stated by the applicant.

LES Response

The basis of the 100-year rainfall is from the 2000 International Plumbing Code (IPC). The IPC requirement is based on a storm of 1-hour duration and a 100-year return period (reference IPC Chapter 11, Paragraph 1106.1 and Figure 1106.1). For Hobbs, NM, the rainfall rate is given as 3.0 inches per hour (reference IPC Chapter 11, Figure 1106.1 and Appendix B).

As an independent check, this value was compared to the 1-hour duration, 100-year return period, reported by the National Weather Service for the Hobbs and Eunice, NM areas and found to be in close agreement.

ISA-16 Section 3.3.2.2.6.2, p. 3.3-28

The application states, "The second event is localized intense rainfall associated with the Design Basis Flood. The rainfall distribution to this event is discussed in Section 3.2." The staff is unable to locate this discussion in Section 3.2 of the Safety Analysis Report. The first paragraph in Section 3.2.3.4.4 of the Safety Analysis Report discusses local intense probable maximum precipitation. The second paragraph in Section 3.2.4.3 of the Safety Analysis Report indicates no design basis flood is considered for the NEF site.

Under 10 CFR 70.62(c)(iv), an applicant is required to address potential accident sequences caused by external events, including natural phenomena. The Standard Review Plan, NUREG-1520, on page 3-12, (1)c, states that characterization of natural phenomena (e.g., tornadoes, hurricanes, floods, and earthquakes) and other external events is needed to assess their impact on facility safety and to assess their likelihood of occurrence.

Clarify these inconsistent statements. Indicate clearly where in Section 3.2 of the Safety Analysis Report the localized intense rainfall associated with the Design Basis Flood is discussed.

LES Response

The statements in SAR Section 3.3.2.2.6.2 requiring clarification should read as follows: "The second event is localized intense rainfall. Refer to Section 3.2.3.4.4 for further discussion." Delete the phrase "associated with the Design Basis Flood," and the sentence "the rainfall distribution for this event is discussed in Section 3.2."

As indicated in SAR Section 3.2.4.3, the nearest water conveyance to the NEF site is Monument Draw, which is typically dry. As described in SAR Section 3.2.4.3, Monument Draw is not a source of site flooding at the NEF. Since there are no perennial-flow surface water rivers or streams in the site vicinity, reference in Section 3.3.2.2.6.2 was not appropriate, and the conclusion in SAR Section 3.2.4.3 that states a flood is not a design basis event, is correct.

ISA-17 Section 3.3.2.2.6.2, p. 3.3-28

Clarify if the rainfall load resulting from the Design Basis Flood (the load equals the depth of water accumulated in excess of the roof drains capability) will be addressed in designing the safety significant areas by ensuring this load does not exceed the normal roof design live load or if this rainfall load will be treated as an additional design load.

Under 10 CFR 70.62(c)(iv), an applicant is required to address potential accident sequences caused by external events, including natural phenomena. The Standard Review Plan, NUREG-1520, on page 3-12, (1)c, states that characterization of natural phenomena (e.g., tornadoes, hurricanes, floods, and earthquakes) and other external events is needed to assess their impact on facility safety and to assess their likelihood of occurrence.

The staff requires the additional information to properly consider the roof design loads.

LES Response

Depending on the final roof configuration, it may not be possible to limit the roof load due to the Design Basis Flood (DBFL) to the normal roof design live load, nor will the load be treated as an additional design load. The roof loads incurred due to the effects of a DBFL would be accounted for by using an additional load combination similar to load combination no. A.10 (discussed in SAR Section 3.3.2.2.8.3), substituting DBFL for Wt, similar to the recommendations of ACI 349-90 "Code Requirements for Nuclear Safety Related Concrete Structures," dated 1990, Chapter 9, paragraph 9.2.7.

U = D + F + L + H + T + Ro + DBFL

ISA-18 Sections 3.3.2.2.7.1, p. 3.3-29; 3.3.2.2.7.2, p. 3.3-29; and 3.3.2.2.7.4, p. 3.3-29

The equipment, piping, and electrical tray loads are given in the Safety Analysis Report as the sum of dead and live loads; no individual values are provided for these dead and live loads. Explain how the combined dead and live loads will be included in the load combination applications using the strength method for concrete design, given the load factors for dead loads and live loads are different (load combination applications A and B in Section 3.3.2.2.8.3 of the Safety Analysis Report).

Under 10 CFR 70.62(c)(iii), an applicant is required to address facility hazards that could affect the safety of licensed materials and thus present an increased radiological risk. The Standard Review Plan (NUREG-1520) on page 3-13, (3)c, states that process design and equipment information needs to include a discussion of process design, equipment, and instrumentation that is sufficiently detailed to permit an adequate understanding of the results of the ISA. As appropriate, it includes schematics indicating safety interrelationships of parts of the process.

The staff requires the additional information to ensure that the equipment, piping, and electrical tray loads have been properly considered.

LES Response

For equipment loads, piping loads, and electrical tray and conduit loads, the load values provided are "minimum" loads expected to be used. For the purpose of establishing the minimum load value, all loads are assumed to be live loads. This is a conservative approach since the factor for live loads is higher than the factor for dead loads. During final design, the individual loads will be identified and the appropriate load factors for live and dead loads (from the equations in SAR Section 3.3.2.2.8.3.A and B) will be applied to the individual loads. The resulting actual load, with the appropriate load factors applied, will be compared to the "minimum" load value with the load factor for live loads applied. Where the factored actual load value exceeds the factored minimum load value, the factored actual load value will be used in the analysis.

ISA-19 Section 3.3.2.3, p. 3.3-33

Provide technical justification to support that allowable bearing pressure for rock at the site is 10,000 psf and it is 3,000 psf for existing and new fills.

Under 10 CFR 70.62(c)(iii), an applicant is required to address facility hazards that could affect the safety of licensed materials and thus present an increased radiological risk. The Standard

Review Plan (NUREG-1520) on page 3-13, (3)c, states that process design and equipment information needs to include a discussion of process design, equipment, and instrumentation that is sufficiently detailed to permit an adequate understanding of the results of the ISA. As appropriate, it includes schematics indicating safety interrelationships of parts of the process.

Due to the difference in the allowable bearing pressures, the staff requires a technical justification that is currently not provided in Section 3.3.2.3 of the Safety Analysis Report.

LES Response

The 10,000 lb/ft^2 for insitu rock is a typographical error. In the next revision to the SAR it will be revised to read 7,000 lb/ft^2 for insitu firm and dense sands.

The 7,000 lb/ft^2 allowable bearing pressure was determined from a limited geotechnical testing program performed by MACTEC Engineering and Consulting, Inc. of Knoxville, TN. Soil test boring records are shown in Figures 3.2-10 thru 3.2-15 of the SAR. The 3,000 lb/ft^2 allowable bearing pressure is the value recommended by MACTEC for column loads up to 300 kips.

Rock or undisturbed insitu firm and dense sands are the result of hundreds of thousands of years of densification and cementing that can occur in nature. Existing fill soils are more recent and new fill soils will be even more recent. Often, fill soils cannot be densified/compacted to the same level as insitu soils by standard or economical means of placement and compaction, therefore the allowable bearing pressure for fill soils is set at a lower, more attainable number. It is anticipated that the foundations for all critical buildings for this project will be founded on the insitu firm and dense sands. All of this information will be verified by an extensive geotechnical investigation to be performed prior to the beginning of project design. New fill will be placed and compacted in accordance with procedures determined by a geotechnical engineer and tested after placement for proper compaction.

ISA-20 Section 3.4, General

The regulations, 10 CFR 70.22(a)(7), require that the applicant provide a description of equipment and facilities which will be used to protect health and minimize danger to life or property. The following information is needed to evaluate the instrumentation and control (I&C) systems:

a. Submit an I&C software system architecture block diagram showing interrelationship of the major software functions with the hardware, process, and plant systems. Clearly identify which software functions are involved with IROFS. Submit codes and standards framework used and correlate with hardware functions.

b. For IROFS involving software, firmware, microcode, etc., discuss the software design process used to develop the programmable logic controller (PLC) software, as well as software quality assurance programs, including configuration management. Reference any codes and/or consensus standards regarding hardware and software quality (e.g., IEEE, ASME).

c. For all systems with interfaces to the plant control system (PCS), describe the interfaces to the PCS, including the Central Control System (CCS) and the Local Control System (LCC). Particular attention should be paid to the interconnection of IROFS to the LCC (and CCS, if applicable). Provide information on how the safety functions are independent from the process control system components at the LCC and CCS.

d. Section 3.1.7.J states, in part, "Instrumentation and control systems shall be designed to fail into a safe state or to assume a state demonstrated to be acceptable on some other basis if conditions such as disconnection, loss of energy or motive power, or adverse environments are encountered." For IROFS relying on "fail safe" instrumentation, describe the conditions that cause a safe failure and how these conditions are sensed and corrected/masked by the "fail safe" function in the IROFS. Explain how this conforms to Section 3.1.7.J by describing the implementation of the "fail safe" capability (such as on-board diagnostics and/or condition monitoring) and the kinds of failures against which the design protects (such as random failures, circuit failures, software failures, malicious failures, etc.).

e. For IROFS involving instrumentation, provide information regarding the approach used to determine the setpoint and the measurement uncertainties. Account for all uncertainties in the measurement path, from the sensor along the signal lines through the data acquisition and data conversion components to the data processing element, as appropriate.

f. Section 3.1 states, "When failure probabilities are required for an event, Table 3.1-10, Failure Probability Index Numbers, provides the index values." Section 3.8, Table 3.8-1, provides failure probability index numbers for the IROFS. For IROFS involving instrumentation and control equipment relying upon both hardware and software, describe the process(es) (including testing, analysis, and/or industry experience) that was (were) used to establish the failure probabilities (i.e., Probabilities of Failure on Demand in Table 3.1-10 and Table 3.8-1). In the discussion, clearly explain how equipment and or processes from vendors (such as Urenco, and other third-party equipment suppliers) were evaluated by LES. Provide criteria and or data upon which values in Table 3.8-1 were based (for those IROFS involving hardware and software). Reference applicable consensus standards, if applicable.

g. The regulations, 10 CFR 70.64(a)(10), require that the design must provide for inclusion of instrumentation and control system to monitor and control the behavior of IROFS. Section 3.1.7.J in the Safety Analysis Report states, in part, "Instrumentation and control systems shall be provided to monitor variables and operating systems that are significant to safety over anticipated ranges for normal operation, for abnormal operation, for accident conditions, and for safe shutdown." Describe these instrumentation and control systems and how they meet the requirements of 10 CFR 70.64(a)(10). Include reference to codes or consensus standards, if applicable.

h. The regulations, 10 CFR 70.64(a)(4), state the design must provide for adequate protection from environmental conditions and dynamic effects associated with normal operation, maintenance, testing, and postulated accidents that could lead to loss of safety functions. These include Electromagnetic Interference/Radio Frequency Interference, temperature, and humidity. Section 3.5.9.1, p. 3.5-43, states "field-proven designs fabricated from proven materials for intended...operating conditions are specified, as well as process instrumentation qualified for use in uranium enrichment plants." For IROFS utilizing instrumentation, describe how the design complies with 10 CFR 70.64(a)(4). Reference applicable consensus standards, if applicable.

LES Response

a. A system architecture block diagram showing interrelationships of the major software functions with typical hardware, process, and plant systems is provided in ISA-20a in Attachment 3 to this letter.

Text removed under 10 CFR 2.390.

b. Currently, the IROFS are defined at the functional level. Upon completion of the design of each of the IROFS, the IROFS boundaries will be defined. In defining the boundaries for each IROFS, LES procedure DP-ISA-1.1, "IROFS Boundary Definition," will be used. This procedure requires the identification of each of the components necessary to ensure the IROFS is capable of performing its specified safety function, including support systems and components. As such, if after final design any software, firmware, microcode, PLCs, etc., is used for IROFS it will be identified at that time. For IROFS identified as using software, firmware, microcode, PLCs, etc., the applicable guidance of the following regulatory guides, including the endorsed IEEE standards, and industry standards will be used for implementing these features.

1) 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 for Computer Program Testing. (Refer to SAR Chapter 11, Appendix A, Section 3)

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

3) EPRI Topical Report (TR) -102323, Guidelines for Electromagnetic Interference Testing in Power Plants, Revision 1, December 1996.

4) EPRI TR-106439, Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications, October 1996.

5) Regulatory Guide 1.152, Criteria for Digital Computers in Safety Systems in Nuclear Power Plants, Revision 1, January 1996.

6) Regulatory Guide 1.168, Verification, Validation, Reviews, and Audits for Digital Software Used in Safety Systems of Nuclear Power Plants, Revision 1, February 2004.

7) Regulatory Guide 1.169, Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants, September 1997.

8) Regulatory Guide 1.170, Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants, September 1997.

9) Regulatory Guide 1.172, Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants, September 1997.

10) Regulatory Guide 1.173, Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants, September 1997.

c. Currently, the IROFS are defined at the functional level. SAR section 3.8.1 states that 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 and that control systems will not be used to perform IROFS functions. Upon completion of the design of each of the IROFS, the IROFS boundaries will be defined. In defining the boundaries for each IROFS, LES procedure DP-ISA-1.1, "IROFS Boundary Definition," will be used. This procedure requires the identification of each of the components necessary to ensure the IROFS is capable of performing its specified safety function, including electrical separation and isolation between IROFS systems and any interfaces with the process control system that might be present in the final design. This procedure also requires identification of the management measures necessary to ensure that the IROFS availability and reliability are maintained consistent with the assumptions of the ISA (as will be described in SAR Section 3.1.8.3 -- refer to response to RAI ISA-43).

d. SAR section 3.8.1 states that IROFS systems will be designed to be fail-safe. Fail-safe is a term applied to a process or component for which removal of a utility results in the process or component to fail in a safe configuration. Upon completion of the design of each of the IROFS, the IROFS boundaries will be defined using LES procedure DP-ISA-1.1, "IROFS Boundary Definition." This procedure requires the identification of each of the components necessary to ensure the IROFS is capable of performing its specified safety function, including defining the method or device used to satisfy the fail-safe criteria. This procedure also requires identification of the management measures necessary to ensure that the IROFS availability and reliability are maintained consistent with the assumptions of the ISA (as will be described in SAR Section 3.1.8.3 -- refer to response to RAI ISA-43).

e. SAR Section 3.8.1 states that the calibration of IROFS will be consistent with setpoint calculations as applicable. For hardware IROFS involving instrumentation which provides automatic prevention or mitigation of events documented in the ISA Summary, setpoint calculations will be 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. This commitment will be reflected in the next revision to the SAR. The setpoint methodology will describe the approach to determine the applicable IROFS trip setpoints and associated measurement uncertainties. These setpoint calculations will account for all uncertainties in the measurement path, from the process and sensor along the signal lines through the data

acquisition and data conversion components to the data processing element, as appropriate.

f. The guidance provided in NUREG-1520, Section A.5, Risk Index Evaluation Summary, was used to develop the NEF risk index methodology described in SAR Section 3.1.1.4. This methodology was applied for each accident sequence that could lead to high or intermediate consequences of concern. In applying this methodology, two parameters are required for each IROFS: (1) the Failure Probability Index Number, and (2) the Failure Duration Index Number.

   The Failure Probability Index Numbers for the IROFS presented in Table 3.8-1 were not based on the historical failure data for the facility equipment. The Failure Probability Index Number for each IROFS was assigned based on the type of IROFS using the data provided in SAR Table 3.1-10. The data in this SAR table is consistent with the guidance provided in NUREG-1520, Table A-10. The Failure Duration Index Number for NEF was assigned based on the specific reliability management measures applied to the IROFS and the data provided in SAR Table 3.1-11. The data in SAR Table 3.1-11 is consistent with the guidance provided in Table A-11 in NUREG-1520. The application of this methodology for the NEF is consistent with the guidance provided in NUREG-1520. An example of the application of this methodology for an IROFS from NEF is provided below.

   For IROFS2, an active engineered control (AEC), the Failure Probability Index Number was assigned a "-2" from Table 3.1-10. The value of "-2" was selected by taking the lower range of the values (i.e., "-2 to -3") assigned to a single active engineered IROFS (AEC). The Failure Duration Index Number of "0" from SAR Table 3.8-11 was then assigned for the IROFS2 based on an annual test frequency (as will be described in SAR Section 3.1.8.3 -- refer to response to RAI ISA-43). The annual test frequency assures that an undetected failure of IROFS2 would not exceed 1 year.

   Once the NEF is operating, failure data for the IROFS will be trended and the impact of this failure data on the values assumed in the ISA will be evaluated.

g. For hardware IROFS involving instrumentation that provides automatic prevention or mitigation of events documented in the ISA Summary, status and operation will be monitored by the PCS by means of an alarm. This alarm will be provided by an isolated, hardwired digital signal from the associated IROFS to the PCS PLC. This signal will only be directed from the associated IROFS to the PCS PLC. The required isolation is provided at the IROFS hardware interface in the process equipment for the connections to the PCS PLC. Consistent with ANSI/IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," the isolation devices will be classified as part of the IROFS boundary and will be designed such that no credible failure at the output of the isolation device shall prevent the associated IROFS from meeting it specified safety function. This commitment will be reflected in the next revision to the SAR.

h. SAR Section 3.8.1 states that the 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, as required by the ISA. The listed conditions in SAR Section 3.8.1 are examples of environmental conditions for which the IROFS must be qualified. Although it was not intended to be an all-inclusive list, in the next revision to SAR Section 3.8.1, the statement will include electromagnetic interference (EMI) and radiofrequency interference (RFI) in the list of environmental conditions for which the IROFS must be qualified. As such, the IROFS components and systems will be designed, procured, constructed, and maintained such that environmental conditions and dynamic

effects associated with normal operations, maintenance, testing, and postulated accidents will not lead to a loss of the IROFS safety function, including EMI and RFI. To accomplish this, 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. Documentation of this evaluation of each piece of equipment of IROFS components and systems will be used to demonstrate that the IROFS components and systems will perform their safety functions under environmental and dynamic service conditions in which they will be required to function.

ISA-21 Sections 3.4, 3.5, 3.8, and Table 3.8-1

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In defining the boundaries for each IROFS (which requires completion of the design of IROFS), including IROFS15 and IROFS16, LES procedure DP-ISA-1.1, "IROFS Boundary Definition," will be used. This procedure requires the identification of each of the components necessary to ensure the IROFS are capable of performing their specified safety function. This procedure also requires identification of the management measures necessary to ensure that the IROFS availability and reliability are maintained consistent with the assumptions of the ISA (as will be described in SAR Section 3.1.8.3 -- refer to response to RAI ISA-43). For IROFS15 and IROFS16, the management measures will include the performance of a functional test and calibration of the associated instrumentation annually.

ISA-30 Section 3.4.3.3, p. 3.4-16

Describe the safety margins needed to assure that loads resulting from a centrifuge failure do not result in rotor debris penetration of the casing or break away of the floor mounting elements (flomels). Identification of specific industry codes or standards or operational test results is acceptable.

The regulations, 10 CFR 70.22(a)(7), require a description of the equipment and facilities which will be used to protect health and minimize danger to life or property.

Section 3.4.3.3, Design Description [Cascade System], states that the resultant loads from centrifuge failures are restrained by the casing and the floor mounting element. These components are designed so rotor debris does not penetrate the casing and the flomels do not break away from the floor.

LES Response

The centrifuge and its mounting system, including the bolts, are designed to withstand single and multiple crashes (i.e., centrifuge rotor failures). As part of planned testing and qualification, single centrifuges were crash-tested under controlled laboratory conditions that included the following enhancements.

- Specially adapted components were provided that resulted in a higher than normal stress on both the centrifuge recipient and its mounting system.
- Extensive instrumentation was provided to measure the strain in the centrifuge and its mounting system bolts.

In addition, the crash-tests were carried out at higher than normal operating frequencies.

After the completion of the crash tests, the data were analyzed using the results from the strain measurements to determine the location and magnitude of the maximum stresses to ensure that expected safety factors were not exceeded.

Additional details are provided in the original Urenco TC-12 centrifuge mechanical qualification report, "TC-12 Safety Report, Safety of the Recipient Against Penetration of Rotor Debris, Safety of the Floor Mounting System Against Rupture, JVDOC300/21, Issue 2, June 1980." This classified (i.e., confidential national security information (CNSI)) report was provided to the NRC during the Claiborne Enrichment Center licensing proceedings. The original design reflected in this report has since been modified. As a result, proprietary report UPD/0202662B, containing a description of the modification and associated qualification, supplements the original Urenco TC-12 centrifuge mechanical qualification report. Proprietary report

UPD/0202662B was submitted to the NRC by letter NEF#04-008 dated May 7, 2004, from R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC).

In addition to extensive testing and qualification, Urenco had over 10 years of operating experience with the Model TC-12 centrifuges with no recorded failures of their mounting systems following a statistically significant number of crashes.

ISA-31 Section 3.4.4, pp. 3.4-18 through 3.4-26

Provide a list of light and intermediate weight gases that could be trapped in cold traps. Are any of these gases explosive or combustible alone or in combination with other light or intermediate weight gases?

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, 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.

The discussion of cold traps in the product take-off system does not provide information on potentially combustible or explosive gases that might be collected in the cold traps.

LES Response

The following gases could be expected to be present: uranium hexafluoride, air, hydrogen fluoride, and trace quantities of gases such as nitrogen and fluorine.

There are no individual vapors/gases or combinations of vapors/gases expected to form in the cold traps that would be explosive, flammable, or combustible based on the constituents in the system during normal operation.

ISA-32 Section 3.4.9, pp. 3.4-54 through 3.4-62; Section 4.6.1, pp. 4.6-1 through 4.6-2

Provide information related to codes and standards for GEVS design and in-place filter testing.

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

In Section 3.4.9 of the application, the applicant indicates that a prefilter with an efficiency of 85 percent and a charcoal filter with an efficiency of 99.9 percent will be used. However, no reference to testing standards are provided. The staff assumes the High Efficiency Particulate Air (HEPA) filter efficiency is based on removal of 0.3 micron particles and will meet the requirements of American Society of Mechanical Engineers (ASME) AG-1, "Code on Nuclear Air and Gas Treatment," Section FC.

Section 3.4.9 indicates that gas monitors are provided to continuously monitor effluents from the GEVS. What are the sensitivities of the gamma and HF monitors?

In Sections 3.4.9 and 4.6.1 of the application, the applicant describes the ventilation program and air cleaning systems. However, reference is not made to the most current ventilation system design standards in ASME AG-1. Will filtration systems be designed in accordance with ASME AG-1?

In Section 4.6.1 of the application, the applicant states that filter inspection and testing will be performed in accordance with written procedures. A general statement referring to ASME N510, "Testing of Nuclear Air-Cleaning Systems," is made. However, no specific information is provided on in-place filter testing frequencies or leakage efficiency goals for HEPAs or for the charcoal adsorbers.

Sections 3.4.9 and 4.6.1 of the application do not discuss temperature instrumentation downstream of the filter assemblies to detect high temperatures in the event of filter unit fires. Do temperature monitors with alarms and the capability to shut down fans exist in the system? If not, justify why this instrumentation is not included.

LES Response

The filtered ventilation systems at NEF, i.e., Separations Building GEVS, the Technical Services Building GEVS, the Centrifuge Test and Post Mortem Facilities Exhaust Filtration System, and the confinement function of the TSB Heating, Ventilation, and Air Conditioning System (HVAC), are not credited for the prevention or mitigation of any event documented in the ISA Summary. The design and in-place testing from these filtration systems will be consistent with the applicable guidance in Regulatory Guide 1.140, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units for Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Revision 2, dated June 2001, ASME AG-1-1997, and ANSI N510-1989. The NEF filtrations systems include potassium carbonate impregnated activated charcoal filters for hydrogen fluoride (HF) removal. As such, the portions of Regulatory Guide 1.140, ASME AG-1, and ANSI N510, which address activated charcoal filters for radioiodine removal are not applicable.

The prefilter efficiency (85%) is based on testing in accordance with ASME AG-1-1997. The HEPA filter efficiency (99.97%) is based on removal of 0.3 micron particles when tested in accordance with ASME-AG-1. The impregnated charcoal filter efficiency (99.9%) for removal of HF is based on the Urenco reference design documents. In-place testing and inspections of the filters will be performed in accordance the guidance in Regulatory Guidance 1.140. The frequency for performance of in-place filter testing and the acceptance criteria for penetration and leakage (or bypass) will be consistent with the guidance in Regulatory Guide 1.140. Laboratory testing of the impregnated charcoal filter of charcoal samples will be performed on an annual basis. Throughout the useful life of the impregnated charcoal, the impregnate is progressively consumed. The laboratory testing will determine the impregnant content within the sample. The amount of impregnant present in the sample is indicative of the remaining life of charcoal bed for removal of HF.

The gamma monitors to be used in the NEF Separation Building GEVS and the TSB GEVS will be selected during final design. As such, the sensitivity of these gamma monitors has not been specified. However, the gamma monitors to be supplied will be suitable, with appropriate sensitivity, and qualified for use in the environment in which they are required to function.

It is expected that the HF monitors to be used in the NEF Separations Building GEVS and the TSB GEV will be similar to typical HF monitors used in other Urenco facilities. The associated parameter values for one of these typical HF monitors are as follows.

Range:	0.04 to 50 mg/m^3
Lowest Detection Limit:	0.04 mg/ m^3

During final design, temperature monitors will be provided for detecting high temperatures that would be indicative of a filter unit fire in the charcoal filters of the NEF filtered ventilation systems. These temperature monitors are not credited for meeting 10 CFR 70.61, "Performance requirements," and therefore do not need to be IROFS.

ISA-33 Section 3.5.1, pp. 3.5-1 through 3.5-12

Justify the lack of air effluent monitoring in areas where dispersible forms of uranium are stored or processed, which are not serviced by filtered exhaust systems with continuous monitoring.

10 CFR 20.1501 requires that surveys be made to measure the levels of radioactive material and the potential radiological hazards.

Further, NRC Regulatory Guide 4.16, Regulatory Position C.2., states "Gaseous effluents from all operations associated with the plant, including such nonprocessing areas as laboratories, experimental areas, storage areas, and fuel element assembly areas, should be sampled. For gaseous effluents from process confinement systems and process areas where material is handled in dispersible form, a representative sample of the effluent from each stack, vent, or other point of release should be collected continually for subsequent determination of quantities and average concentrations of radionuclides released. This sampling should be conducted regardless of the concentrations of radioactive material in the effluent."

In the Environmental Report, Table 6.1-1, "Effluent Sampling Program," the applicant proposes to sample process areas only as required to complement the bioassay program. Presumably, the samplers referenced in Table 6.1-1 are those described in Section 4.8.1.2 of the Safety Analysis Report. However, these samplers would not be sufficient to permit a determination of the quantities of radionuclides and the average concentration of radionuclides being discharged from the plant.

Areas of specific concern to the staff include: (1) the Blending and Liquid Sampling Area; (2) Process Services Corridors (3 modules); (3) Link Corridors (3 modules); (4) UF6 Handling Areas (3 modules); (5) Vacuum Pump Rebuild and ME&I Workshops; (6) Chemical and Mass Spectrometry Lab and Environmental Laboratory; (7) and the Cylinder Receipt and Dispatch Building.

LES Response

During the final design phase, facilities will be evaluated using the guidance provided in Regulatory Guide 4.16, "Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Plants," Revision 1, dated December 1985. Using the results of this evaluation, periodic sampling or continuous sampling provisions, as appropriate, shall be implemented in accordance with the guidance in Regulatory Guide 4.16.

The Blending and Liquid Sampling Area, Process Services Corridors (3 modules), Link Corridors (3 modules) and UF_6 Handling Areas (3 modules) are process areas or areas where material is handled in dispersible form. Therefore, in accordance with Regulatory Guide 4.16, during the final design phase provisions will be made for a representative sample from the effluent release point for these areas to be collected continually.

The Chemical Laboratory is used to analyze solid and liquid samples taken from all areas of the plant. The Mass Spectroscopy Laboratory is used to measure the isotopic abundance of various uranium isotopes in prepared liquid samples. These samples are analyzed using an inductively coupled plasma-mass spectrometer (ICP-MS). As such, the gaseous effluents from these two laboratories will be monitored. The monitoring will be performed using the TSB GEVS. The TSB GEVS provides filtered exhaust for potentially hazardous contaminants via fume hoods for the Chemical Laboratory and Mass Spectrometry Laboratory. The gaseous argon effluent from the ICP-MS is also routed to the TSB GEVS. The TSB GEVS provides for continuous monitoring and sampling of the gaseous effluent in the exhaust stack in accordance with the guidance in Regulatory Guide 4.16.

The Vacuum Pump Rebuild Workshop is designed to provide space for the maintenance and rebuilding of plant equipment, mainly pumps which have been decontaminated prior to placement in the workshop, and other miscellaneous plant equipment. The TSB GEVS provides filtered exhaust for potentially hazardous contaminants via fume hoods for the Vacuum Pump Rebuild Workshop. The TSB GEVS provides for continuous monitoring and sampling of the gaseous effluent in the exhaust stack in accordance with the guidance in Regulatory Guide 4.16.

The Environmental Laboratory is used to prepare and analyze samples associated with safety or regulatory compliance. Typical samples are water, urine, gaseous effluents, soils, and grass. The radioactivity levels associated with these samples are very low and do not present a radiological concern. In accordance with Regulatory Guide 4.16, Regulatory Position C.2.1, periodic sampling or other means will be used to show that the radioactivity in the effluents from this laboratory is insignificant. The effluents from this laboratory will be sampled at least quarterly. It will be demonstrated that these samples are representative of actual releases from the Environmental Laboratory.

All UF_6 feed cylinders, empty product cylinders, uranium byproduct cylinders and final product cylinders enter or leave the facility through the Cylinder Receipt and Dispatch Building (CRDB). This building is designed to include space for loading, unloading, storage, and weighing of cylinders. In accordance with Regulatory Guide 4.16, Regulatory Position C.2.1, periodic sampling or other means will be used to show that the radioactivity in the effluents from the CRDB is insignificant. The effluents from the CRDB will be sampled at least quarterly. It will be demonstrated that these samples are representative of actual releases from the CRDB.

The ME&I Workshop is designed to provide space for the normal maintenance of uncontaminated plant equipment. There are no process confinement systems in the workshop and no radioactive material in dispersible form is handled in the area. Therefore, no radioactivity is present in the effluents from the ME&I Workshop. Nevertheless, during the final design phase, the ME&I Workshop will be evaluated using the guidance provided in Regulatory Guide 4.16. The guidance contained in the Regulatory Guide will be implemented as appropriate.

ISA-34 Section 3.5.1, pp. 3.5-1 through 3.5-12

Provide room volumes, room volumetric flow, and Heating Ventilating and Air Conditioning (HVAC) exhaust flow rates for each likely configuration of the HVAC systems described in Section 3.5.1 of the Safety Analysis Report.

10 CFR 70.65(b)(3) states that the ISA Summary must contain "a general description of the facility with emphasis on those areas that could affect safety."

The acceptance criteria in Standard Review Plan section 3.4.3.2(3), Processes, states that a description at a systems level is acceptable, provided that it permits the NRC reviewer to adequately evaluate (1) the completeness of the hazard and accident identification tasks and (2) the likelihood and consequences of the accidents identified.

The staff requires room volumes, room volumetric flow, and HVAC exhaust flow rates to independently evaluate the consequences of the accidents identified in the ISA Summary.

LES Response

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ISA-37 Section 3.5.12.1.1.8, p. 3.5-58

Explain the means by which a representative sample is collected from the Treated Effluent Monitor Tanks.

The regulations, 10 CFR 20.1302, requires appropriate surveys and measurement to be conducted to demonstrate that dose limits are met. NRC Regulatory Guide 4.16, Regulatory Position C.2.2, states "Representative samples should be collected at each liquid release point for the subsequent determination of the quantities and average concentrations of radionuclides discharged in any liquid effluents that could reach an unrestricted area, including discharges to a sanitary sewerage system."

LES Response

Liquids are treated by precipitation and evaporation before entering the Treated Effluent Monitor Tanks. If necessary, additional treatment using mixed bed demineralization is available. The treatment system (precipitation and evaporation) is designed for high uranium removal efficiency. As such, the concentration of uranic material in the Treated Effluent Monitor Tanks is expected to be low. Sample collection from the Treated Effluent Monitor Tanks was intended to be from the bottom of the tanks, which should conservatively estimate sample concentration. However, Regulatory Guide 4.16, "Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Plants," Revision 1, dated December 1985, states that representative samples of each batch of liquid released should be collected. Accordingly, during final design, the Treated Effluent Monitor Tanks will be provided with mechanical agitators or recirculation capability to ensure samples from the tanks are representative.

ISA-38 Section 3.5.12.1.4, p. 3.5-60

Describe the bookkeeping measures needed to ensure that no tank holds more than a safe mass of uranium.

10 CFR 70.61(a) requires each applicant to evaluate, in the ISA performed in accordance with §70.62, its compliance with the performance requirements in paragraphs (b), (c) and (d) of this section.

The application states that bookkeeping measures ensure that no tank holds more than a safe mass of uranium. In Section 3.5.12.1.5, the applicant states that the uranium content of tanks is important to prevent a criticality accident. None of the tanks in the collection and treatment system are "geometrically safe" or "geometrically favorable". Administrative controls (by mass) are applied to prevent a criticality accident. Additional information on the bookkeeping measures is needed to assess the effectiveness of this provision.

LES Response

The criticality safety for tanks that are not "geometrically safe" or "geometrically favorable" will utilize two independent IROFS for mass control -- one of which is referred to as "bookkeeping measures" in SAR Section 3.5.12.1.4 (the second IROFS is referred to as "sampled and analyzed"). The "bookkeeping measures" is a process to calculate the potential mass of uranium in the tank for any batch operation. 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.

For NEF, the "bookkeeping measures" are only applied to tanks where the mass of uranium involved, even when double batching error is considered, is far below the safe value. Bookkeeping measures are a documented running inventory estimate of the total uranium mass in a particular tank. The mass inventory for each batch operation is calculated based on the mass of material to be transferred during each batch operation and the mass inventory in the tank prior to the addition of the material from the batch operation.

There are two types of batch operations that are considered. The first is liquid transfer between tanks based on moving a volume of liquid with uranic material present in the volume. The second is transferring a number of components into the tank with the uranic material contained within or on the components transferred in each batch operation. For both types of operations, the initial mass inventory is conservatively set after emptying, cleaning, and readying the tank for receipt of uranic material. For each batch operation, the amount of uranic material to be transferred during a particular batch operation is conservatively estimated. This quantity of material is then credited/debited to/from each tank as appropriate. A new mass inventory in each tank is calculated. The receiving tank mass inventory is then compared to the mass limit for the tank.

For the second type -- a transfer of a number of facility components into a tank during a batch operation, the mass inventory on/within the components is conservatively estimated, and that mass is credited to the receiving tank. The final mass inventory in the tank is calculated and the total is compared to the mass limit for the tank.

Although not necessary for IROFS performance as documented in the ISA, one feature of the audit and assessment management measure for these IROFS is that the estimated mass inventory at any time is also compared to the final mass determined based on the "sampled and analyzed" method and any divergence beyond a conservatively established limit of the two methods will require investigation in accordance with the corrective action process.

ISA-39 Section 3.5.15, pp. 3.5-79 through 3.5-81

Provide the combustion characteristics of Fomblin oil (flash point, fire point, heat of combustion, etc.)

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.

The discussion in the Safety Analysis Report of Fomblin oil does not state that this oil is noncombustible nor does it provide any discussion of potential fire hazards presented by the oil.

LES Response

Fomblin oil is noncombustible. If exposed to fire, Fomblin oil will begin to decompose at temperatures above 290°C with the evolution of carbonyl fluoride and hydrogen fluoride, but it will not burn.

A specification sheet and/or MSDS of the specific manufacturer's oil brand to be used will be available at the completion of final design.

ISA-40 Section 3.5.17, p. 3.5-84

Text removed under 10 CFR 2.390.

ISA-41 Section 3.7.1, Table 3.7-2 and Section 3.7.2, Table 3.7-3

Identify whether the environmental performance requirement in 10 CFR 70.61(c)(3) was met for each of the events described in the ISA Summary.

10 CFR 70.65(b)(4) requires that the ISA Summary contain information that demonstrates the licensee's compliance with the performance requirements of 10 CFR 70.61.

NRC acceptance criteria in Standard Review Plan, Section 9.4.3.2.3 states that the applicant's ISA is acceptable if adequate engineering or administrative controls are identified for each accident sequence of environmental significance. However, in the ISA Summary, the applicant did not indicate whether the performance requirement in 10 CFR 70.61(c)(3) was met for each of the events summarized in Tables 3.7-2 and 3.7-3.

LES Response

The ISA originally evaluated the environmental performance requirement from 10 CFR 70.61(c)(3) at the site boundary fence. The calculated 24-hour average uranium concentration was below the Category 2 Intermediate Consequence as listed in SAR Table 3.1-3 for all accident sequences. Therefore, results were not reported in the ISA Summary since the consequences were Category 1, Low Consequence.

LES has re-evaluated the environmental performance requirement and as a result has defined a revised Restricted Area for the NEF. Figure ISA-41, "Projected Radiological Zones," shown in Attachment 3 to this letter, shows this revised Restricted Area. The 24-hour averaged release of radioactive material outside the revised Restricted Area is below the concentrations exceeding 5,000 times the values in Table 2 of Appendix B to Part 20 for all accident sequences summarized in SAR Tables 3.7-2 and 3.7-3. Therefore, the revised Restricted Area meets the environmental performance requirement from 10 CFR 70.61(c)(3).

In the next revision to the SAR, Figure 4.7-2 will be revised to reflect this change.

<u>ISA-42 Sections 3.8.1 and 5.1.1 and Tables 3.7-1, 3.7-3, 3.7-4, 3.8-1 and 3.8-2</u>

Clarify what was meant by the Sole IROFS in Table 3.8-2.

10 CFR 70.65(b)(4) requires in the ISA Summary information to demonstrate compliance with the performance requirements of 10 CFR 70.61. 10 CFR 70.65(b)(8) requires in the ISA Summary a descriptive list that identifies all items relied on for safety that are the sole item preventing or mitigating an accident sequence that exceeds the performance requirements of 10 CFR 70.61.

For example, in nuclear criticality safety, given the requirements of 10 CFR 70.65(b)(8), the commitment to the double contingency principle in Chapter 5.0, and the RAI question regarding what an IROFS is (i.e., ISA-1), it is unclear how there could be any nuclear criticality safety sole item relied on for safety.

LES Response

The current SAR Chapter 3 contains a number of "sole IROFS" associated with criticality sequences, which creates a perception that "double contingency protection" may not be provided. For these IROFS, a number of components of the criticality safety features were combined into a sole IROFS, which provided sufficient protection to reduce the accident sequence likelihood to an acceptable value. However, the implicit details (which contained double contingency protection) were not presented in sufficient detail so that double contingency protection was evident.

In the next revision to SAR Chapter 3, LES will present these IROFS in a more fundamental fashion that will identify two IROFS for each criticality sequence. LES has performed a preliminary review of the accident sequences to gain confidence that this approach will result in

all criticality sequences protected by at least two IROFS – demonstrating adequate double contingency protection. Formal ISA Team methodology will be applied to produce the required documentation to support the SAR (ISA Summary) change.

For illustrative purposes, examples of the sole IROFS in the current SAR are presented with proposed revised IROFS in the tables below. Once formalized, the revised IROFS will be propagated throughout the appropriate SAR Tables 3.7-1, 3.7-2, 3.8-1, and 3.8-2 and any accompanying text, in the next revision to the SAR.

Additionally, in response to RAI ISA-46, the descriptive text for IROFS will be simplified to focus solely on the safety function item, and exclude descriptive text associated with management measures. Also, reflected in these illustrative tables, in response to several RAIs regarding management measures (e.g., ISA-43), the Table 3.8-1 Management Measures column will reflect only unique management measures associated with providing "enhanced" availability and/or reliability assumptions associated with a more negative risk index than standard for the IROFS classification. These anticipated SAR changes are also reflected in the following tables.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

ISA-43 Section 3.8.1, pgs. 3.8-1 - 3.8-2 and Tables 3.7-1, 3.7-3, 3.7-4, 3.8-1 and 3.8-2

Clarify why there appears to be no management measures associated with IROFS25, IROFS27, and eight other NCS IROFS.

10 CFR 70.65(b)(4) requires in the ISA Summary information to demonstrate compliance with the performance requirements of 10 CFR 70.61, including a description of the management measures.

All applicable management measures need to be applied to all IROFS.

LES Response

In the next revision to the SAR, LES will clarify that all management measures (as defined in 10 CFR 70.4) apply to all IROFS for the NEF. A generic description of each management measure will be added to SAR Section 3.1.8.3. This description for each management measure reflects the general requirements applicable to each IROFS identified in Table 3.8-1. The information provided under the column heading "Reliability Management Measures" in Table 3.8-1 will remove references to standard management measures that are consistent with the description in Section 3.1.8.3. Any management measure explicitly identified in the applicable column of Table 3.8-1 reflects specific requirements (which are consistent with the performance requirements assumed in the ISA documentation in order to establish the Failure Duration Index Number) that deviate from the general requirements described in this section (i.e., relaxed, enhanced, or additional requirements). The basis for deviation of each management measure listed in Table 3.8-1 is discussed in Section 3.8.3.

Additional detail regarding implementation of management measures for IROFS, and any items that may affect the function of IROFS (as well as non-IROFS management measures), will continue to be found in SAR Chapter 11.

ISA-44 Tables 3.7-1 and 3.7-2

Clarify whether the criticality event assumed in accident sequence EC4-2 results in an intermediate consequence to the worker.

10 CFR 70.61(a) requires each applicant to evaluate, in the ISA performed in accordance with §70.62, its compliance with the performance requirements in paragraphs (b), (c) and (d) of this section.

Table 3.7-2, Accident Sequence Descriptions, accident sequence EC4-2 states that this event is assumed to have an intermediate consequence to the worker and the public. However, Table 3.7-1, Accident Sequence and Risk Index, identifies this as a high consequence event.

LES Response

The Table 3.7-2 description for accident sequence EC4-2 will reflect "high" consequence in the next revision to the SAR. Table 3.7-1 properly identified this sequence as high consequence.

ISA-45 Tables 3.7-1, 3.7-3, 3.7-4, 3.8-1 and 3.8-2

Clarify whether it was your intent to rely upon the design of your facility in the license application in lieu of designating IROFS for components/equipment when evaluating accident sequences for compliance with 10 CFR 70.61.

10 CFR 70.62(a) requires the establishment and maintenance of a safety program to demonstrate compliance with the performance requirements of 10 CFR 70.61.
10 CFR 70.65(a) requires a description of this safety program to be submitted in the license application. 10 CFR 70.65(b)(4) requires in the ISA Summary information to demonstrate compliance with the performance requirements of 10 CFR 70.61.

In the criticality analyses that were performed to support the ISA, the applicant made certain assumptions (e.g., design included favorable geometry equipment) and concluded that certain event sequences would be prevented. These assumptions related to specific systems or components that were not identified as IROFS. In addition, Table 5.1-2 lists other specific design attributes of the facility that are also not identified as IROFS. The applicant has placed these design assumptions and attributes under the Configuration Management system whereby any changes would be specifically evaluated. The staff considers that these design attributes are fundamental to the application review, and, if changes are made, in addition to the Configuration Management controls, the changes would need to be submitted for staff review in a license amendment as required under 10 CFR 70.72(c)(1)(i). Under 10 CFR 70.72(c)(1)(i), no changes to the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel, without prior Commission approval, that have not been previously described in the ISA Summary.

LES Response

10 CFR 70.61, Performance requirements, requires only "credible" events to have IROFS identified. For "credible" accident sequences, IROFS have been designated as required to meet the requirements of 10 CFR 70.61. Designs that inherently preclude "credible" events of consequence would not be identified as IROFS. In the next revision to the SAR, Section 3.1.1.3.2 will include a more detailed definition of "credible." The definition will include detail from NUREG-1520, Section 3.4.3.2(9), Acceptance Criteria for the Definition of "Credible." Furthermore, a justification basis will also be included such that, for certain designs of favorable geometry, any postulated event may be of "negligible likelihood" allowing failure of the design to be deemed a "not credible" event. This additional portion of the definition is provided below.

> *Additionally, the determination of "not credible" can apply to permanently installed passive design features (e.g., tanks, piping) of the facility that do not rely on human interface to perform the safety function (i.e., termed "safe by geometry"). These features must also meet the criterion that the only credible means to effect a change that might result in a failure to function, would be to implement a design change (e.g., geometry deformation as a result of a credible event would not adversely impact the performance of the safety function). The evaluation of potential credible means (which would include items such as bulging, corrosion, breach of confinement and subsequent accumulation of material), includes consideration of adequate controls to ensure the double contingency principle is met. The configuration management system required by 10 CFR 70.72 (implemented by the NEF Configuration Management Program described in Chapter 11), adequately ensures maintaining the safety function and assures compliance with the double contingency principle, as well as the defense-in-depth criterion of 10 CFR 70.64(b).*

It is recognized that passive design features that are intermittently utilized by operator interface (e.g., carts, racks) have a human interface relationship that raises the potential event sequences to a "credible" category.

This definition of "not credible" is consistent with the NUREG-1520 (same Section cited above) discussion of "negligible likelihood" and "not credible."

Apart from the question regarding "favorable geometry" categorization, this RAI introduced an NRC Staff position on the application of 10 CFR 70.72, "Facility changes and change process" (which requires a licensee to establish a configuration management system). The following discussion is proposed to clarify the LES position on implementation of 10 CFR 70.72.

Whether or not a design attribute is identified as an IROFS, 10 CFR 70.72(c) lists the criteria that, if satisfied, would allow licensee to implement design changes <u>without</u> NRC prior review or approval.

(a) <u>Criterion of 10 CFR 70.72(c)(1)(i)</u> (*i.e., the subpart cited in RAI*): This criterion would require NRC pre-approval only if the change:

(1) Create new types of accident sequences <u>not</u> described in the ISA Summary, and

(2) Only if the new sequence requires mitigation or prevention to meet the performance requirements of 10 CFR 70.61.

Since LES does not present "favorable geometry equipment" as IROFS, they are not explicitly described in the ISA Summary, the 10 CFR 70.72(c)(1)(i) criterion would apply to any design changes to evaluate whether NRC prior approval is required. If a change to favorable geometry equipment were desired, such that mitigation or prevention becomes necessary (i.e., an event becomes "credible"), prior NRC approval would be required. If changes are evaluated to maintain the definition of "not credible," that change could be implemented without NRC prior approval.

However, if favorable geometry equipment were made IROFS, the process would be governed by 10 CFR 70.72(c)(2) and (3). Furthermore, since the favorable geometry is associated with criticality safety, it is expected that a second IROFS for the associated sequence(s) would be identified in the ISA Summary, as reflected in the LES response to RAI ISA-42.

(b) <u>Criteria of 10 CFR 70.72(c)(2) and (3)</u>: If the design attribute were listed as an IROFS (and therefore described in the ISA Summary):

(1) 10 CFR 70.72(c)(2) would allow the licensee to "remove" (without prior NRC review or approval) an IROFS that is listed in the ISA Summary provided there is "at least an equivalent replacement of the safety function." The licensee has the flexibility to include in the change new and/or enhanced safety functions that provide "equivalent" safety, and not require NRC prior approval.

(2) 10 CFR 70.72(c)(3) would allow the licensee to "alter" (without prior NRC review or approval) an IROFS that is listed in the ISA Summary provided it was not a sole IROFS. "Altering" favorable geometry equipment could also be possible under this provision without prior NRC approval.

If LES were to reorganize these favorable geometry design attributes to IROFS (and identify additional IROFS for the event sequence), a wider range of configuration management control would appear to be possible without prior NRC review or approval, as allowed by 10 CFR 70.72.

Additional management measure attributes, specific to these design features, are discussed further below. General Management Measures and QA Program attributes associated with personnel training and qualification, audits and assessments, and records also apply as discussed in SAR Chapter 11, "Management Measures," and its Appendix A, "QA Program Description."

Procedures, Surveillance Testing, Maintenance: In addition to the design change control portion of the configuration management system, all plant activities, including surveillance testing and maintenance activities of design features that have any potential to impact the favorable design attribute, will be performed only in accordance with written and approved procedures. Procedure approval configuration management controls will assure activities are not implemented that could create new types of accident sequences not described in the ISA Summary (per criterion of 10 CFR 70.72(c)(1)(i)).

In the next revision to SAR Section 11.4.4, "Changes to Procedures," the requirement for "criticality safety review" will be clarified (see also response to RAI ISA-2) to reflect the performance of a nuclear criticality safety evaluation for all changes with the potential to affect criticality safety. This independent criticality safety evaluation is developed and independently reviewed by qualified criticality safety engineers.

Degraded and Non-Conforming Conditions and Corrective Action Process: Identification of any nonconforming condition that potentially impacts a design feature will clearly identify and describe the characteristics that do not conform to specified criteria. Nonconformance documentation will be reviewed by the responsible affected organization and recommended dispositions of nonconforming items proposed in accordance with approved procedures. The review will include determining the need for additional corrective actions according to the requirements of Corrective Action Process. In addition, organizations affected by the nonconformance will be notified.

Reporting: Any "favorable geometry" design feature that is discovered to function outside its analyzed design basis, in conjunction with Corrective Action Process activities described above, will be reported to the NRC. 10 CFR 70, Appendix A, (b)(1), requires reporting to the NRC, with follow-up written report, any event or condition that results in the facility being in a state that was not analyzed, was improperly analyzed, or is different from that analyzed in the Integrated Safety Analysis, and which results in failure to meet the performance requirements of 10 CFR 70.61.

Based on the discussion above, it is concluded that requirements of the regulations, i.e., the configuration management and change process requirements of 10 CFR 70.72, provide adequate controls to ensure changes receive appropriate NRC prior review and approval.

ISA-46 Tables 3.7-1, 3.7-3, 3.7-4, 3.8-1 and 3.8-2

Clarify how IROFS relate to the accident sequences, other IROFS, and management measures.

10 CFR 70.65(b)(4) requires in the ISA Summary information to demonstrate compliance with the performance requirements of 10 CFR 70.61.

There needs to be a clear understanding of each IROFS. For example, in nuclear criticality safety: some of the IROFS appear to be several IROFS rolled up into a single IROFS (e.g., IROFS6, IROFS9, and others); some appear to be IROFS plus management measures (e.g., IFORS6, IROFS15, and others); and some appear to be programs (e.g., IROFS16 (moderator control) and IROFS19 (mass control)).

LES Response

In the next revision to the SAR, the IROFS description will be re-written such that the descriptive text for IROFS is simplified to focus solely on the safety function that is applied to a specific accident sequence. The management measures will be removed from the description of the IROFS in Table 3.8-1 and described generically in Section 3.1.8.3. In addition, the programmatic IROFS will be re-written to identify the specific controlling parameter or parameters, and the associated limit(s) that are being relied on for the safety function (at this time a descriptive limit as opposed to an explicit value) applied to the accident sequence. Examples of these anticipated changes, and additional clarifying changes are presented in the response to RAI ISA-42.

<u>ISA-47 Tables 3.7-1, 3.7-3, 3.7-4, 3.8-1 and 3.8-2</u>

Clarify the basis for the frequency index number (FFIN) for enhanced administrative controls and administrative controls.

10 CFR 70.65(b)(4) requires in the ISA Summary information to demonstrate compliance with the performance requirements of 10 CFR 70.61.

For example, in nuclear criticality safety, most active engineered controls have FFINs of -2 while most enhanced administrative controls and administrative controls have FFINs of -3, which is a more robust value. This appears to be inconsistent because one would expect an active engineered control to be more robust than either an enhanced administrative or administrative control.

LES Response

For passive or active engineered controls or administrative controls, the Failure Frequency Index Number (FFIN) was assigned based on the type of IROFS using the information provided in SAR Table 3.1-10. For example for an Active Engineered Control (AEC), the FFIN would be assigned a value of "-2." This value of "-2" is at the low range of the recommended values from this table. Similarly, an administrative IROFS for routine planned operations would also be assigned a value of "–2" (the low range of the recommended values).

The concept of making the FFIN more (or less) robust is presented in the Table 3.1-9, Footnote **. As such, conceptually, an administrative control could be adequately justified to allow an FFIN of "–3."

However, in the response to RAI ISA-42, LES committed rewrite criticality "sole IROFS" to clarify the controls being applied to these accident sequences. One result of this rewrite will be the elimination of all criticality sole IROFS. This will also eliminate the use of any of the *currently* defined "enhanced" administrative controls to prevent criticality accident sequences.

This issue of defining "enhanced" reliability controls is addressed in more detail in the LES response to RAIs ISA-43 and ISA-58. In these responses, the revised presentation committed

to apply general high-quality management measures to all IROFS. "Enhanced" management measures that may be identified in Table 3.8-1 would reflect specific requirements that deviate from the general management measure requirements (i.e., relaxed, enhanced, or additional requirements). These "enhanced" management measures would be consistent with the performance requirements assumed in the ISA documentation. This revised definition and revised use of "enhanced" controls is not reflected in any IROFS currently presented in Table 3.8-1.

ISA-48 Tables 3.7-1, 3.7-3, 3.7-4, 3.8-1 and 3.8-2

Clarify what was meant by listing only a few management measures under the column 'Reliability Management Measures' in Table 3.8-1.

10 CFR 70.65(b)(4) requires in the ISA Summary information to demonstrate compliance with the performance requirements of 10 CFR 70.61, including a description of the management measures.

All applicable management measures need to be applied to all IROFS; however, there are only a few or no management measures listed for each IROFS in Table 3.8-1. The following examples are the only descriptions of the 'Reliability Management Measures' from Table 3.8-1 for nuclear criticality safety accident sequence IROFS:

> Annual Inspection - IROFS14 and IROFS17
> Annual Test - IROFS3, IROFS8, IROFS13, IROFS20, IROFS21, and IROFS22
> Annual Test, Operator Training, and Annual Refresher - IROFS9
> Operator Training and Annual Refresher - IROFS6, IROFS15, IROFS16, IROFS18, IROFS19, IROFS45 , IROFSC1, IROFSC6, IROFSC7, and IROFSC14
> Personnel Training and Annual Refresher - IROFS40
> N/A - IROFS25, IROFS29, IROFS30, IROFS31, IROFS32, IROFS33, IROFS34, and IROFS44

LES Response

The presentation of the ISA methods employed to demonstrate the performance requirements of 10 CFR 70.61 currently in the NEF SAR, includes "sole IROFS," "enhanced administrative controls," "Reliability Management Measures," and an inconsistent presentation of applicable management measures. In response to various other RAIs (e.g., ISA-42, -43, -47, -59), portions of each topic are discussed. The following summary provides an overall description of related clarifications proposed to be included in the next revision to the SAR.

"Sole IROFS" for criticality sequences currently reflect multiple controls "rolled-up" into a single IROFS. These controls are typically presented as "enhanced administrative controls" and have specific "Reliability Management Measures" shown in Table 3.8-1. The next SAR revision will present these IROFS in a more fundamental fashion that will identify two IROFS for each criticality sequence. This will also eliminate the use of any of the currently defined "enhanced" administrative controls to prevent criticality accident sequences. Since the current presentation of sole IROFS contained multiple controls, and attempted to address the special preventive measures being applied for criticality control, this "packaging" of controls was referred to as "enhanced" and would typically be reflected with Failure Frequency Index Numbers more negative (i.e., "-3") than non-enhanced administrative controls. Furthermore, since the Reliability Management Measures currently shown in Table 3.8-1 do not convey a consistent

approach to management measures, use of this information also did not adequately convey the intent of "enhanced administrative controls."

To provide a consistent presentation of management measures, a clear commitment to apply each management measure defined in 10 CFR 70.4, and required by 10 CFR 70.62(d), to each IROFS will be included in the next revision to the SAR. This commitment and a generic description of each management measure will be added to SAR Section 3.1.8.3. The general description for each management measure reflects the minimum requirements applicable to each IROFS identified in Table 3.8-1. The information provided under the column heading "Reliability Management Measures" in Table 3.8-1 will remove references to standard management measures that are consistent with the description in Section 3.1.8.3. In the next revision to Table 3.8-1 any management measure explicitly identified in the "Reliability Management Measures" column will reflect specific requirements that deviate from the general management measure requirements (i.e., relaxed, enhanced, or additional requirements). These Reliability Management Measures will be consistent with the performance requirements assumed in the ISA documentation used to establish the Failure Frequency Index Number and/or Failure Duration Index Number. The basis for deviation of each management measure listed in Table 3.8-1 will be discussed in Section 3.8.3.

Formal ISA Team methodology will be applied to produce the required documentation to support the SAR (ISA Summary) changes summarized above.

ISA-49 Tables 3.7-1, 3.7-3, 3.7-4, 3.8-1 and 3.8-2

Correct information in the tables for consistency.

10 CFR 70.65(b)(4) requires in the ISA Summary information to demonstrate compliance with the performance requirements of 10 CFR 70.61.

Text removed under 10 CFR 2.390.

"AC" is used to designate both administrative control and enhanced administrative control. These need to have two different designations.

LES Response

The tables will be corrected in the next revision to the SAR as described below.

Accident consequences for accident sequence "EE-Local Precip" are both: (1) chemical due to the potential release of UF_6, and (2) criticality due to the local flooding. This is reflected in the Accident Description in SAR Table 3.7-4. The SAR Tables 3.7-3, 3.8-1 and 3.8-2 will be revised to properly reflect that this is both a chemical and criticality consequence accident sequence.

Accident consequences for accident sequence "EE-Internal Flooding from On-site Tanks and Water Impoundments" are both: (1) chemical due to the potential release of UF_6, and (2) criticality due to the local flooding. This is properly reflected in the Accident Description in SAR Table 3.7-4 as well as in SAR Tables 3.8-1 and 3.8-2. SAR Table 3.7-3 will be revised to properly reflect that this is both a chemical and criticality consequence accident sequence.

In regard to the need for an additional identifier to designate the enhanced administrative control, in response to RAI ISA-42, LES committed to rewrite criticality "sole IROFS" to clarify the controls being applied to these accident sequences. One result of this rewrite will be the elimination of all criticality sole IROFS. This will also eliminate the use of any of the *currently* defined "enhanced" administrative controls to prevent criticality accident sequences. LES does not anticipate the need for additional designations.

ISA-50 Tables 3.7-1, 3.7-3, 3.7-4, 3.8-1 and 3.8-2

Clarify the criteria for selecting the Initiating Event Frequency (IEF) for NCS accident sequences.

10 CFR 70.65(b)(4) requires in the ISA Summary information to demonstrate compliance with the performance requirements of 10 CFR 70.61.

There needs to be a clear description of the method used for selecting the IEFs for NCS accident sequences. That method needs to be applied consistently. The method of selecting the IEFs appears to be based on the frequency of the consequences of the accident (i.e., criticality occurring) rather than on the frequency of the initiating event of the accident sequence.

LES Response

The NEF methodology utilizes "Risk Index Evaluation" as described in SAR Section 3.1.1.4 and Section 3.7. In this methodology, initiating event "index numbers" are identified – specific initiating event "frequencies" (IEFs) are not identified or discussed. As such, this response will focus on discussion of the assignment of these indices, which is also in agreement with acceptable example methodology found in NUREG-1520, Chapter 3, Appendix A.

The initiating event index assigned in Tables 3.7-1 and 3.7-3 are taken from the "Frequency Index No." column of SAR Table 3.1-9. The Urenco European operating experience was used to select the appropriate index value from Table 3.1-9 from the appropriate "Based on Evidence" column. The summary description basis for each of the selected index values is presented in the Accident Sequence Descriptions of SAR Tables 3.7-2 and 3.7-4. The "initiating event" description necessarily includes recognition of the need for the event sequence to result in consequences (e.g., "criticality"). Only failures that lead to events with consequences of concern are presented in the ISA Summary. Stated another way, only the uncontrolled (i.e., no IROFS are assumed to function) accident event sequence(s) that lead to (for example) criticality, are required to be presented. The frequency index is selected based on Urenco European operating experience and engineering judgement of the stated event sequence (which results in consequence of concern) assuming no IROFS existed.

As such, the initiating event (i.e., uncontrolled accident sequence), and its frequency of occurrence, is conservatively assumed to always result in events that have consequences. This is required by the methodology. One could logically equate this "frequency of the initiating event" to a "frequency of the consequences," but only if it is understood that the "frequency of the consequences" is a theoretical frequency that might occur if IROFS did not function (i.e., the uncontrolled event). The actual "frequency of the consequences," given the existence of IROFS (or in the Urenco operating experience, given the hardware and administrative controls in place) would be orders of magnitude lower (e.g., 1E-6).

Therefore, the methodology utilized for the NEF risk index evaluation appropriately considers the "frequency of the initiating event" in assigning the frequency index.

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ISA-55 Table 3.7-4, pp 21. through 24

Provide a rationale for the assumed source term for scenarios FF21-2, FF23-2, and FF25-2.

The regulation 10 CFR 70.62(c)(1)(iii) requires that the ISA identifies facility hazards that could effect the safety of licensed materials and thus present an increased radiological risk. Furthermore, 10 CFR 70.65(b)(6) requires that the ISA summary contain a list briefly describing each item relied upon for safety that is identified pursuant to 10 CFR 70.61(e) in sufficient detail to understand their functions in relation to the performance requirements of 10 CFR 70.61.

The basis for the amount of material released is not apparent from the discussion and needs to be consistent with both operational and physical fire considerations.

LES Response

The uranium content identified as the material at risk in determining the source term for each of the scenarios listed were based on the judgment and input of Urenco operations representatives who served on the ISA team. These individuals provided estimates based on their knowledge of comparable operations sequences performed in Urenco's Almelo, Netherlands facility, on which the NEF conceptual design is based.

For scenario FF21-2 and 23-2, it is presumed that open 12-liter containers are being bulked into larger 210 liter metal containers and that the aggregate uranium content would not exceed 4 kg in each scenario.

For scenario 25-2, the quantity of material at risk represents the quantity of uranium content that might be present in open traps (12 liter) and an open waste drum (210 liter) while individual carbon traps are being bulked into the drum. It was estimated that the uranium content would not exceed 50 kg.

The material at risk will be confirmed through the formulation of plant specific operating procedures for each transfer/bulking operation. These sequences will also be documented in summary fashion in a revision to the NEF Fire Hazards Analysis during final design.

With respect to the release fractions used in developing the source term, the ISA team presumed the release of 10% of the UO_2F_2/other uranic material when bound to carbon (atmospheric release fraction = 0.2, respirable fraction = 0.5). These values were selected not withstanding the significant thermal mass bulked granules of carbon represent and the heating needed to generate airborne components from this solid material. The ISA team considered that given this, failures with airborne release from sealed metal traps, containers, and drums would not occur.

A point of reference offered for comparative purpose is Response Technical Manual (RTM) 96, which is issued by the NRC. Methods F.2 and F.3 of this document identify that fire release fraction for Uranium and/or non-volatile compounds is 0.001. LES applied significantly more conservative values and considered that the scenarios being postulated for fire induced releases are reasonable.

ISA-56 Section 3.8.1, pp. 3.8-1 and 3.8-2

Describe how the attributes and boundaries of each IROFS will be identified to plant personnel, including operations, maintenance, and engineering, once final design is completed (i.e., define how appropriate information concerning IROFS will "flowdown" to the plant staff).

10 CFR 70.62(c)(1)(vi) requires each applicant to conduct and maintain an ISA of appropriate detail for the complexity of the process, that identifies for each IROFS, the characteristics of its preventive, mitigative, or other safety function, and the assumptions and conditions under which the item is relied on to support compliance with the performance requirements of 10 CFR 70.61.

Section 3.8.1, IROFS, states that management measures will ensure that IROFS are designed, implemented and maintained, as necessary, to be available and reliable to perform their safety functions when needed. Information related to IROFS hardware design details, identification of essential utilities, operating ranges and limits, etc., will be available onsite in the ISA

documentation, once final design is complete. Table 3.8-1, Items Relied on for Safety (IROFS), describes the IROFS safety function and reliability management measures.

LES Response

Documentation of IROFS boundaries requires completion of the design, and finalization of the design-specific IROFS (which will be included in the ISA Summary). Upon completion of the design and identification of IROFS, the IROFS boundaries will be defined using LES procedure DP-ISA-1.1, "IROFS Boundary Definition." This procedure will identify each design attribute (e.g., fail-safe, independence from other IROFS, system interface isolation), support system, component, process steps (e.g., operator actions), and specifics of each management measure necessary to ensure the IROFS is capable of performing its specified safety function, including availability and reliability, consistent with the assumptions presented in the ISA documentation. This procedure may require certain other processes to develop appropriate engineering inputs (e.g., applicable design and testing codes and standards; specific preventive maintenance tasks and frequencies based on industry or manufacturer experience), however, the output will provide the IROFS boundary definition.

Once the IROFS boundary definition is identified as described above, all configuration management attributes, including the specific implementing procedures for all applicable management measures (e.g., maintenance, preventive maintenance, functional test [preoperational and operational], inspections, calibrations, response time tests, personnel training and qualifications, audits and assessments), will be implemented (or verified to be in place). The ongoing process of design finalization, IROFS identification, IROFS boundary identification, and development of applicable implementation processes (i.e., "flow-down" to applicable plant staff and procedures), will be conducted in accordance with the NEF QA Program, which will audit and assess all aspects of implementation.

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ISA-58 Table 3.8-1, pp. 10, 11, and of 14

Provide a definition of an "enhanced " administrative IROFS. Confirm that any enhanced administrative controls will be captured in written procedures. Also, provide definitions of "passive engineered controls," "active engineered controls," and "administrative controls."

The regulation 70.61(e) requires that each engineered or administrative control or control system necessary to comply with the performance requirements of section 10 CFR 70.61 be designated as an item relied on for safety. In addition, 70.65(b)(6) requires that the ISA summary contain a list briefly describing each IROFS that is identified pursuant to 10 CFR 70.61(e) in sufficient detail to understand their functions in relation to the performance requirements of 10 CFR 70.61.

For example, IROFS36 is often referred to as an "enhanced" administrative IROFS by the applicant and assigned a failure probability index of -3. What is the difference between an administrative control and an enhanced administrative IROFS as used by the applicant?

Text removed under 10 CFR 2.390.

LES Response

The presentation of the ISA methods employed to demonstrate the performance requirements of 10 CFR 70.61 currently in the NEF SAR, includes "sole IROFS," "enhanced administrative controls," "Reliability Management Measures," and an inconsistent presentation of applicable management measures. In response to various other RAIs (e.g., ISA-42, -43, -47, -59), portions of each topic are discussed. The following summary provides an overall description of related clarifications proposed to be included in the next revision to the SAR.

"Sole IROFS" for criticality sequences currently reflect multiple controls "rolled-up" into a single IROFS. These controls are typically presented as "enhanced administrative controls" and have specific "Reliability Management Measures" shown in Table 3.8-1. The next SAR revision will present these IROFS in a more fundamental fashion that will identify two IROFS for each criticality sequence. This will also eliminate the use of any of the currently defined "enhanced" administrative controls to prevent criticality accident sequences. Since the current presentation of sole IROFS contained multiple controls, and attempted to address the special preventive measures being applied for criticality control, this "packaging" of controls was referred to as "enhanced" and would typically be reflected with Failure Frequency Index Numbers more negative (i.e., "-3") than non-enhanced administrative controls. Furthermore, since the Reliability Management Measures currently shown in Table 3.8-1 do not convey a consistent approach to management measures, use of this information also did not adequately convey the intent of "enhanced administrative controls."

To provide a consistent presentation of management measures, a clear commitment to apply each management measure defined in 10 CFR 70.4, and required by 10 CFR 70.62(d), to each IROFS will be included in the next revision to the SAR. This commitment and a generic description of each management measure will be added to SAR Section 3.1.8.3. The general description for each management measure reflects the minimum requirements applicable to each IROFS identified in Table 3.8-1. The information provided under the column heading "Reliability Management Measures" in Table 3.8-1 will remove references to standard management measures that are consistent with the description in Section 3.1.8.3. In the next revision to Table 3.8-1 any management measure explicitly identified in the "Reliability Management Measures" column will reflect specific requirements that deviate from the general management measure requirements (i.e., relaxed, enhanced, or additional requirements). These Reliability Management Measures will be consistent with the performance requirements assumed in the ISA documentation used to establish the Failure Frequency Index Number and/or Failure Duration Index Number. The basis for deviation of each management measure listed in Table 3.8-1 will be discussed in Section 3.8.3.

Text removed under 10 CFR 2.390.

Formal ISA Team methodology will be applied to produce the required documentation to support the SAR (ISA Summary) changes summarized above.

SAR Sections 5.4.1, 5.4.2, and 5.4.3 provide generalized discussions and specific examples of criticality prevention using passive engineered controls, active engineered controls, and administrative controls that adequately define these terms. In addition, administrative IROFS will be implemented using written procedures.

<u>ISA-59</u> <u>Table 3.8-1, p. 12 of 14</u>

Define the term "independent verification."

10 CFR 70.62(c)(1)(vi) requires each applicant to conduct and maintain an ISA of appropriate detail for the complexity of the process, that identifies for each IROFS, the characteristics of its preventive, mitigative, or other safety function, and the assumptions and conditions under which the item is relied on to support compliance with the performance requirements of 10 CFR 70.61.

The identified reliability management measure is personnel training and annual refresher training. However, what constitutes an "independent " verification, whether proceduralized and separated by time, method or personnel performing the action is not defined.

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Independent verification has been applied to IROFS that require the potential for personnel errors or procedural errors to be reduced. Conduct of these activities and the associated independent verifications will be in accordance with approved procedures and documented checklists. In the next revision to the SAR, Section 11.4, LES will include the following commitment to address what constitutes "independent" verification of conditions by personnel:

> 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."

When required (as in the case of IROFS40), independent verification of each applicable IROFS safety function 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 has been performed correctly (e.g., verify a condition) or re-perform the task (e.g., collect and analyze a sample). 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 or process (as applicable) involved and its relationship to facility safety.

Each employee is trained to comply with procedures. If compliance cannot be achieved, employees are trained to stop the activities, place the system or component in a safe condition, and notify management. The LES policy is to maintain a safe work place for its employees, to assure operational compliance with the terms and conditions of the license and applicable regulations, and promote a safety conscious work environment at the NEF. This helps ensure that employees assigned the responsibility to perform independent verification are able to perform verification accurately without fear of reprisal.

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ISA-62 Table 3.8-1

Provide the technical basis for demonstrating that administrative control IROFs will meet the performance requirements of 10 CFR 70.61.

The regulations in 10 CFR 70.61 and 70.62 require that an applicant perform an ISA of the hazards associated with the proposed facility and demonstrate compliance with the performance requirements in10 CFR 70.61(b), (c), and (d).

Table 3.8-1 lists approximately fifteen (15) additional administrative control IROFS not designated as Class A. While these are not classified as sole IROFS, the applicant needs to demonstrate that it can meet the performance requirements in 10 CFR 70.61 with administrative controls. The applicant needs to provide an improved technical basis.

LES Response

In the next revision to the SAR, for the IROFS designated as Class "N/A," accident sequences will be developed and the Risk Index Evaluation described in SAR Section 3.1.1.4 will be applied to demonstrate compliance with 10 CFR 70.61. The results will be incorporated (as applicable) into SAR Table 3.7-1 Accident Sequence and Risk Index, Table 3.7-2,Accident Sequence Descriptions, Table 3.8-1, Items Relied on for Safety (IROFS) and Table 3.8-2, Sole Items Relied on for Safety (IROFS).

The acceptability of all IROFS (including all classes of administrative control IROFS) to demonstrate compliance with 10 CFR 70.61 is determined by applying the methodology described in SAR Section 3.1.1.4. For each accident sequence, a risk index number was calculated based on selecting the appropriate parameters from SAR Table 3.1-9 through Table 3.1-11 in accordance with the methodology described in SAR Section 3.1.1.4. The acceptable risk for these sequences was demonstrated by comparing the calculated risk index number to the acceptable risk values provided in SAR Table 3.1-6 for the consequence level for the accident sequence. The risk index results for the accident sequences, as documented in SAR Tables 3.7-1 and 3.7-3, demonstrate compliance with the performance criteria of 10 CFR 70.61.

The appropriate application of management measures to administrative IROFS, as reflected in reliability and availability assumptions documented in the ISA, provides the remaining support for the technical adequacy of administrative controls. As discussed in responses to RAIs ISA-43, ISA-48, ISA-58, and NCS-6, in the next revision to the SAR, LES will clarify the presentation of management measures – specifically clarifying application to administrative IROFS.

To provide a consistent presentation of management measures, a clear commitment to apply each management measure defined in 10 CFR 70.4, and required by 10 CFR 70.62(d), to each IROFS will be included in the next revision to the SAR. This commitment and a generic description of each management measure will be added to SAR Section 3.1.8.3. The general description for each management measure reflects the minimum requirements applicable to each IROFS identified in Table 3.8-1. The information provided under the column heading "Reliability Management Measures" in Table 3.8-1 will remove references to standard management measures that are consistent with the description in Section 3.1.8.3. In the next revision to Table 3.8-1 any management measure explicitly identified in the "Reliability Management Measures" column will reflect specific requirements that deviate from the general management measure requirements (i.e., relaxed, enhanced, or additional requirements). These Reliability Management Measures will be consistent with the performance requirements assumed in the ISA documentation used to establish the Failure Frequency Index Number and/or Failure Duration Index Number. The basis for deviation of each management measure listed in Table 3.8-1 will be discussed in Section 3.8.3.

Formal ISA Team methodology will be applied to produce the required documentation to support the SAR (ISA Summary) changes summarized above.

ISA-63 Table 3.8-1, pp. 10 of 14 through 11 of 14; Table 3.7-4, pp. 12 of 29 through 29 of 29

Provide a comprehensive description of how the applicable attributes of IROFS36 provide a prevention or mitigation role in the sequences in which they are used. In addition, discuss how design margin and surveillances are incorporated into combustible loading controls to achieve the desired reliability.

The regulation 10 CFR 70.65(b)(6) requires that the ISA summary contain a list briefly describing each item relied upon for safety that is identified pursuant to 10 CFR 70.61(e) in sufficient detail to understand their functions in relation to the performance requirements of 10 CFR 70.61.

In Table 3.7-4, IROFS36 is used in a number of different fire accident sequences but it is not clear, in many cases, what attribute is being credited and why it is being credited.

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Chapter 4.0 Radiation Protection

RP-1 Section 4.1, pp. 4.1-1

Please explain the applicant's intent to comply with 20.1101(d).

Applicants must provide a radiation protection program that is adequate to protect the radiological health and safety of workers and members of the public in accordance with 10 CFR Part 20.

In Section 4.1 of the application, the applicant states that the radiation protection program meets the requirements of 10 CFR 20, Subpart B - Radiation Protection Programs. The applicant goes on to further outline its commitment to 10 CFR 20.1101(a-c), however, no specific information is provided with respect to 10 CFR 20.1101(d).

LES Response

LES stated in the initial sentence to Section 4.1 the commitment for the radiation program to meet the requirements of 10 CFR 20, Subpart B (i.e., 10 CFR 20.1101), which adequately infers all portions thereof. Additional explicit reference to 10 CFR 20.1101, paragraph (d), will be reflected in the next revision to the SAR.

RP-2 Section 4.4.1, pp. 4.4-1

Please explain the qualifications for a "radiation specialist."

Applicants must provide a radiation protection program that is adequate to protect the radiological health and safety of workers and members of the public in accordance with 10 CFR Part 20 and 10 CFR 70.22(a)(6).

In section 4.4.1 of the application, the applicant states that, at a minimum, the Radiation Work Permit requires approval by a staff member who is a radiation specialist, however, there is no discussion of the qualifications of a "radiation specialist."

LES Response

Reference to "radiation specialist" will be removed in the next revision to the SAR. In this revision, RWPs will require "approval by the Radiation Protection Manager or designee. The designee must meet the requirements of Section 4.1.2, Staffing of the Radiation Protection Program."

Chapter 5.0 Nuclear Criticality Safety

NCS-1 Section 5, General

For each process or equipment component that has Special Nuclear Material (SNM) associated with it, identify the amount, type, and location of fissionable material that will be present.

10 CFR 70.62(a) requires the establishment and maintenance of a safety program to demonstrate compliance with the performance requirements of 10 CFR 70.61.

10 CFR 70.65(a) requires a description of this safety program to be submitted in the license application.

The locations and descriptions of SNM to be used in the facility are needed to assess hazards.

LES Response

During an April 26, 2004, conference call between LES and NRC representatives, the NRC clarified that the information requested for NCS-1 is as follows.

> For the criticality accident sequences in SAR Chapter 3, "Integrated Safety Analysis Summary," involving areas with non-favorable geometry, provide the amount, type, and enrichment of fissile material in these areas.

This information is provided in the following Table.

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Text removed under 10 CFR 2.390.

<u>NCS-2 Section 5.0, General</u>

Clarify if you intend to commit to using the American Nuclear Society (ANS)-8 series of national standards for NCS that are applicable to the proposed facility or provide specific justifications for using alternative approaches.

10 CFR 70.62(a) requires the establishment and maintenance of a safety program to demonstrate compliance with the performance requirements of 10 CFR 70.61.
10 CFR 70.65(a) requires a description of this safety program to be submitted in the license application.

In Chapter 5.0, there needs to be a clear commitment to the requirements and recommendations of the appropriate ANS-8 series of national standards for NCS or justifications for using alternative approaches (e.g., there is no reference to ANS-8.22, "NCS Based on Limiting and Controlling Moderators" and ANS-8.23, "Nuclear Criticality Accident Emergency Planning and Response). The latest versions of the standards should be committed to (e.g., the application refers to ANS-8.1-1983, "NCS in Operations with Fissionable Materials Outside Reactors" instead of the most recent 1998 version). If no alternative approaches are provided, then the commitment needs to be to all the requirements and recommendations in the standards, rather than to some of the requirements and recommendations in the standards (e.g., Section 2.3.3 does not commit to all the requirements and recommendations of ANS-8.19, "Administrative Practices for NCS" and ANS-8.20, "NCS Training"). For guidance only, see NUREG-1520 Sections 5.4 through 5.4.2.

LES Response

NEF SAR Section 5.0 presents the commitment that the NCS Program is in accordance with Regulatory Guide 3.71, "Nuclear Criticality Safety Standards for Fuels and Materials Facilities," dated August, 1998, which encompasses commitments to multiple ANSI/ANS-8 series standards, including ANSI/ANS-8.22-1997, "NCS Based on Limiting and Controlling Moderators" and ANSI/ANS-8.23-1997, "Nuclear Criticality Accident Emergency Planning and Response." Furthermore, in response to RAI NCS-4, an explicit reference to ANSI/ANS-8.22-1997 will be included in the next revision to the SAR.

In the next revision to SAR Section 2.2.1 (refer to response to RAI "OA-4"), a commitment to ANSI/ANS-8.19-1996, "Administrative Practices for Nuclear Criticality Safety," will address the management responsibilities, supervisory responsibilities, and the nuclear criticality safety engineering staff responsibilities that address training. SAR Section 2.3.3 currently provides a commitment to ANSI/ANS-8.20-1991, "Nuclear Criticality Safety Training," without exception.

Two of the standards included in Regulatory Guide 3.71 have later versions that will be reflected in all associated references the next revision to the SAR. These include:

- ANSI/ANS-8.1, updated to the 1998 revision
- ANSI/ANS-8.7, updated to the 1998 revision

Additionally, the next revision to the SAR will remove reference to ANSI/ANS-8.9-1987, "Nuclear Criticality Safety Criteria for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Material," since this standard has been withdrawn by the ANS-8 working group. Piping configurations containing aqueous solutions of fissile material will be evaluated in accordance with ANSI/ANS-8.1-1998, using validated methods to determine subcritical limits. This exception to Regulatory Guide 3.71 will also be shown in SAR Section 5.0 in the next revision.

Certain other ANSI/ANS standards from Regulatory Guide 3.71 are not currently reflected in the NEF SAR since they do not currently apply to any design attribute. Should need for these designs arise, the commitment to the applicable standard is imposed by the commitment to Regulatory Guide 3.71.

NCS-3 Section 5.1.1, p. 5.1-1; and Section 5.2.1.3.2, p. 5.2-3

Provide in Sections 5.1.1 and 5.2.1.3.2 a discussion of the 1.5 wt.% U-235 control limit on enrichment for the Contingency Dump System.

10 CFR 70.62(a) requires the establishment and maintenance of a safety program to demonstrate compliance with the performance requirements of 10 CFR 70.61.
10 CFR 70.65(a) requires a description of this safety program to be submitted in the license application.

A discussion of the 1.5 weight percent U-235 enrichment control limit for the Contingency Dump System is needed. Sections 5.1.1 and 5.2.1.3.2 describe an enrichment control limit of 5.0 weight percent U-235. However, the Contingency Dump System has an enrichment control limit of 1.5 weight percent U-235.

LES Response

The average enrichment of the cascade hold-up is calculated to be less than 1.5% weight percent ^{235}U. The design of the contingency dump piping connection from the cascade ensures that the contingency dump system will draw off the average hold-up throughout its operation, as positioned at the tails end of the cascade.

NCS-4 Sections 5.2, pgs. 5.2-1 - 5.2-5; 5.4, pgs. 5.4-1 - 5.4-3; and 5.3.16, pgs. 5.3-12 - 5.3-13

Clarify that the safety program in Chapter 5.0 includes all the programmatic commitments and descriptions of how to meet the commitments for the NCS program.

10 CFR 70.62(a) requires the establishment and maintenance of a safety program to demonstrate compliance with the performance requirements of 10 CFR 70.61.
10 CFR 70.65(a) requires a description of this safety program to be submitted in the license application.

For example, moderator control and mass control appear to be used but are not described in Section 5.4. For guidance only, see Chapter 5.0 of NUREG-1520 with emphasis on Sections 5.4.3.4.1, 5.4.3.4.2, and 5.4.3.4.4.

LES Response

In the next revision to the SAR, Chapter 5 will be clarified to appropriately and consistently address nuclear criticality controls utilized for the NEF. Currently (for example), Section 5.1.2 contains discussion of moderator and mass control and Table 5.1-1 similarly presents safe values for these parameters. However, Section 5.4 may introduce lack of overall clarity by not describing these controls. It is anticipated that certain Chapter 5 subsections will be combined, and the discussions consolidated. In general, the following summary of nuclear criticality controls utilized for the NEF will be presented in the appropriate Chapter 5 Section.

The major controlling parameters used in the facility are enrichment control, geometry control, moderation control, and/or limitations on the mass as a function of enrichment. In addition, reflection, interaction, and heterogeneous effects are important parameters considered and applied where appropriate in nuclear criticality safety analyses. Nuclear Criticality Safety Evaluations and Analyses are used to identify the significant parameters affected within a particular system. All assumptions relating to process, equipment, material function, and operation, including credible abnormal conditions, are justified, documented, and independently reviewed.

The specific control parameters considered are:

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 NEF 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

Geometry/volume control may be used to limit the shape of uranium contained within specific process operations or vessels, and within storage containers.

The geometry/volume limits are chosen to ensure $k_{eff} \leq 0.95$.

The safe values of geometry/volume define the characteristic dimension of importance for a single unit of a specified shape such that nuclear criticality safety is not dependent on any other parameter assuming 6w/o ^{235}U for safety margin.

Moderation

Water and oil are the moderators considered in NEF. At NEF the only system where moderation is used as a control parameter is in the product cylinders. Moderation control is established consistent with the guidelines of ANSI/ANS-8.22-1997 and incorporates the criteria below:

- Controls are established to limit the amount of moderation entering the cylinders.
- When moderation is the only parameter used for criticality control, the following additional criteria are applied. These controls assure that at least two independent controls would have to fail before a criticality accident is possible.
  - Two independent controls are utilized to verify cylinder moderator content.
  - These controls are established to monitor and limit uncontrolled moderator prior to returning a cylinder to production thereby limiting the amount of uncontrolled moderator from entering a system to an acceptable limit.
  - The evaluation of the cylinders under moderation control includes the establishment of limits for the ratio of maximum moderator-to-fissile material for

both normal operating and credible abnormal conditions. This analysis has been supported by parametric studies.

- When moderation is not considered a control parameter, either optimum moderation or worst case H/U ratio is assumed when performing criticality safety analysis.

Mass

Mass control may be utilized to limit the quantity of uranium within specific process operations, vessels, or storage containers. Mass control may be used on its own or in combination with other control methods. Analysis or sampling is employed to verify the mass of the material. Conservative administrative limits for each operation are specified in the operating procedures.

Whenever mass control is established for a container, records are maintained for mass transfers into and out of the container. Establishment of mass limits for a container involves consideration of potential moderation, reflection, geometry, spacing, and enrichment. The evaluation considers normal operations and credible abnormal conditions for determination of the operating mass limit for the container and for the definition of subsequent controls necessary to prevent reaching the safety limits. When only administrative controls are used for mass controlled systems, double batching is conservatively assumed in the analysis.

Reflection

Reflection is considered when performing Nuclear Criticality Safety Evaluations and Analyses. The possibility of full water reflection is considered but the layout of the NEF is a very open design and it is highly unlikely that those vessels and plant components requiring criticality control could become flooded from a source of water within the plant. In addition, neither automatic sprinkler nor standpipe and hose systems are provided in the TSB, Separation Buildings, Blending and Liquid Sampling, CRDB, CAB, and Centrifuge Post Mortem areas. Therefore, full water reflection of vessels has therefore been discounted. However, some select analyses have been performed using full reflection for conservatism. Partial reflection of 2.5 cm of water is assumed where limited moderating materials (including humans) may be present It is recognized that concrete can be a more efficient reflector than water; therefore, it is modeled in analyses where it is present. When moderation control is identified in the ISA Summary, it is established consistent with the guidelines of ANSI/ANS-8.22-1997.

Interaction

Nuclear criticality safety evaluations and analyses 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. Units may be considered non-interacting when they are separated by more than 60 cm (23.6 inches).

If a unit is considered interacting, nuclear criticality safety analyses are performed. Individual unit multiplication and array interaction are evaluated using the Monte Carlo computer code MONK8A to ensure $k_{eff} \leq 0.95$.

Concentration, Density and Neutron Absorbers

NEF does not use mass concentration, density, or neutron absorbers as a criticality control parameter.

NCS-5 Section 5.2.1, p. 5.2-1

Clarify that the MONK8A code used for NCS calculations will be controlled under the Quality Assurance Program Description.

10 CFR 70.62(a) requires the establishment and maintenance of a safety program to demonstrate compliance with the performance requirements of 10 CFR 70.61.
10 CFR 70.65(a) requires a description of this safety program to be submitted in the license application. 10 CFR 70.65(d) requires the establishment of management measures to ensure compliance with performance requirements of 10 CFR 70.61.

NCS computer codes need to be controlled under the Quality Assurance Program to ensure that results are reliable and computer codes properly documented.

LES Response

The MONK8A computer code and JEF 2.2 library are considered within the scope of the QA Program. Specifically, SAR Chapter 11, Appendix A, Section 3, Design Control, addresses "Computer Software Controls" and imposes "Part II ASME NQA-1-1994 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 I, Supplement 11S-2, *Supplementary Requirements for Computer Program Testing.*"

NCS-6 Section 5.4, General

Provide a discussion of criticality prevention using "enhanced administrative controls."

10 CFR 70.62(a) requires the establishment and maintenance of a safety program to demonstrate compliance with the performance requirements of 10 CFR 70.61.
10 CFR 70.65(a) requires a description of this safety program to be submitted in the license application.

Sections 5.4.1, 5.4.2, and 5.4.3 provide discussions of criticality prevention using passive engineered controls, active engineered controls, and administrative controls. A discussion of criticality prevention using enhanced administrative controls is needed.

LES Response

The presentation of the ISA methods employed to demonstrate the performance requirements of 10 CFR 70.61 currently in the NEF SAR, includes "sole IROFS," "enhanced administrative controls," "Reliability Management Measures," and an inconsistent presentation of applicable management measures. In response to various other RAIs (e.g., ISA-42, -43, -47, -59), portions of each topic are discussed. The following summary provides an overall description of related clarifications proposed to be included in the next revision to the SAR.

"Sole IROFS" for criticality sequences currently reflect multiple controls "rolled-up" into a single IROFS. These controls are typically presented as "enhanced administrative controls" and have specific "Reliability Management Measures" shown in Table 3.8-1. The next SAR revision will present these IROFS in a more fundamental fashion that will identify two IROFS for each criticality sequence. This will also eliminate the use of any of the currently defined "enhanced" administrative controls to prevent criticality accident sequences. Since the current presentation of sole IROFS contained multiple controls, and attempted to address the special preventive measures being applied for criticality control, this "packaging" of controls was referred to as "enhanced" and would typically be reflected with Failure Frequency Index Numbers more negative (i.e., "-3") than non-enhanced administrative controls. Furthermore, since the Reliability Management Measures currently shown in Table 3.8-1 do not convey a consistent approach to management measures, use of this information also did not adequately convey the intent of "enhanced administrative controls."

To provide a consistent presentation of management measures, a clear commitment to apply each management measure defined in 10 CFR 70.4, and required by 10 CFR 70.62(d), to each IROFS will be included in the next revision to the SAR. This commitment and a generic description of each management measure will be added to SAR Section 3.1.8.3. The general description for each management measure reflects the minimum requirements applicable to each IROFS identified in Table 3.8-1. The information provided under the column heading "Reliability Management Measures" in Table 3.8-1 will remove references to standard management measures that are consistent with the description in Section 3.1.8.3. In the next revision to Table 3.8-1 any management measure explicitly identified in the "Reliability Management Measures" column will reflect specific requirements that deviate from the general management measure requirements (i.e., relaxed, enhanced, or additional requirements). These Reliability Management Measures will be consistent with the performance requirements assumed in the ISA documentation used to establish the Failure Frequency Index Number and/or Failure Duration Index Number. The basis for deviation of each management measure listed in Table 3.8-1 will be discussed in Section 3.8.3.

Formal ISA Team methodology will be applied to produce the required documentation to support the SAR (ISA Summary) changes summarized above.

Chapter 6.0 Chemical Process Safety

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

CS-3 Table 6.3-5

Provide a rational basis for adjusting acute chemical release limits through the use of a time-weighted average (TWA) method, and confirm that the proposed Acute Exposure Guideline Level (AEGL) values are based on the latest published figures.

10 CFR 70.65 (b)(7) requires that the ISA Summary contain a description of the proposed quantitative standards used to assess the consequences to an individual from acute chemical exposure to licensed material or chemicals produced from licensed materials.

Table 6.3-5, Enhanced Definition of Consequence Severity Categories, represents enhanced derived values as extrapolated from the HF and UF_6 (as soluble uranium) AEGLs. The table utilizes a 1-minute and 5-minute acute chemical exposure time for a local worker and a worker in the room, respectively, and a 30-minute exposure time for outside the controlled area. NUREG-1520, Section 6.4.3.1, Process Chemical Risk and Accident Sequences, notes that acute chemical release limits may not be adjusted by a TWA calculation unless a rational basis is provided in the ISA Summary. Use of an approach endorsed by an internationally recognized

committee, such as contained in the National Academy of Sciences latest revision to the AEGLs (2004) would be acceptable.

LES Response

LES adjusted the acute chemical release limits for the "Worker Elsewhere in the Room" in Table 6.3-5. For the "Local Worker," LES used values derived in NUREG-1391, "Chemical Toxicity of Uranium Hexafluoride Compare to Acute Effects of Radiation," February 1991. For "Outside the Controlled Area," LES adopted the AEGL-1 value.

The "Worker Elsewhere in the Room" exposure values in Table 6.3-5 were calculated by applying "time scaling" to the proposed AEGLs for uranium hexafluoride and hydrogen fluoride. Time scaling, unlike a time-weighted average method, uses the relationship between exposure concentration (*C*) and exposure duration (*t*) expressed by the equation $C^n \times t = k$, where *k* is a constant and *n* defines the relationship for a given chemical and a specific health effect endpoint.

Time scaling is used by the National Advisory Committee for Acute Exposure Guideline Levels for Hazardous Substances (NAC/AEGL Committee) in its efforts to develop scientifically credible short-term exposure limits for approximately 400 to 500 acutely toxic substances. The method is employed by the NAC/AEGL Committee because experimental animal and controlled human exposure-response data and data from human exposure incidents often involve exposure durations differing from those specified for AEGLs. Therefore, AEGL development usually requires extrapolation from reported exposure duration and chemical concentration of a toxic endpoint to an equivalent concentration for an AEGL-specified exposure period.

In addition, this time scaling was applied to the proposed AEGL-3 and AEGL-2 values for uranium hexafluoride and hydrogen fluoride to develop the "Worker Elsewhere in the Room," high and intermediate consequences categories, respectively. However, since submittal of the License Application in December 2003, the proposed AEGL values used to determine the values in Table 6.3-5 have been made final and published. The final AEGL values for some exposure periods have changed for both uranium hexafluoride and hydrogen fluoride.

As a result, LES has updated its calculations based on the final AEGL values, which will affect values in Table 6.3-5. In the next revision to the SAR, Table 6.3-5 will be revised to reflect these changes.

Chapter 7.0 Fire Safety

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Text removed under 10 CFR 2.390.

Chapter 10.0 Decommissioning

D-1 Section 10.1.3, pp. 10.1-1 and 10.1-2

Provide the increased level of detail described in NUREG-1757, Volume 3, for the decommissioning cost estimate.

The regulations in 10 CFR 70.25 require applicants for a uranium enrichment facility to have decommissioning funding plans. NUREG-1757, Volume 3, "Consolidated NMSS Decommissioning Guidance," specifically Chapter 4, Section 4.1, "Cost Estimate (As Contained in a Decommissioning Funding Plan or Decommissioning Plan)" provides the information and acceptance criteria that should be used in both developing and evaluating the decommissioning cost estimate. Appendix A, Section A.3.1, "Preparing the Site Specific Cost Estimate," provides specific guidance on the information to be included in the cost estimate for the staff to be able to make a finding regarding the adequacy of the cost estimate, and reasonable assurance regarding funding to support decommissioning.

The applicant's cost estimate for the facility is a summary of the decommissioning costs and is presented in Table 10.1-1, "Total Decommissioning Costs," and does not include the supporting basis for how the applicant arrived at the summary estimates for each activity. For the staff to evaluate the decommissioning cost estimate, the applicant needs to provide the level of detail described to NUREG-1757, Volume 3, Section A.3.1.

LES Response

The decommissioning cost estimate tables in SAR Section 10.1 will be revised to be consistent with Tables D-1.1A through D-1.14 presented in Attachment 2. The re-presentation is intended to provide the level of detail described in NUREG–1757, Volume 3, to the extent practicable. These revised tables do not contain classified information. The decommissioning cost estimate tables containing classified information are being provided by Urenco and will follow shortly. The estimated costs in the attached Tables are based on recent commercial decommissioning cost estimates and anticipated activity durations, and are intended to be in sufficient detail such

that the NRC staff would be able to make their finding regarding the adequacy of the cost estimate and reasonable assurance regarding funding to support decommissioning. The next revision to the SAR will also incorporate these attached Tables.

D-2 Section 10.1.3, pp. 10.1-1 and 10.1.2

Provide a thorough justification for using a contingency factor less than 25 percent.

The regulations in 10 CFR 70.25 require applicants for a uranium enrichment facility to have decommissioning funding plans. NUREG-1757, Volume 3, "Consolidated NMSS Decommissioning Guidance," specifically Chapter 4, Section 4.1, "Cost Estimate (As Contained in a Decommissioning Funding Plan or Decommissioning Plan)" has recommended a contingency factor of 25 percent.

The applicant is using a contingency factor of 10 percent and has based the reduced factor on past experience. While the staff agrees that a contingency factor lower than 25 percent may be warranted based on past experience at similar facilities, the applicant needs to provide a stronger supporting basis for a reduced contingency, and although a reduced contingency may be warranted, the staff believes 10 percent may not be sufficient. In addition, the staff believes that the contingency factor needs to be applied across the board, and includes applying the contingency factor to the cost of the tails disposition (also see Comment D-4).

LES Response

In the next revision to the SAR, the 10% contingency factor will be revised to 25% (which is also shown in Attachment 2 Table D-1.14) as recommended in NUREG-1757 for all decommissioning costs except those associated with tails disposition. The contingency factor associated with tails disposition is addressed in the response to RAI D-4.

D-3 Section 10.2.1, p. 10.2-1

Provide an unexecuted copy of the surety mechanism for decommissioning financial assurance.

The regulations in 10 CFR 70.25 require applicants for a uranium enrichment facility to have decommissioning funding plans. Decommissioning funding plans include a certification that financial assurance for decommissioning has been provided in the amount of a site-specific cost estimate and a signed original of the financial assurance instrument used. Under "Consolidated NMSS Decommissioning Guidance," NUREG-1757, it is acceptable to provide the executed surety instruments prior to the commencement of licensed activities or receipt of licensed material.

An unexecuted copy of the financial assurance instrument proposed to be used by the applicant needs to be reviewed to ensure that it meets the requirements in 10 CFR 70.25.

LES Response

LES is supplying an unexecuted copy of a surety bond as the mechanism for decommissioning fund financial assistance in the following pages. Information not available at this time has been left blank and will be provided when available.

PAYMENT SURETY BOND

Date bond executed:

Effective date:

Principal: Louisiana Energy Services, L.P.
100 Sun Avenue NE, Suite 204
Albuquerque, NM 87109

Type of organization: Limited Partnership

State of incorporation: Delaware

NRC license number, name and address of facility, and amount for decommissioning activities guaranteed by this bond:

Surety: [*Insert name and business address*]

Type of organization: [*Insert "proprietorship," "partnership," or "corporation"*]

State of incorporation: *(if applicable)*

Surety's qualification in jurisdiction where licensed facility is located.

Surety's bond number:

Total penal sum of bond: $

Know all persons by these presents, that we, the Principal and Surety hereto, are firmly bound to the U.S. Nuclear Regulatory Commission (hereinafter called NRC) in the above penal sum for the payment of which we bind ourselves, our heirs, executors, administrators, successors, and assigns jointly and severally; provided that, where the Sureties are corporations acting as co-sureties, we, the Sureties, bind ourselves in such sum "jointly and severally" only for the purpose of allowing a joint action or actions against any or all of us, and for all other purposes each Surety binds itself, jointly and severally with the Principal, for the payment of such sum only as is set forth opposite the name of such Surety; but if no limit of liability is indicated, the limit of liability shall be the full amount of the penal sum.

WHEREAS, the NRC, an agency of the U.S. Government, pursuant to the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974, has promulgated regulations in title 10, Chapter I of the *Code of Federal Regulations*, Parts 30, 40, and 70, applicable to the Principal, which require that a license holder or an applicant for a facility license provide financial assurance that funds will be available when needed for facility decommissioning;

NOW, THEREFORE, the conditions of the obligation are such that if the Principal shall faithfully, before the beginning of decommissioning of each facility identified above, fund the standby trust fund in the amount(s) identified above for the facility;

Or, if the Principal shall fund the standby trust fund in such amount(s) after an order to begin facility decommissioning is issued by NRC or a U.S. District Court or other court of competent jurisdiction;

Or, if the Principal shall provide alternative financial assurance, and obtain NRC's written approval of such assurance, within 30 days after the date a notice of cancellation from the Surety is received by both the Principal and NRC, then this obligation shall be null and void; otherwise it is to remain in full force and effect.

The Surety shall become liable on this bond obligation only when the Principal has failed to fulfill the conditions described above. Upon notification by NRC that the Principal has failed to perform as guaranteed by this bond, the Surety shall place funds in the amount guaranteed for the facility into the standby trust fund.

The liability of the Surety shall not be discharged by any payment or succession of payments hereunder, unless and until such payment or payments shall amount in the aggregate to the penal sum of the bond, but in no event shall the obligation of the Surety hereunder exceed the amount of said penal sum.

The Surety may cancel the bond by sending notice of cancellation by certified mail to the Principal and to NRC provided, however, that cancellation shall not occur during the 90 days beginning on the date of receipt of the notice of cancellation by both the Principal and NRC, as evidenced by the return receipts.

The Principal may terminate this bond by sending written notice to NRC and to the Surety 90 days prior to the proposed date of termination, provided, however, that no such notice shall become effective until the Surety receives written authorization for termination of the bond from NRC.

The Principal and Surety hereby agree to adjust the penal sum of the bond yearly so that it guarantees a new amount, provided that the penal sum does not increase by more than 20 percent in any one year and no decrease in the penal sum takes place without the written permission of NRC.

If any part of this agreement is invalid, it shall not affect the remaining provisions that will remain valid and enforceable.

In Witness Whereof, the Principal and Surety have executed this financial guarantee bond and have affixed their seals on the date set forth above.

The persons whose signatures appear below hereby certify that they are authorized to execute this surety bond on behalf of the Principal and Surety.

Principal

[*Signatures*]
E. James Ferland
President, Louisiana Energy Services, L.P.
[*Corporate seal*]

Corporate Surety

[*Name and address*]

State of incorporation:

Liability limit: $

[*Signatures*]
[*Names and titles*]
[*Corporate seal*]

Bond Premium: $

D-4 Section 10.3, pp. 10.3-1 through 10.3-3

Provide a contingency factor for the processing and disposal of depleted uranium. Also, provide copies of the four reports used to prepare the tails disposition cost estimates.

The regulations in 10 CFR 70.25 require applicants for a uranium enrichment facility to have decommissioning funding plans. Decommissioning funding plans include a certification that financial assurance for decommissioning has been provided in the amount of a site-specific cost estimate and a signed original of the financial assurance instrument used.

NUREG-1757, Volume 3, "Consolidated NMSS Decommissioning Guidance," specifically Chapter 4, Section 4.1, "Cost Estimate (As Contained in a Decommissioning Funding Plan or Decommissioning Plan)" has recommended a contingency factor of 25 percent. Chapter 10.3 estimates 132,942 MT of depleted uranium will be generated over the thirty-year life of the facility. The cost of waste processing and disposal cost for the depleted uranium is estimated to be $5.50 per MTU resulting in a total cost of $731,181,000. The cost was based on a comparison of four studies which were developed between 1993 and 2002 and the earlier studies were escalated to 2002 dollars. The cost for disposal of the depleted uranium varied significantly. Because the disposition of the depleted uranium may not take place for more than 30 years, and due to the uncertainty in the studies, LES needs to include a contingency factor in the estimated costs of disposition of the depleted uranium.

In Section 10.3, the applicant has prepared a cost estimate using four references. The staff needs these cost estimate references to evaluate the cost estimate basis.

LES Response

Referencing NEF SAR Section 10.3 and Table 10.3-1, the $5.50 per kgU used to calculate the tails disposal cost as a component of total decommission costs, is based on LES consideration of three sets of relevant cost information sources:

(1) A 1997 study by Lawrence Livermore National Laboratory (LLNL) ("Cost Analysis Report for the Long-Term Management of Depleted Uranium Hexafluoride," (UCRL-AR-127650)) from which LES derived a cost of $5.06 per kgU;

(2) The 2002 Uranium Disposition Services, LLC (UDS) contract with the U.S. Department of Energy (DOE) ("DUF_6 Contract" (DE-AC05-02OR22717)) from which LES derived a cost of $3.92 per kgU;

(3) Depleted uranium tails disposition cost estimates submitted to the NRC in connection with the Claiborne Enrichment Center license application in June 1993 ("Claiborne Enrichment Center (CEC) Disposition of Depleted Uranium Hexafluoride," letter dated June 30, 1993) from which LES derived a cost of $6.74 per kgU.

The simple average of these estimates yields a value of $5.24 per kgU {(5.06+3.92+6.74)/3}. It is noteworthy that available information has supported the calculation of lower estimates of the cost of tails disposal over time. That is, the more recent the date of the cost estimate, the lower the estimated disposal cost. Nonetheless, LES conservatively selected $5.50 per kgU as its estimated unit cost for depleted tails disposition. Information from Urenco, who has operational experience with respect to the disposition of depleted uranium tails, is supportive of this cost estimate of $5.50 per kgU.

It should also be noted that the highest cost estimate ($6.74 per kgU) is at least 10 years old and was based on the information available at that time. The value of $5.50 per kgU used in the decommissioning cost estimate is 22% above the average of the more recent LLNL and UDS cost estimates, which is $4.49 per kgU {(5.06+3.92)/2}. The LLNL Cost Analysis Report (page 30) states that its cost estimate already includes a 30% contingency in the capital costs of the process and manufacturing facilities, a 20% contingency in the capital costs of the balance of plant; and a minimum of a 30% contingency in the capital costs of process and manufacturing equipment.

Also, the 1997 LLNL cost information is five years older than the more recent 2002 UDS cost information. The value of $5.50 per kgU used in the decommissioning cost estimate for tails disposition is 40% greater than the 2002 UDS-based cost estimate of $3.92 per kgU, which does not include offset credits for HF sales or proceeds from the sale of recycled products.

In summary, there is already substantial margin between the value of $5.50 per kgU being used by LES in the decommissioning cost estimate and the most recent information (2002 UDS) from which LES derived a cost estimate of $3.92 per kgU. Accordingly, LES does not believe that a further contingency is warranted.

The three referenced documents (non-proprietary, and cited above) supporting the tails disposition cost estimate have been submitted by letter NEF#04-013 dated May 12, 2004, from R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC). Urenco confirmatory information (proprietary material) supporting the tails disposition cost estimate will be submitted once clearance is obtained from Urenco and its uranium disposition contractor.

Chapter 11.0 Management Measures

MM-1 Section 11.1, p. 11.1-1 through 11.1-12

Describe the Configuration Management (CM) process and controls that are in place during design, license application review, construction, and operation to assure that the design, engineering, procurement, and construction drawings and documents and the ISA are consistent and current.

10 CFR 70.72(a) requires that 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.

Section 11.1 states that the ISA will be under the CM program, but additional clarification of the design/procurement/construction and ISA interfaces and how the design basis is controlled and assured is needed to clarify the management measure adequacy in this area.

LES Response

The configuration management controls are discussed in three phases, i.e., design, construction, and operation, as outlined in SAR Sections 11.1.4.1, 11.1.4.2, and 11.1.4.3, respectively. Full implementation of the regulatory required 10 CFR 70.72, configuration management system, will not occur until the time of license issuance (i.e., during the latter part of the design phase). SAR Section 11.1.4.1 and Section 11.1.4.2 will be revised to more clearly reflect the requirement to have configuration management system procedures and processes that fully comply with 10 CFR 70.72 upon issuance of the NEF Materials License. During the

design phase, the applicable change control process is as specified in the NRC approved LES QA Program Description.

At this time, the ISA documentation, and all associated interface documents, as well as all other design basis documentation and the various license application documents, are being maintained by the responsible organizations, with cross-organizational reviews to any proposed changes, as necessary. Additionally, ISA Team members independently screen all changes for potential impact to the ISA Summary, and will ensure ISA Team review when appropriate.

LES retains responsibility to assure all applicable revisions to the license application are processed and submitted to the NRC in a timely fashion. In accordance with the NRC approved LES QA Program, LES has accepted each organization's internal QA program to adequately assure that the design basis and technical baseline for the facility is accurate and that appropriate input is provided to LES such that the license application will be maintained up-to-date.

An LES proceduralized configuration management change control process is being developed. This procedure, and appropriate complementary change to the other organization's procedures and processes, will be implemented prior to formally entering the "design phase" (which occurs after LES formal acceptance of the Architect Engineer). This "design phase" configuration management system, which will govern the change control process prior to issuance of the NEF Materials License, will ensure the necessary reviews, approvals, and revisions to ISA documentation, and all associated interface documents, as well as all other design basis documentation and the various license application documents. This configuration management system will provide:

(1) Living list of design and licensing basis documents;

(2) Documented consideration of the impacts to all design and licensing basis documents with each change being considered, which becomes part of the change package;

(3) Preparation of concurrent changes (or actions to track required changes) to all impacted design and licensing basis documents for each change being considered, which becomes part of the change package;

(4) Evaluation of potential impact to ISA documentation, including IROFS, or items that could potentially affect the function of IROFS. Changes potentially affecting the ISA will also require ISA Team review. This evaluation becomes part of the change package;

(5) Documented interdisciplinary reviews for each change package prepared, which becomes part of the change package;

(6) ISA Team review, if required, which becomes part of the change package;

(7) Appropriate management approvals will be required before the change is considered approved. LES will notify the NRC of changes that reduce the level of commitments or margin of safety in the design bases of IROFS with appropriate and timely revisions to the license application, including changes to the ISA Summary.

(8) Changes that do not affect the license application, but result in changes that reduce the level of commitments or margin of safety in the design bases of IROFS will be reported to the NRC prior to implementation (i.e., prior to considering the change approved).

Upon issuance of the NEF Materials License, LES will implement a change process that fully implements the provisions of 10 CFR 70.72, including reporting of changes made without prior NRC approval as required by 10 CFR 70.72(d) (2) and (3). Any change that requires Commission approval, will be submitted as a license amendment request as required by 10 CFR 70.72(d)(1).

MM-2 Section 11.1.1, p. 11.1-1

Clarify what selective documentation is controlled by the CM program. Please amplify the scope of the selective documentation or identify documentation types that will be under the CM program and provide specific examples of documents that would not be under the CM program.

10 CFR 70.72(a) requires that 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.

Section 11.1.1.1 states that selective documentation is controlled under the CM program, but does not identify the documents, other than the ISA.

LES Response

Refer to response to RAI MM-1 for additional clarification and discussion. Upon issuance of the NEF Materials License, the Configuration Management program will be in full compliance with the 10 CFR 70.72(a) requirement that each change to the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel will be evaluated. While LES acknowledges that the vast majority of NEF-associated documentation will be continually controlled under this program, documents that do not reflect or involve a "change" to items listed in 10 CFR 70.72(a) would not require CM evaluation. Correspondence, departmental status reports, and personnel evaluations are obvious categories of documentation that would not require CM processes for revision. Other documents, after appropriate consideration, could be screened under the 10 CFR 70.72 process as not having the ability to create a "change to the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel," and could also be excluded.

MM-3 Section 11.1.1.1, p. 11.1-3

Confirm that the scope of structures, systems, and components (SSC) under CM includes all SSCs and each change to them, and not just IROFS, and any items which may affect the function of the IROFS.

10 CFR 70.72(a) requires that 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.

Section 11.1.1.1 states that the scope of the SSCs under CM includes IROFS and any items which may affect the function of the IROFS.

LES Response

LES confirms that compliance with 10 CFR 70.72(a) requires that the configuration management system to evaluate, implement, and track "each [*i.e., all*] change to the site,

structures, processes, systems, equipment, components, computer programs, and activities of personnel." Various statements to this effect are presented in other SAR Sections. For example:

- SAR Section 11.1.1 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, 2003e), as applicable."
- SAR Section 11.1.4.3 states: "LES will implement a change process that fully implements the provisions of 10 CFR 70.72 (CFR, 2003e)."

Refer to response to RAI MM-1 for additional clarification and commitment to clarify other sections of the SAR.

MM-4 Section 11.1.5, p. 11.1-12

Confirm a commitment to ensuring comprehensive program oversight through audits and assessments of the CM program, initially and at least once every year in accordance with the Quality Assurance Program Description (QAPD) and Quality Assurance (QA) procedure requirements.

10 CFR 70.72(a) requires that 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.

The LES QAPD, Revision 0, Section 18, requires audits at least once per year, but does not address if the schedule will be adjusted annually upon evaluation based on an assessment of the applicable QA program elements. Section 11.1.5 states that periodic audits and assessments will be performed of the CM program, but does not identify a frequency.

LES Response

In the next revision to the SAR, LES will include in Section 11.1.5 a reference to QAPD, Section 18, audit frequency, which requires minimum annual internal audits of LES QA Level 1 activities. Furthermore, the referenced QAPD section requires the audit schedule to be developed annually and revised as necessary, basing the frequency of audits on all applicable and active elements of the LES QAPD.

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ATTACHMENT 2

Louisiana Energy Services
Response to April 19, 2004
Request for Additional Information

Tables Referenced from Responses

Table ISA-10.1 All Earthquakes within 322 km (200 mi) of NEF Site

NEF Site Coordinates			Longitude	Latitude							
			-103.0820	32.4360							
Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth (km)	Focal Depth (mi)	MAG	MAG Type	Epicentral Distance (km)	Epicentral Distance (mi)	Data Sources[1]
1931	8	16	-104.60	30.70			3.60	M	240.3	149.3	UTIG
1931	8	16	-104.60	30.70			3.00	M	240.3	149.3	UTIG
1931	8	16	-104.60	30.70			6.00	M	240.3	149.3	UTIG
1931	8	16	-104.60	30.70			3.00	M	240.3	149.3	UTIG
1931	8	16	-104.60	30.70			3.30	M	240.3	149.3	UTIG
1931	8	16	-104.60	30.70			3.60	M	240.3	149.3	UTIG
1931	8	18	-104.60	30.70			3.00	M	240.3	149.3	UTIG
1931	8	18	-104.60	30.70			4.20	M	240.3	149.3	UTIG
1931	8	18	-104.60	30.70			3.00	M	240.3	149.3	UTIG
1931	8	26	-104.60	30.70			3.60	M	240.3	149.3	UTIG
1931	11	3	-104.60	30.70			3.00	M	240.3	149.3	UTIG
1949	5	23	-105.20	34.60			4.50	M	310.0	192.6	NMTH
1955	1	27	-104.50	30.60			3.30	M	244.0	151.6	UTIG
1962	1	3	-103.75	34.85			2.90	M	274.8	170.8	NMTR
1962	3	6	-104.80	31.20			3.50	M	212.3	131.9	UTIG
1963	12	19	-104.27	34.82			3.40	M	287.0	178.3	NMTR
1964	2	11	-103.94	34.23			2.10	M	214.2	133.1	NMTR
1964	3	3	-103.60	34.84			2.90	M	271.0	168.4	NMTR
1964	6	19	-105.77	32.95			1.90	M	257.4	159.9	NMTR
1964	8	14	-102.94	31.97			1.90	M	53.1	33.0	NMTR
1964	9	7	-102.92	31.94			1.60	M	56.9	35.3	NMTR
1964	11	8	-103.10	31.90			3.00	M	59.5	37.0	UTIG
1964	11	21	-103.10	31.90			3.10	M	59.5	37.0	UTIG
1964	11	27	-102.97	31.89			1.90	M	61.1	38.0	NMTR
1965	1	21	-102.85	32.02			1.30	M	50.9	31.6	NMTR
1965	2	3	-103.10	31.90			3.30	M	59.5	37.0	UTIG
1965	8	30	-103.00	31.90			3.50	M	60.0	37.3	UTIG
1966	8	14	-103.00	31.90			3.40	M	60.0	37.3	UTIG
1966	9	17	-103.98	34.89			2.70	M	284.6	176.9	NMTR
1966	10	6	-104.12	35.13			2.90	M	314.4	195.4	NMTR
1966	11	26	-105.44	30.95			3.50	M	277.5	172.4	NMTR
1968	3	23	-105.91	32.67			2.60	M	265.7	165.1	NMTR
1968	5	2	-105.24	33.10			2.60	M	214.3	133.1	NMTR
1969	6	1	-105.21	34.20			1.90	M	277.7	172.5	NMTR
1969	6	8	-105.19	34.15			2.60	M	272.8	169.5	NMTR
1971	7	30	-103.00	31.72	10.0	6.2	3.00	mb	79.9	49.6	ANSS
1971	7	31	-103.06	31.70	10.0	6.2	3.40	mb	81.4	50.6	ANSS
1971	9	24	-103.20	31.60			3.20	M	93.5	58.1	UTIG
1972	7	26	-104.01	32.57			3.10	M	88.3	54.9	NMTR
1973	3	17	-102.36	31.59			2.50	M	115.7	71.9	NMTR
1973	8	2	-105.56	31.04			3.60	M	280.7	174.5	NMTR
1973	8	4	-103.22	35.11			3.00	M	296.6	184.3	NMTR
1974	7	31	-104.19	33.11			0.00	M	128.0	79.5	NMTR
1974	10	2	-100.86	31.87			0.00	M	217.7	135.3	NMTR
1974	10	27	-104.83	30.63			0.00	M	259.6	161.3	NMTR
1974	11	12	-102.67	32.14			0.00	M	51.0	31.7	NMTR
1974	11	21	-102.75	32.07			0.00	M	51.0	31.7	NMTR

Table ISA-10.1 All Earthquakes within 322 km (200 mi) of NEF Site

NEF Site Coordinates			Longitude	Latitude							
			-103.0820	32.4360							
Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth (km)	Focal Depth (mi)	MAG	MAG Type	Epicentral Distance (km)	Epicentral Distance (mi)	Data Sources[1]
1974	11	22	-101.26	32.94			0.00	M	179.2	111.3	NMTR
1974	11	22	-105.21	33.78			0.00	M	247.7	153.9	NMTR
1974	11	28	-103.94	32.58			0.00	M	82.2	51.1	NMTR
1974	11	28	-104.14	32.31	5.0	3.1	3.90	mb	100.4	62.4	ANSS
1974	12	30	-103.10	30.90			3.70	M	170.5	106.0	UTIG
1975	1	30	-103.08	30.95			2.10	M	165.1	102.6	NMTR
1975	2	2	-103.19	35.05			3.00	M	290.7	180.6	NMTR
1975	4	8	-101.69	32.18			0.00	M	133.9	83.2	NMTR
1975	7	25	-102.62	29.82			0.00	M	293.4	182.3	NMTR
1975	8	1	-104.60	30.49			0.00	M	259.5	161.3	NMTR
1975	8	1	-104.00	31.40			3.00	M	143.9	89.4	UTIG
1975	8	3	-104.45	30.71			0.00	M	231.0	143.5	NMTR
1975	10	10	-105.02	33.36			0.00	M	207.4	128.9	NMTR
1975	12	12	-102.31	31.61			3.00	M	117.5	73.0	NMTR
1976	1	10	-102.76	31.79			0.00	M	78.4	48.7	NMTR
1976	1	15	-102.32	30.98			0.00	M	176.6	109.7	NMTR
1976	1	19	-103.09	31.90			3.50	M	59.5	37.0	UTIG
1976	1	21	-102.29	30.95			0.00	M	180.8	112.4	NMTR
1976	1	22	-103.07	31.90	1.0	0.6	2.80	un	59.5	37.0	ANSS
1976	1	25	-103.08	31.90	2.0	1.2	3.90	un	59.3	36.8	ANSS
1976	1	28	-100.89	31.99			0.00	M	211.8	131.6	NMTR
1976	2	4	-103.53	31.68			0.00	M	94.1	58.4	NMTR
1976	2	14	-102.47	31.63			0.00	M	106.2	66.0	NMTR
1976	3	5	-102.25	31.66			0.00	M	116.7	72.5	NMTR
1976	3	15	-102.58	32.50			0.00	M	47.3	29.4	NMTR
1976	3	18	-102.96	32.33			0.00	M	16.5	10.3	NMTR
1976	3	20	-104.94	31.27			0.00	M	217.4	135.1	NMTR
1976	3	20	-103.06	32.22			0.00	M	24.4	15.2	NMTR
1976	3	27	-103.07	32.22			0.00	M	23.7	14.7	NMTR
1976	4	3	-103.10	31.24			0.00	M	132.5	82.3	NMTR
1976	4	12	-103.00	32.27			0.00	M	20.2	12.5	NMTR
1976	4	21	-102.89	32.25			0.00	M	27.7	17.2	NMTR
1976	4	30	-103.09	31.98			0.00	M	50.7	31.5	NMTR
1976	4	30	-103.11	31.92			0.00	M	57.6	35.8	NMTR
1976	5	1	-103.06	32.37			0.00	M	8.0	5.0	NMTR
1976	5	3	-105.66	32.41			0.00	M	241.7	150.2	NMTR
1976	5	3	-103.20	32.03			0.00	M	47.0	29.2	NMTR
1976	5	3	-103.03	32.03			0.00	M	45.6	28.3	NMTR
1976	5	4	-103.23	31.86			0.00	M	65.3	40.6	NMTR
1976	5	6	-103.18	31.97			0.00	M	53.1	33.0	NMTR
1976	5	6	-103.16	31.87			0.00	M	63.3	39.3	NMTR
1976	5	11	-102.92	32.29			0.00	M	22.2	13.8	NMTR
1976	5	21	-105.59	32.49			0.00	M	234.9	146.0	NMTR
1976	6	14	-102.49	31.52			0.00	M	116.5	72.4	NMTR
1976	6	15	-102.34	31.56			0.00	M	120.0	74.6	NMTR
1976	6	15	-102.37	31.60			0.00	M	115.0	71.5	NMTR
1976	7	28	-102.29	33.02			0.00	M	98.7	61.4	NMTR

Table ISA-10.1 All Earthquakes within 322 km (200 mi) of NEF Site

NEF Site Coordinates			Longitude	Latitude
			-103.0820	32.4360

Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth (km)	(mi)	MAG	MAG Type	Epicentral Distance (km)	(mi)	Data Sources[1]
1976	8	5	-101.73	30.87			0.00	M	216.3	134.4	NMTR
1976	8	5	-103.00	31.60			3.00	M	93.1	57.9	UTIG
1976	8	6	-102.59	31.78			2.10	M	86.3	53.6	NMTR
1976	8	10	-102.03	31.77			0.00	M	123.8	76.9	NMTR
1976	8	10	-102.06	31.79			0.00	M	119.5	74.3	NMTR
1976	8	25	-101.94	31.55			0.00	M	146.1	90.8	NMTR
1976	8	26	-102.01	31.84			0.00	M	120.8	75.1	NMTR
1976	8	30	-101.98	31.57			0.00	M	141.7	88.0	NMTR
1976	8	31	-102.18	31.46			0.00	M	137.4	85.4	NMTR
1976	9	3	-103.48	31.55			2.00	M	105.2	65.4	NMTR
1976	9	5	-102.74	32.23			0.00	M	39.3	24.4	NMTR
1976	9	17	-103.06	32.24			0.00	M	22.4	13.9	NMTR
1976	9	17	-102.50	31.40			3.10	M	127.4	79.2	UTIG
1976	9	19	-104.57	30.47			0.00	M	259.7	161.4	NMTR
1976	10	22	-102.16	31.55			0.00	M	131.6	81.8	NMTR
1976	10	23	-102.38	31.62			0.00	M	112.2	69.7	NMTR
1976	10	25	-102.53	31.84			0.00	M	84.3	52.4	NMTR
1976	10	26	-103.28	31.33			2.40	M	124.2	77.2	NMTR
1976	11	3	-102.27	30.92			0.00	M	185.6	115.3	NMTR
1976	12	12	-102.46	31.57			2.80	M	112.5	69.9	NMTR
1976	12	12	-102.49	31.61			1.90	M	107.3	66.6	NMTR
1976	12	15	-102.22	31.59			1.40	M	124.2	77.2	NMTR
1976	12	18	-103.02	31.62			1.80	M	90.8	56.4	NMTR
1976	12	19	-102.45	31.87			2.20	M	86.0	53.5	NMTR
1976	12	19	-103.14	32.25			1.80	M	20.9	13.0	NMTR
1976	12	19	-103.08	32.27			2.70	M	18.7	11.6	NMTR
1977	1	29	-104.59	30.58			0.00	M	250.3	155.5	NMTR
1977	2	4	-104.70	30.59			0.00	M	256.1	159.2	NMTR
1977	2	18	-103.05	32.24			0.00	M	21.7	13.5	NMTR
1977	3	5	-102.66	31.16			0.00	M	146.9	91.3	NMTR
1977	3	14	-101.01	33.04			0.00	M	204.7	127.2	NMTR
1977	3	20	-103.10	32.21			0.00	M	25.5	15.8	NMTR
1977	3	29	-103.28	31.60			0.00	M	94.2	58.5	NMTR
1977	4	3	-103.17	31.49			1.90	M	105.3	65.5	NMTR
1977	4	3	-103.20	31.47			0.00	M	107.8	67.0	NMTR
1977	4	4	-103.36	31.00			0.00	M	161.4	100.3	NMTR
1977	4	7	-103.05	32.19			0.00	M	27.7	17.2	NMTR
1977	4	7	-102.70	31.32			0.00	M	129.3	80.3	NMTR
1977	4	7	-102.94	31.35			0.00	M	120.9	75.1	NMTR
1977	4	12	-102.55	31.28			0.00	M	137.4	85.4	NMTR
1977	4	17	-102.35	31.50			0.00	M	124.7	77.5	NMTR
1977	4	18	-103.25	31.60			0.00	M	93.7	58.2	NMTR
1977	4	22	-103.02	32.18			0.00	M	28.8	17.9	NMTR
1977	4	25	-102.81	32.07			0.00	M	47.9	29.8	NMTR
1977	4	26	-103.08	31.90	4.0	2.5	3.30	un	59.3	36.8	ANSS
1977	4	28	-102.52	31.83			0.00	M	86.1	53.5	NMTR
1977	4	28	-101.99	31.87			0.00	M	120.6	75.0	NMTR

Table ISA-10.1 All Earthquakes within 322 km (200 mi) of NEF Site

NEF Site Coordinates: Longitude -103.0820, Latitude 32.4360

Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth (km)	Focal Depth (mi)	MAG	MAG Type	Epicentral Distance (km)	Epicentral Distance (mi)	Data Sources[1]
1977	4	29	-102.65	31.77			0.00	M	84.0	52.2	NMTR
1977	6	7	-100.75	33.06	5.0	3.1	4.00	un	228.5	142.0	ANSS
1977	6	8	-100.83	32.83			0.00	M	215.4	133.9	NMTR
1977	6	8	-100.82	32.92			0.00	M	218.4	135.7	NMTR
1977	6	8	-101.04	32.87			0.00	M	196.4	122.1	NMTR
1977	6	17	-100.95	32.90			2.70	M	206.1	128.1	NMTR
1977	6	28	-103.30	31.54			2.30	M	101.6	63.1	NMTR
1977	7	1	-103.34	31.50			2.00	M	106.7	66.3	NMTR
1977	7	11	-102.62	31.80			0.00	M	83.1	51.6	NMTR
1977	7	11	-102.68	31.79			0.00	M	81.4	50.6	NMTR
1977	7	12	-102.64	31.77			0.00	M	84.6	52.6	NMTR
1977	7	18	-102.70	31.78			0.00	M	81.4	50.6	NMTR
1977	7	22	-102.72	31.80			0.00	M	78.2	48.6	NMTR
1977	7	22	-102.70	31.80			3.00	M	79.2	49.2	UTIG
1977	7	24	-102.70	31.79			0.00	M	79.7	49.5	NMTR
1977	8	20	-103.33	31.60			1.90	M	95.7	59.5	NMTR
1977	8	21	-104.91	30.54			0.00	M	272.4	169.3	NMTR
1977	10	13	-100.81	32.91			2.20	M	218.8	135.9	NMTR
1977	10	17	-102.46	31.57			1.80	M	112.6	69.9	NMTR
1977	11	14	-104.96	31.52			0.00	M	203.7	126.6	NMTR
1977	11	27	-101.14	33.02			0.00	M	192.7	119.8	NMTR
1977	11	28	-100.84	32.95	5.0	3.1	3.50	un	217.4	135.1	ANSS
1977	12	16	-102.40	31.52			0.00	M	120.2	74.7	NMTR
1977	12	21	-102.41	31.52			0.00	M	120.3	74.7	NMTR
1977	12	31	-102.46	31.60			2.10	M	109.7	68.2	NMTR
1978	1	2	-102.53	31.60			2.20	M	106.3	66.1	NMTR
1978	1	12	-102.30	31.49			0.00	M	128.1	79.6	NMTR
1978	1	15	-101.70	31.36			0.00	M	177.0	110.0	NMTR
1978	1	18	-103.23	31.61			0.00	M	92.9	57.7	NMTR
1978	1	19	-103.71	32.56			0.00	M	60.5	37.6	NMTR
1978	2	5	-102.60	31.89			0.00	M	76.2	47.4	NMTR
1978	2	5	-104.55	31.41			0.00	M	179.5	111.5	NMTR
1978	2	18	-104.69	31.21			2.30	M	203.8	126.6	NMTR
1978	3	2	-103.06	32.82			1.50	M	42.5	26.4	NMTR
1978	3	2	-102.38	31.58			3.30	M	115.4	71.7	NMTR
1978	3	2	-102.61	31.59			2.10	M	103.9	64.6	NMTR
1978	3	2	-102.56	31.55			3.50	M	109.9	68.3	UTIG
1978	3	19	-102.49	31.47			1.60	M	120.5	74.9	NMTR
1978	6	16	-100.80	33.00			3.40	M	222.1	138.0	UTIG
1978	6	16	-100.77	33.03	10.0	6.2	5.30	un	226.1	140.5	ANSS
1978	6	29	-102.42	31.08			3.20	M	163.1	101.4	NMTR
1978	7	5	-102.20	31.61			0.00	M	123.2	76.5	NMTR
1978	7	18	-104.36	30.36			0.00	M	260.4	161.8	NMTR
1978	7	21	-102.77	31.34			0.00	M	125.0	77.7	NMTR
1978	8	14	-102.18	31.58			2.20	M	127.4	79.2	NMTR
1978	9	29	-102.42	31.52			0.00	M	119.2	74.1	NMTR
1978	9	30	-102.17	31.36			0.00	M	146.7	91.1	NMTR

Table ISA-10.1 All Earthquakes within 322 km (200 mi) of NEF Site

NEF Site Coordinates			Longitude	Latitude							
			-103.0820	32.4360							
Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth (km)	Focal Depth (mi)	MAG	MAG Type	Epicentral Distance (km)	Epicentral Distance (mi)	Data Sources[1]
1978	10	2	-102.43	31.53			0.00	M	117.6	73.1	NMTR
1978	10	2	-102.19	31.51			0.00	M	132.5	82.3	NMTR
1978	10	2	-102.36	31.48			0.00	M	126.4	78.5	NMTR
1978	10	3	-102.99	31.90			0.00	M	59.7	37.1	NMTR
1978	10	6	-102.36	31.55			0.00	M	119.8	74.4	NMTR
1979	4	28	-104.72	30.47			0.00	M	267.7	166.3	NMTR
1979	7	17	-103.73	32.65			2.00	M	65.4	40.6	NMTR
1979	8	3	-100.81	32.87			2.40	M	217.5	135.1	NMTR
1980	1	21	-105.00	34.20			1.30	M	264.2	164.2	NMTR
1980	3	21	-102.34	31.57			1.60	M	118.5	73.6	NMTR
1981	8	13	-102.70	31.90			2.20	M	69.7	43.3	NMTR
1981	9	16	-105.23	33.72			1.80	M	245.2	152.4	NMTR
1982	1	4	-102.49	31.18	5.0	3.1	3.90	un	149.9	93.2	ANSS
1982	4	26	-100.84	33.02	5.0	3.1	2.80	un	218.8	136.0	ANSS
1982	5	1	-103.04	32.33			2.10	M	12.3	7.6	NMTR
1982	10	17	-102.71	30.90			2.00	M	174.0	108.1	NMTR
1982	10	26	-103.59	33.67			1.50	M	144.6	89.8	NMTR
1982	10	26	-103.61	33.63			1.50	M	141.3	87.8	NMTR
1982	11	25	-100.78	32.89			2.30	M	220.7	137.1	NMTR
1982	11	28	-100.84	33.00	5.0	3.1	3.30	un	218.4	135.7	ANSS
1983	1	9	-104.19	30.65			1.90	M	224.3	139.4	NMTR
1983	1	12	-105.19	34.32			1.50	M	286.7	178.2	NMTR
1983	1	29	-102.08	31.75			2.20	M	121.2	75.3	NMTR
1983	3	3	-104.35	29.96			2.80	M	299.6	186.2	NMTR
1983	6	5	-105.35	32.52			1.30	M	212.6	132.1	NMTR
1983	6	21	-103.58	33.63			1.60	M	140.9	87.5	NMTR
1983	7	21	-105.14	30.97			1.60	M	253.4	157.5	NMTR
1983	8	4	-105.14	32.57			1.30	M	193.4	120.2	NMTR
1983	8	19	-102.23	31.31			1.80	M	148.8	92.5	NMTR
1983	8	22	-105.08	34.06			1.30	M	258.6	160.7	NMTR
1983	8	23	-105.52	31.17			2.10	M	269.7	167.6	NMTR
1983	8	26	-102.53	33.62			1.60	M	140.9	87.5	NMTR
1983	8	29	-100.62	31.80			2.60	M	242.0	150.4	NMTR
1983	9	15	-104.43	34.92			3.10	M	302.6	188.1	NMTR
1983	9	29	-104.45	34.89			2.70	M	300.0	186.4	NMTR
1983	9	30	-103.97	30.57			1.70	M	224.0	139.2	NMTR
1983	12	1	-101.99	31.86			1.40	M	121.1	75.3	NMTR
1983	12	3	-103.32	30.97			2.10	M	164.1	102.0	NMTR
1983	12	26	-102.88	30.77			1.70	M	186.4	115.8	NMTR
1984	1	2	-102.12	31.81			1.80	M	114.4	71.1	NMTR
1984	1	3	-102.69	31.21			1.70	M	141.3	87.8	NMTR
1984	1	3	-103.04	30.76			2.00	M	186.3	115.8	NMTR
1984	1	16	-102.20	31.56			1.40	M	127.5	79.2	NMTR
1984	3	2	-104.84	30.81			1.90	M	245.5	152.5	NMTR
1984	3	23	-100.78	32.45			1.50	M	215.2	133.7	NMTR
1984	5	21	-102.59	31.14			1.30	M	151.3	94.0	NMTR
1984	5	21	-102.23	35.07	5.0	3.1	3.10	un	302.5	188.0	ANSS

Table ISA-10.1 All Earthquakes within 322 km (200 mi) of NEF Site

NEF Site Coordinates			Longitude	Latitude							
			-103.0820	32.4360							
Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth (km)	Focal Depth (mi)	MAG	MAG Type	Epicentral Distance (km)	Epicentral Distance (mi)	Data Sources[1]
1984	6	27	-102.48	31.22			2.00	M	146.5	91.0	NMTR
1984	7	17	-105.77	32.85			1.30	M	255.7	158.9	NMTR
1984	8	18	-103.56	30.78			1.80	M	189.8	118.0	NMTR
1984	8	24	-104.48	30.67			1.30	M	236.8	147.1	NMTR
1984	8	26	-104.27	30.38			2.10	M	254.4	158.1	NMTR
1984	9	11	-100.70	31.99	5.0	3.1	3.20	un	229.4	142.5	ANSS
1984	9	19	-100.69	32.03	5.0	3.1	3.00	un	229.3	142.5	ANSS
1984	9	27	-103.42	32.59			1.60	M	36.0	22.4	NMTR
1984	10	4	-102.70	33.58			1.30	M	132.3	82.2	NMTR
1984	10	4	-102.24	31.65			1.30	M	118.4	73.6	NMTR
1984	10	11	-100.56	31.95			2.40	M	243.2	151.1	NMTR
1984	10	27	-104.56	30.62			1.70	M	245.1	152.3	NMTR
1984	11	27	-105.41	33.57			1.60	M	250.6	155.7	NMTR
1984	12	4	-101.93	30.10			2.30	M	281.6	175.0	NMTR
1984	12	4	-103.21	32.64			2.10	M	25.4	15.8	NMTR
1984	12	4	-103.56	32.27	5.0	3.1	2.90	un	48.3	30.0	ANSS
1984	12	12	-105.61	33.36			1.50	M	256.9	159.6	NMTR
1985	2	21	-100.75	32.88			1.40	M	223.3	138.7	NMTR
1985	2	21	-100.81	32.72			1.50	M	214.6	133.4	NMTR
1985	3	9	-105.12	33.97			1.30	M	254.4	158.1	NMTR
1985	5	3	-104.95	31.04			1.90	M	234.5	145.7	NMTR
1985	6	1	-102.83	31.06			1.50	M	154.6	96.0	NMTR
1985	6	2	-102.28	31.18			1.60	M	158.7	98.6	NMTR
1985	6	12	-103.90	34.64			1.60	M	255.9	159.0	NMTR
1985	8	2	-104.34	32.48			1.40	M	118.0	73.3	NMTR
1985	9	5	-103.77	33.66			1.80	M	150.1	93.3	NMTR
1985	9	18	-103.42	30.90			2.00	M	173.1	107.6	NMTR
1985	10	21	-101.88	32.04			1.30	M	121.3	75.4	NMTR
1985	11	13	-103.08	32.10			1.80	M	37.8	23.5	NMTR
1985	11	28	-101.99	31.61			1.80	M	138.2	85.9	NMTR
1985	12	5	-102.94	32.42			1.60	M	13.9	8.6	NMTR
1986	1	25	-100.73	32.06	5.0	3.1	2.90	un	224.3	139.4	ANSS
1986	1	30	-104.01	33.54			1.90	M	150.1	93.3	NMTR
1986	1	30	-100.69	32.07	5.0	3.1	3.30	un	228.0	141.7	ANSS
1986	2	7	-105.44	32.54			1.40	M	221.0	137.3	NMTR
1986	2	14	-100.76	31.53			2.60	M	240.9	149.7	NMTR
1986	3	1	-102.57	31.16			1.70	M	149.6	92.9	NMTR
1986	3	11	-105.08	32.11			2.00	M	190.7	118.5	NMTR
1986	3	21	-105.64	33.43			1.60	M	262.8	163.3	NMTR
1986	5	28	-105.12	31.76			1.60	M	205.8	127.9	NMTR
1986	6	12	-102.22	31.77			1.80	M	109.6	68.1	NMTR
1986	6	27	-102.01	32.06			2.20	M	109.3	67.9	NMTR
1986	7	9	-102.48	31.55			1.60	M	113.3	70.4	NMTR
1986	7	20	-105.00	33.47			1.50	M	212.8	132.2	NMTR
1986	8	2	-103.79	33.68			1.70	M	153.4	95.3	NMTR
1986	8	6	-103.03	33.86			2.40	M	158.4	98.5	NMTR
1986	8	14	-104.66	32.53			1.30	M	148.0	92.0	NMTR

Table ISA-10.1 All Earthquakes within 322 km (200 mi) of NEF Site

NEF Site Coordinates			Longitude -103.0820	Latitude 32.4360							
Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth (km)	Focal Depth (mi)	MAG	MAG Type	Epicentral Distance (km)	Epicentral Distance (mi)	Data Sources[1]
1986	8	15	-103.43	33.14			1.70	M	84.2	52.3	NMTR
1986	8	29	-102.41	31.31			1.40	M	140.1	87.1	NMTR
1986	9	18	-102.37	31.51			1.80	M	123.2	76.5	NMTR
1986	10	18	-102.69	30.07			1.60	M	265.4	164.9	NMTR
1986	10	25	-102.13	31.60			1.70	M	129.0	80.2	NMTR
1986	11	3	-104.64	31.09			2.00	M	209.5	130.2	NMTR
1986	11	6	-104.58	32.55			1.60	M	140.4	87.2	NMTR
1986	11	17	-100.73	33.08			2.00	M	230.6	143.3	NMTR
1986	11	24	-102.16	31.68			2.00	M	121.1	75.3	NMTR
1986	12	6	-102.16	31.59			2.40	M	127.6	79.3	NMTR
1986	12	6	-102.23	31.47			2.10	M	133.9	83.2	NMTR
1986	12	6	-102.17	31.65			1.70	M	122.0	75.8	NMTR
1986	12	6	-102.09	31.72			2.20	M	122.6	76.2	NMTR
1986	12	15	-103.19	35.07			1.50	M	292.9	182.0	NMTR
1986	12	15	-102.02	31.76			1.50	M	125.0	77.7	NMTR
1987	1	25	-104.86	31.74			1.70	M	184.3	114.5	NMTR
1987	2	9	-103.45	30.69			2.30	M	196.8	122.3	NMTR
1987	2	9	-101.96	31.86			1.60	M	123.6	76.8	NMTR
1987	2	12	-101.94	31.66			1.60	M	137.9	85.7	NMTR
1987	2	17	-104.52	30.60			2.10	M	244.8	152.1	NMTR
1987	3	2	-105.08	30.78			1.80	M	263.6	163.8	NMTR
1987	3	3	-105.44	31.17			1.50	M	263.4	163.7	NMTR
1987	3	10	-105.66	31.13			1.50	M	282.7	175.7	NMTR
1987	3	26	-103.28	30.96			2.60	M	165.2	102.6	NMTR
1987	3	31	-104.95	31.52			2.80	M	203.4	126.4	NMTR
1987	4	23	-105.02	32.03			1.60	M	187.7	116.7	NMTR
1987	4	25	-105.22	33.97			1.90	M	261.2	162.3	NMTR
1987	4	29	-105.92	32.67			2.30	M	267.0	165.9	NMTR
1987	7	5	-104.77	30.85			2.00	M	237.5	147.6	NMTR
1987	7	23	-103.03	35.29			1.90	M	316.9	196.9	NMTR
1987	7	30	-103.87	34.54			1.50	M	244.4	151.9	NMTR
1987	8	4	-102.12	31.87			1.70	M	110.1	68.4	NMTR
1987	9	11	-103.62	33.61			2.00	M	139.1	86.4	NMTR
1987	9	21	-103.74	33.68			1.80	M	150.6	93.6	NMTR
1987	10	1	-105.16	30.47			1.60	M	294.1	182.7	NMTR
1987	10	1	-103.76	33.66			1.50	M	150.0	93.2	NMTR
1987	10	9	-104.59	31.07			1.40	M	208.4	129.5	NMTR
1987	10	31	-105.31	32.86			1.30	M	213.8	132.9	NMTR
1987	11	3	-103.71	33.70			1.30	M	151.6	94.2	NMTR
1987	11	17	-101.97	32.06			1.60	M	112.9	70.1	NMTR
1987	12	6	-102.76	31.83			1.60	M	74.2	46.1	NMTR
1987	12	20	-103.07	32.29			2.20	M	15.8	9.8	NMTR
1987	12	28	-102.25	31.47			2.10	M	133.3	82.8	NMTR
1987	12	29	-102.11	31.58			1.50	M	132.1	82.1	NMTR
1988	1	26	-102.42	31.24			2.30	M	146.4	90.9	NMTR
1988	2	14	-102.06	31.78			1.40	M	121.0	75.2	NMTR
1988	2	21	-103.02	30.45			1.40	M	220.3	136.9	NMTR

Table ISA-10.1 All Earthquakes within 322 km (200 mi) of NEF Site

NEF Site Coordinates			Longitude -103.0820	Latitude 32.4360							
Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth (km)	(mi)	MAG	MAG Type	Epicentral Distance (km)	(mi)	Data Sources[1]
1988	2	27	-103.75	33.67			1.80	M	150.3	93.4	NMTR
1988	3	9	-102.44	31.24			1.70	M	146.0	90.7	NMTR
1988	3	15	-105.52	31.72			1.30	M	242.7	150.8	NMTR
1988	3	17	-102.20	31.66			1.60	M	119.8	74.4	NMTR
1988	4	5	-102.33	31.44			2.10	M	131.6	81.8	NMTR
1988	4	6	-102.09	31.94			1.30	M	107.9	67.1	NMTR
1988	5	3	-104.39	30.52			1.30	M	246.2	153.0	NMTR
1988	5	10	-105.20	30.96			1.40	M	258.4	160.6	NMTR
1988	5	27	-102.12	31.78			1.30	M	116.1	72.1	NMTR
1988	5	27	-102.02	32.06			1.30	M	108.3	67.3	NMTR
1988	7	4	-100.74	33.74			2.00	M	261.5	162.5	NMTR
1988	7	11	-103.25	35.28			1.90	M	316.6	196.7	NMTR
1988	7	20	-102.43	29.77			2.20	M	301.9	187.6	NMTR
1988	7	25	-104.91	31.98			1.50	M	178.9	111.2	NMTR
1988	7	26	-105.14	30.94			1.50	M	255.5	158.8	NMTR
1988	8	23	-102.02	32.26			1.50	M	101.1	62.8	NMTR
1988	9	15	-103.32	31.68			1.50	M	86.7	53.9	NMTR
1988	9	19	-102.45	32.46			2.00	M	59.3	36.8	NMTR
1988	10	2	-103.79	33.63			1.30	M	147.8	91.8	NMTR
1988	11	10	-102.40	31.55			1.90	M	117.3	72.9	NMTR
1989	1	9	-102.59	31.44			1.80	M	119.6	74.3	NMTR
1989	1	9	-102.12	31.78			1.30	M	116.5	72.4	NMTR
1989	1	20	-101.97	32.08			1.90	M	112.1	69.6	NMTR
1989	2	21	-103.39	35.29			2.30	M	318.4	197.8	NMTR
1989	3	19	-103.55	31.19			1.50	M	145.2	90.2	NMTR
1989	3	21	-102.33	31.42			1.50	M	133.5	83.0	NMTR
1989	3	30	-102.86	33.24			1.40	M	91.5	56.9	NMTR
1989	6	5	-102.09	32.10			2.10	M	100.1	62.2	NMTR
1989	6	23	-102.23	31.59			1.60	M	123.2	76.6	NMTR
1989	6	28	-105.08	30.93			2.30	M	252.3	156.8	NMTR
1989	7	13	-105.27	33.53			1.50	M	237.1	147.3	NMTR
1989	7	24	-100.93	32.92			1.60	M	208.3	129.5	NMTR
1989	7	25	-101.76	30.90			2.10	M	211.2	131.3	NMTR
1989	8	8	-102.70	31.30			2.30	M	131.3	81.6	NMTR
1989	8	16	-101.96	31.70			1.60	M	133.3	82.8	NMTR
1989	9	5	-102.50	34.25			2.50	M	208.9	129.8	NMTR
1989	11	2	-100.94	33.02			2.00	M	210.4	130.7	NMTR
1989	11	16	-103.12	35.11			2.60	M	296.7	184.4	NMTR
1989	12	7	-103.67	34.58			1.40	M	244.1	151.7	NMTR
1989	12	28	-101.06	31.70			2.10	M	207.6	129.0	NMTR
1989	12	28	-100.96	32.04			1.70	M	203.9	126.7	NMTR
1990	1	16	-105.32	31.74			1.80	M	224.4	139.4	NMTR
1990	3	4	-103.92	30.53			1.70	M	226.3	140.6	NMTR
1990	3	30	-100.53	32.96			2.30	M	245.1	152.3	NMTR
1990	3	30	-100.56	32.99			2.20	M	243.5	151.3	NMTR
1990	4	6	-103.36	31.51			1.90	M	106.3	66.0	NMTR
1990	5	10	-102.37	31.14			2.20	M	159.2	98.9	NMTR

Table ISA-10.1 All Earthquakes within 322 km (200 mi) of NEF Site

NEF Site Coordinates			Longitude	Latitude
			-103.0820	32.4360

Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth (km)	(mi)	MAG	MAG Type	Epicentral Distance (km)	(mi)	Data Sources[1]
1990	5	10	-101.96	32.13			1.60	M	110.9	68.9	NMTR
1990	5	16	-102.04	31.86			2.40	M	117.2	72.8	NMTR
1990	5	22	-102.09	30.24			2.20	M	261.5	162.5	NMTR
1990	6	22	-100.76	32.58			2.20	M	218.3	135.7	NMTR
1990	7	3	-102.22	31.44			1.50	M	137.6	85.5	NMTR
1990	7	13	-101.81	34.86			2.70	M	293.9	182.6	NMTR
1990	8	3	-100.69	32.21			3.40	M	225.6	140.2	NMTR
1990	8	9	-102.67	31.21			1.90	M	141.8	88.1	NMTR
1990	8	14	-102.26	31.39			1.80	M	139.8	86.9	NMTR
1990	8	25	-102.01	31.91			1.80	M	116.0	72.1	NMTR
1990	10	8	-105.12	30.94			1.30	M	254.0	157.8	NMTR
1990	12	20	-103.14	35.27			2.50	M	315.1	195.8	NMTR
1991	1	1	-105.27	32.44			1.60	M	205.4	127.6	NMTR
1991	1	29	-103.04	32.89			1.40	M	50.8	31.6	NMTR
1991	2	3	-104.49	32.81			1.30	M	137.7	85.6	NMTR
1991	2	3	-103.96	35.00			2.10	M	296.2	184.0	NMTR
1991	3	10	-103.97	30.47			2.10	M	234.3	145.6	NMTR
1991	3	10	-103.33	33.58			2.00	M	128.8	80.0	NMTR
1991	4	8	-103.13	34.98			2.10	M	282.4	175.5	NMTR
1991	5	16	-103.75	33.67			2.00	M	150.4	93.5	NMTR
1991	6	4	-102.31	32.05			2.00	M	83.9	52.1	NMTR
1991	7	16	-101.12	33.09			2.10	M	197.3	122.6	NMTR
1991	8	1	-104.02	34.59			2.70	M	254.6	158.2	NMTR
1991	8	7	-104.81	31.62			1.80	M	186.1	115.6	NMTR
1991	8	17	-100.99	32.09			2.00	M	200.2	124.4	NMTR
1991	9	22	-101.30	31.32			2.10	M	209.2	130.0	NMTR
1991	9	28	-103.77	33.63			1.70	M	147.3	91.6	NMTR
1991	9	30	-100.73	31.85			2.20	M	230.5	143.2	NMTR
1991	10	5	-105.41	31.38			2.20	M	248.6	154.5	NMTR
1992	1	2	-103.19	32.30			5.00	M	17.8	11.0	NMTR
1992	1	2	-103.19	32.30			1.80	M	17.8	11.0	NMTR
1992	1	2	-103.19	32.30			1.50	M	17.8	11.0	NMTR
1992	1	2	-103.19	32.30			2.40	M	17.8	11.0	NMTR
1992	1	2	-103.19	32.30			1.80	M	17.8	11.0	NMTR
1992	1	3	-103.19	32.30			1.90	M	17.8	11.0	NMTR
1992	1	4	-103.19	32.30			1.50	M	17.8	11.0	NMTR
1992	1	7	-103.19	32.30			2.40	M	17.8	11.0	NMTR
1992	1	9	-103.19	32.30			2.80	M	17.8	11.0	NMTR
1992	1	11	-103.19	32.30			2.00	M	17.8	11.0	NMTR
1992	1	23	-102.29	31.84			1.90	M	99.2	61.7	NMTR
1992	2	2	-102.86	32.17			1.90	M	36.4	22.6	NMTR
1992	3	15	-104.12	34.92			1.70	M	292.1	181.5	NMTR
1992	3	28	-105.39	33.45			1.80	M	242.2	150.5	NMTR
1992	4	3	-103.03	32.26			2.10	M	19.9	12.4	NMTR
1992	4	6	-102.61	31.86			1.70	M	77.7	48.3	NMTR
1992	4	7	-102.29	31.56			1.60	M	122.6	76.2	NMTR
1992	4	7	-102.29	31.56			2.30	M	122.6	76.2	NMTR

Table ISA-10.1 All Earthquakes within 322 km (200 mi) of NEF Site

NEF Site Coordinates			Longitude	Latitude							
			-103.0820	32.4360							
Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth (km)	Focal Depth (mi)	MAG	MAG Type	Epicentral Distance (km)	Epicentral Distance (mi)	Data Sources[1]
1992	4	7	-102.29	31.56			1.70	M	122.6	76.2	NMTR
1992	4	8	-104.86	32.41			1.60	M	166.9	103.7	NMTR
1992	4	30	-104.31	30.66			1.70	M	229.0	142.3	NMTR
1992	5	9	-104.34	30.49			1.60	M	246.7	153.3	NMTR
1992	5	15	-103.08	32.28			1.60	M	17.5	10.9	NMTR
1992	5	16	-102.34	31.75			1.70	M	103.0	64.0	NMTR
1992	6	14	-103.10	32.30			2.30	M	15.1	9.4	NMTR
1992	6	20	-102.42	31.43			1.60	M	127.5	79.2	NMTR
1992	6	20	-102.42	31.43			1.50	M	127.5	79.2	NMTR
1992	6	29	-102.47	31.42			1.40	M	126.9	78.8	NMTR
1992	6	29	-102.47	31.42			1.40	M	126.9	78.8	NMTR
1992	6	29	-102.47	31.42			2.00	M	126.9	78.8	NMTR
1992	7	5	-102.39	31.88			1.50	M	89.4	55.6	NMTR
1992	7	5	-102.39	31.88			1.30	M	89.4	55.6	NMTR
1992	7	21	-103.13	32.28			1.90	M	17.8	11.1	NMTR
1992	8	12	-102.41	31.39			1.50	M	131.9	82.0	NMTR
1992	8	18	-102.45	31.46			1.90	M	123.5	76.7	NMTR
1992	8	19	-100.92	33.11			2.20	M	215.3	133.8	NMTR
1992	8	26	-102.71	32.17	5.0	3.1	3.00	un	45.6	28.4	ANSS
1992	8	28	-100.98	32.38			1.70	M	197.4	122.6	NMTR
1992	9	4	-102.26	31.42			1.90	M	136.8	85.0	NMTR
1992	9	15	-103.02	32.16			2.20	M	31.6	19.6	NMTR
1992	10	8	-102.81	32.25			1.60	M	33.1	20.6	NMTR
1992	10	10	-102.41	31.71			1.60	M	102.2	63.5	NMTR
1992	10	27	-101.93	34.12			1.30	M	215.1	133.7	NMTR
1992	11	22	-103.16	32.29			1.70	M	18.0	11.2	NMTR
1992	11	27	-102.49	31.44			1.30	M	124.0	77.1	NMTR
1992	12	2	-102.35	31.42			2.40	M	131.5	81.7	NMTR
1992	12	3	-103.74	33.66			1.90	M	149.6	93.0	NMTR
1992	12	5	-102.51	31.87			1.40	M	83.0	51.6	NMTR
1993	1	4	-105.27	31.06			1.30	M	256.5	159.4	NMTR
1993	1	28	-102.58	31.85			1.80	M	80.3	49.9	NMTR
1993	1	31	-104.64	30.60			1.50	M	250.8	155.9	NMTR
1993	2	11	-105.23	31.12			2.00	M	250.1	155.4	NMTR
1993	2	28	-102.43	31.21			1.30	M	149.4	92.8	NMTR
1993	2	28	-102.41	31.22			1.50	M	149.3	92.8	NMTR
1993	3	8	-103.33	30.87			1.60	M	175.9	109.3	NMTR
1993	3	21	-102.37	31.43			1.50	M	130.4	81.0	NMTR
1993	4	23	-102.47	31.21			1.70	M	147.8	91.9	NMTR
1993	5	5	-105.16	32.29			2.10	M	195.3	121.4	NMTR
1993	5	16	-105.06	30.44			2.20	M	290.1	180.2	NMTR
1993	5	17	-102.33	31.42			2.30	M	133.3	82.9	NMTR
1993	5	23	-102.42	31.42			1.60	M	128.7	80.0	NMTR
1993	5	28	-103.12	32.75			2.50	M	34.6	21.5	NMTR
1993	6	17	-102.56	31.80			1.70	M	86.5	53.8	NMTR
1993	6	23	-102.44	31.51			1.40	M	119.5	74.2	NMTR
1993	6	23	-102.54	31.43			2.50	M	123.2	76.6	NMTR

Table ISA-10.1 All Earthquakes within 322 km (200 mi) of NEF Site

NEF Site Coordinates			Longitude	Latitude							
			-103.0820	32.4360							
Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth (km)	(mi)	MAG	MAG Type	Epicentral Distance (km)	(mi)	Data Sources[1]
1993	6	23	-102.52	31.43			2.80	M	123.2	76.5	NMTR
1993	6	23	-102.52	31.43			2.10	M	123.2	76.5	NMTR
1993	6	23	-102.54	29.66			1.90	M	312.3	194.0	NMTR
1993	6	23	-102.51	31.35	5.0	3.1	2.80	un	132.5	82.3	ANSS
1993	6	24	-102.45	31.48			2.10	M	121.9	75.7	NMTR
1993	7	3	-102.43	31.44			1.50	M	126.7	78.7	NMTR
1993	7	3	-102.34	31.50			2.20	M	125.5	78.0	NMTR
1993	7	3	-102.38	31.54			1.60	M	119.3	74.1	NMTR
1993	8	13	-102.52	31.89			1.30	M	80.1	49.8	NMTR
1993	8	29	-102.91	32.35			2.50	M	19.0	11.8	NMTR
1993	9	5	-100.96	32.28			2.00	M	200.1	124.4	NMTR
1993	9	6	-100.91	32.48			1.80	M	203.6	126.5	NMTR
1993	9	11	-103.76	34.72			1.50	M	260.9	162.1	NMTR
1993	9	26	-103.52	35.08			1.50	M	296.6	184.3	NMTR
1993	9	30	-103.80	33.64			1.90	M	149.0	92.6	NMTR
1993	10	3	-103.84	33.61			1.70	M	148.5	92.3	NMTR
1993	11	6	-102.19	31.75			1.50	M	113.6	70.6	NMTR
1993	11	24	-104.74	32.34			1.30	M	156.2	97.1	NMTR
1993	11	25	-102.10	34.27			2.60	M	223.0	138.5	NMTR
1993	11	25	-104.38	30.49			1.30	M	248.6	154.5	NMTR
1993	12	2	-102.34	31.27			1.30	M	147.3	91.5	NMTR
1993	12	3	-102.23	31.68			1.60	M	115.6	71.8	NMTR
1993	12	10	-102.29	31.74			1.60	M	106.8	66.4	NMTR
1993	12	18	-103.41	30.21			1.80	M	249.5	155.0	NMTR
1993	12	22	-105.68	33.33	10.0	6.2	3.20	un	261.9	162.8	ANSS
1994	1	6	-105.09	31.95			2.40	M	196.3	122.0	NMTR
1994	1	7	-102.32	31.24			1.70	M	151.0	93.8	NMTR
1994	3	15	-103.56	30.11			2.00	M	261.9	162.8	NMTR
1994	4	21	-103.12	32.31			1.40	M	14.1	8.8	NMTR
1994	4	25	-104.62	30.60			1.90	M	250.5	155.7	NMTR
1994	5	23	-102.64	32.11			1.60	M	55.0	34.2	NMTR
1994	6	30	-102.33	31.36			1.30	M	138.6	86.2	NMTR
1994	8	22	-102.21	33.34			1.60	M	129.0	80.2	NMTR
1994	8	30	-102.32	31.38			1.40	M	137.3	85.3	NMTR
1994	8	30	-102.32	31.34			1.50	M	141.5	87.9	NMTR
1994	8	30	-102.30	31.42			1.30	M	135.1	84.0	NMTR
1994	9	24	-102.36	31.43			2.00	M	131.1	81.4	NMTR
1994	11	24	-100.80	32.39			2.70	M	214.3	133.2	NMTR
1995	1	1	-102.45	31.77			1.40	M	94.7	58.8	NMTR
1995	1	4	-102.38	31.48			1.30	M	125.0	77.6	NMTR
1995	2	1	-104.09	34.51			1.80	M	248.7	154.6	NMTR
1995	3	19	-104.21	35.00	5.0	3.1	3.30	un	303.1	188.4	ANSS
1995	4	14	-103.35	30.28			5.70	M	240.7	149.5	UTIG
1995	4	14	-103.35	30.28			3.30	M	240.7	149.5	UTIG
1995	4	14	-103.35	30.30	10.0	6.2	2.70	un	238.5	148.2	ANSS
1995	4	14	-103.35	30.30	10.0	6.2	2.80	un	238.5	148.2	ANSS
1995	4	14	-103.35	30.30	10.0	6.2	3.30	un	238.5	148.2	ANSS

Table ISA-10.1 All Earthquakes within 322 km (200 mi) of NEF Site

NEF Site Coordinates	Longitude	Latitude
	-103.0820	32.4360

Year	Month	Day	Longitude (^{0}W)	Latitude (^{0}N)	Focal Depth (km)	(mi)	MAG	MAG Type	Epicentral Distance (km)	(mi)	Data Sources[1]
1995	4	14	-103.35	30.30	10.0	6.2	2.60	un	238.5	148.2	ANSS
1995	4	14	-103.35	30.30	10.0	6.2	2.40	un	238.5	148.2	ANSS
1995	4	14	-103.35	30.30	10.0	6.2	2.70	un	238.5	148.2	ANSS
1995	4	14	-103.35	30.30	10.0	6.2	2.40	un	238.5	148.2	ANSS
1995	4	14	-103.35	30.30	10.0	6.2	2.80	un	238.5	148.2	ANSS
1995	4	14	-103.35	30.30	10.0	6.2	2.90	un	238.5	148.2	ANSS
1995	4	14	-103.35	30.30	10.0	6.2	2.30	un	238.5	148.2	ANSS
1995	4	15	-103.35	30.28			4.00	M	240.7	149.5	UTIG
1995	4	15	-103.35	30.30	10.0	6.2	2.40	un	238.5	148.2	ANSS
1995	4	15	-103.32	30.27	10.0	6.2	4.00	un	241.4	150.0	ANSS
1995	4	16	-103.35	30.30	10.0	6.2	2.30	un	238.5	148.2	ANSS
1995	4	16	-103.35	30.30	10.0	6.2	2.50	un	238.5	148.2	ANSS
1995	4	16	-103.35	30.30	10.0	6.2	2.40	un	238.5	148.2	ANSS
1995	4	17	-103.35	30.30	10.0	6.2	2.50	un	238.5	148.2	ANSS
1995	4	18	-102.27	31.44			1.90	M	134.5	83.6	NMTR
1995	4	18	-105.34	31.10			1.60	M	259.8	161.4	NMTR
1995	4	21	-103.35	30.30	10.0	6.2	2.90	un	238.5	148.2	ANSS
1995	5	11	-105.20	32.71			2.40	M	200.4	124.5	NMTR
1995	5	15	-102.42	31.40			1.80	M	131.1	81.5	NMTR
1995	5	27	-102.34	31.34			2.30	M	140.1	87.0	NMTR
1995	5	30	-105.21	32.71			2.10	M	200.9	124.8	NMTR
1995	6	1	-103.35	30.30	10.0	6.2	3.50	un	238.5	148.2	ANSS
1995	7	6	-103.35	30.30	10.0	6.2	2.70	un	238.5	148.2	ANSS
1995	7	6	-103.35	30.30	10.0	6.2	2.60	un	238.5	148.2	ANSS
1995	7	11	-105.06	30.87			1.80	M	255.5	158.8	NMTR
1995	7	17	-104.94	31.15			1.40	M	226.0	140.4	NMTR
1995	8	1	-105.27	33.14			1.30	M	218.9	136.0	NMTR
1995	8	2	-103.36	30.31			1.80	M	237.2	147.4	NMTR
1995	8	12	-103.07	30.79			1.90	M	183.1	113.8	NMTR
1995	8	14	-102.96	30.41			1.50	M	225.3	140.0	NMTR
1995	10	19	-104.84	32.05			2.00	M	170.4	105.9	NMTR
1995	10	25	-103.42	30.35			2.20	M	233.6	145.2	NMTR
1995	11	12	-103.35	30.30	10.0	6.2	3.60	ML	238.5	148.2	ANSS
1995	12	3	-104.90	31.93			1.50	M	180.1	111.9	NMTR
1995	12	4	-104.90	31.93			1.40	M	180.1	111.9	NMTR
1995	12	4	-104.90	31.93			1.30	M	180.1	111.9	NMTR
1996	3	15	-105.69	33.59	10.0	6.2	2.90	ML	274.6	170.6	ANSS
1998	4	15	-103.30	30.19	10.0	6.2	3.60	ML	250.4	155.6	ANSS
1999	3	1	-104.66	32.57	1.0	0.6	2.90	ML	148.1	92.0	ANSS
1999	3	14	-104.63	32.59	1.0	0.6	4.00	ML	145.9	90.7	ANSS
1999	3	17	-104.67	32.58	1.0	0.6	3.50	Mc	149.7	93.0	ANSS
1999	5	30	-104.66	32.58	10.0	6.2	3.90	ML	148.9	92.5	ANSS
1999	8	9	-104.59	32.57	5.0	3.1	2.90	Mc	142.0	88.3	ANSS
2000	2	2	-104.63	32.58	5.0	3.1	2.70	ML	145.7	90.5	ANSS
2000	2	26	-103.61	30.24	5.0	3.1	2.80	ML	248.6	154.5	ANSS
2001	6	2	-103.14	32.33	5.0	3.1	3.30	ML	12.6	7.8	ANSS
2001	11	22	-102.63	31.79	5.0	3.1	3.10	ML	83.7	52.0	ANSS

Table ISA-10.1 All Earthquakes within 322 km (200 mi) of NEF Site

NEF Site Coordinates			Longitude -103.0820	Latitude 32.4360							
Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth (km)	(mi)	MAG	MAG Type	Epicentral Distance (km)	(mi)	Data Sources[1]
2002	9	17	-104.63	32.58	10.0	6.2	3.50	ML	145.8	90.6	ANSS
2002	9	17	-104.63	32.58	10.0	6.2	3.30	ML	145.8	90.6	ANSS
2003	6	21	-104.51	32.67	5.0	3.1	3.60	ML	135.5	84.2	ANSS

[1] Data Sources
UTIG - University of Texas Institute for Geophysics
NMTH - New Mexico Tech Historical Catalog
NMTR - New Mexico Tech Regional Catalog, Exclusive of Socorro NM Events
ANSS - Advanced National Seismic System

Table ISA-10.2 Earthquakes of Magnitude 3.0 and Greater within 322 km (200 mi) of NEF Site

NEF Site Coordinates	Longitude	Latitude
	-103.0820	32.4360

Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth (km)	Focal Depth (mi)	MAG	MAG Type	Epicentral Distance (km)	Epicentral Distance (mi)	Data Sources[1]
1931	8	16	-104.60	30.70			3.60	M	240.3	149.3	UTIG
1931	8	16	-104.60	30.70			3.00	M	240.3	149.3	UTIG
1931	8	16	-104.60	30.70			6.00	M	240.3	149.3	UTIG
1931	8	16	-104.60	30.70			3.00	M	240.3	149.3	UTIG
1931	8	16	-104.60	30.70			3.30	M	240.3	149.3	UTIG
1931	8	16	-104.60	30.70			3.60	M	240.3	149.3	UTIG
1931	8	18	-104.60	30.70			3.00	M	240.3	149.3	UTIG
1931	8	18	-104.60	30.70			4.20	M	240.3	149.3	UTIG
1931	8	18	-104.60	30.70			3.00	M	240.3	149.3	UTIG
1931	8	26	-104.60	30.70			3.60	M	240.3	149.3	UTIG
1931	11	3	-104.60	30.70			3.00	M	240.3	149.3	UTIG
1949	5	23	-105.20	34.60			4.50	M	310.0	192.6	NMTH
1955	1	27	-104.50	30.60			3.30	M	244.0	151.6	UTIG
1962	3	6	-104.80	31.20			3.50	M	212.3	131.9	UTIG
1963	12	19	-104.27	34.82			3.40	M	287.0	178.3	NMTR
1964	11	8	-103.10	31.90			3.00	M	59.5	37.0	UTIG
1964	11	21	-103.10	31.90			3.10	M	59.5	37.0	UTIG
1965	2	3	-103.10	31.90			3.30	M	59.5	37.0	UTIG
1965	8	30	-103.00	31.90			3.50	M	60.0	37.3	UTIG
1966	8	14	-103.00	31.90			3.40	M	60.0	37.3	UTIG
1966	11	26	-105.44	30.95			3.50	M	277.5	172.4	NMTR
1971	7	30	-103.00	31.72	10.0	6.2	3.00	mb	79.9	49.6	ANSS
1971	7	31	-103.06	31.70	10.0	6.2	3.40	mb	81.4	50.6	ANSS
1971	9	24	-103.20	31.60			3.20	M	93.5	58.1	UTIG
1972	7	26	-104.01	32.57			3.10	M	88.3	54.9	NMTR
1973	8	2	-105.56	31.04			3.60	M	280.7	174.5	NMTR
1973	8	4	-103.22	35.11			3.00	M	296.6	184.3	NMTR
1974	11	28	-104.14	32.31	5.0	3.1	3.90	mb	100.4	62.4	ANSS
1974	12	30	-103.10	30.90			3.70	M	170.5	106.0	UTIG
1975	2	2	-103.19	35.05			3.00	M	290.7	180.6	NMTR
1975	8	1	-104.00	31.40			3.00	M	143.9	89.4	UTIG
1975	12	12	-102.31	31.61			3.00	M	117.5	73.0	NMTR
1976	1	19	-103.09	31.90			3.50	M	59.5	37.0	UTIG
1976	1	25	-103.08	31.90	2.0	1.2	3.90	un	59.3	36.8	ANSS
1976	8	5	-103.00	31.60			3.00	M	93.1	57.9	UTIG
1976	9	17	-102.50	31.40			3.10	M	127.4	79.2	UTIG
1977	4	26	-103.08	31.90	4.0	2.5	3.30	un	59.3	36.8	ANSS
1977	6	7	-100.75	33.06	5.0	3.1	4.00	un	228.5	142.0	ANSS
1977	7	22	-102.70	31.80			3.00	M	79.2	49.2	UTIG
1977	11	28	-100.84	32.95	5.0	3.1	3.50	un	217.4	135.1	ANSS
1978	3	2	-102.38	31.58			3.30	M	115.4	71.7	NMTR
1978	3	2	-102.56	31.55			3.50	M	109.9	68.3	UTIG
1978	6	16	-100.80	33.00			3.40	M	222.1	138.0	UTIG
1978	6	16	-100.77	33.03	10.0	6.2	5.30	un	226.1	140.5	ANSS
1978	6	29	-102.42	31.08			3.20	M	163.1	101.4	NMTR
1982	1	4	-102.49	31.18	5.0	3.1	3.90	un	149.9	93.2	ANSS
1982	11	28	-100.84	33.00	5.0	3.1	3.30	un	218.4	135.7	ANSS

Table ISA-10.2 Earthquakes of Magnitude 3.0 and Greater within 322 km (200 mi) of NEF Site

NEF Site Coordinates			Longitude	Latitude							
			-103.0820	32.4360							
Year	Month	Day	Longitude (°W)	Latitude (°N)	Focal Depth (km)	Focal Depth (mi)	MAG	MAG Type	Epicentral Distance (km)	Epicentral Distance (mi)	Data Sources[1]
1983	9	15	-104.43	34.92			3.10	M	302.6	188.1	NMTR
1984	5	21	-102.23	35.07	5.0	3.1	3.10	un	302.5	188.0	ANSS
1984	9	11	-100.70	31.99	5.0	3.1	3.20	un	229.4	142.5	ANSS
1984	9	19	-100.69	32.03	5.0	3.1	3.00	un	229.3	142.5	ANSS
1986	1	30	-100.69	32.07	5.0	3.1	3.30	un	228.0	141.7	ANSS
1990	8	3	-100.69	32.21			3.40	M	225.6	140.2	NMTR
1992	1	2	-103.19	32.30			5.00	M	17.8	11.0	NMTR
1992	8	26	-102.71	32.17	5.0	3.1	3.00	un	45.6	28.4	ANSS
1993	12	22	-105.68	33.33	10.0	6.2	3.20	un	261.9	162.8	ANSS
1995	3	19	-104.21	35.00	5.0	3.1	3.30	un	303.1	188.4	ANSS
1995	4	14	-103.35	30.28			5.70	M	240.7	149.5	UTIG
1995	4	14	-103.35	30.28			3.30	M	240.7	149.5	UTIG
1995	4	14	-103.35	30.30	10.0	6.2	3.30	un	238.5	148.2	ANSS
1995	4	15	-103.35	30.28			4.00	M	240.7	149.5	UTIG
1995	4	15	-103.32	30.27	10.0	6.2	4.00	un	241.4	150.0	ANSS
1995	6	1	-103.35	30.30	10.0	6.2	3.50	un	238.5	148.2	ANSS
1995	11	12	-103.35	30.30	10.0	6.2	3.60	ML	238.5	148.2	ANSS
1998	4	15	-103.30	30.19	10.0	6.2	3.60	ML	250.4	155.6	ANSS
1999	3	14	-104.63	32.59	1.0	0.6	4.00	ML	145.9	90.7	ANSS
1999	3	17	-104.67	32.58	1.0	0.6	3.50	Mc	149.7	93.0	ANSS
1999	5	30	-104.66	32.58	10.0	6.2	3.90	ML	148.9	92.5	ANSS
2001	6	2	-103.14	32.33	5.0	3.1	3.30	ML	12.6	7.8	ANSS
2001	11	22	-102.63	31.79	5.0	3.1	3.10	ML	83.7	52.0	ANSS
2002	9	17	-104.63	32.58	10.0	6.2	3.50	ML	145.8	90.6	ANSS
2002	9	17	-104.63	32.58	10.0	6.2	3.30	ML	145.8	90.6	ANSS
2003	6	21	-104.51	32.67	5.0	3.1	3.60	ML	135.5	84.2	ANSS

[1] Data Sources
UTIG - University of Texas Institute for Geophysics
NMTH - New Mexico Tech Historical Catalog
NMTR - New Mexico Tech Regional Catalog, Exclusive of Socorro NM Events
ANSS - Advanced National Seismic System

DECOMMISSIONING COST TABLES

Table D-1.1A
Number and Dimensions of Facility Components

Separations Modules (see Note 1)

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes			
Fume Cupboards			
Lab Benches			
Sinks			
Drains			
Floors			
Walls			
Ceilings			
Ventilation/Ductwork			
Hot Cells			
Equipment/Materials			
Soil Plots			
Storage Tanks			
Storage Areas			
Radwaste Areas			
Scrap Recovery Areas			
Maintenance Shop			
Equipment Decontamination Areas			
Other			

Notes:

1. More than 97% of the decommissioning costs for the facility are attributed to the dismantling, decontamination, processing, and disposal of centrifuges and other equipment in the Separations Building Modules, which are considered classified. Given the classified nature of these buildings, the data presented in these Tables have been structured to meet the applicable NUREG-1757 recommendations, to the extent practicable. However, specific information such as numbers of components and unit rates has been intentionally excluded to protect the classified nature of the data. Classified data will be submitted under separate cover.

Table D-1.1B
Number and Dimensions of Facility Components

Decommission Decontamination Facility

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes	None	NA	NA
Fume Cupboards	None	NA	NA
Lab Benches	10	Various sizes of lab and workshop benches ranging from 6.5 to 13 feet long by 2.5 feet wide	see Note 1
Sinks	6	Standard laboratory sinks and hand wash basins	see Note 1
Drains	6	Standard laboratory type drains	see Note 1
Floors	1 Lot (see Note 2)	see Note 1	see Note 1
Walls	1 Lot (see Note 2)	see Note 1	see Note 1
Ceilings	1 Lot (see Note 2)	see Note 1	see Note 1
Ventilation/Ductwork	see Note 3	Various sizes of ductwork ranging from 3 to 18 inches plus dampers, valves and flexibles	640 feet
Hot Cells	None	NA	NA
Equipment/Materials	20	Various pieces of equipment including citric cleaning tanks, centrifuge cutting machines	see Note 1
Soil Plots	None	NA	NA
Storage Tanks	1 Lot (see Note 2)	Various storage tanks	see Note 1
Storage Areas	1	Storage area for centrifuges and pipe work	see Note 1
Radwaste Areas	None	NA	NA
Scrap Recovery Areas	None	NA	NA
Maintenance Shop	None	NA	NA
Equipment Decontamination Areas	None	NA	NA
Other	1 Lot (see Note 2)	Hand tools and consumables that become contaminated while carrying out dismantling/decontamination work, unmeasured work and scaffolding	see Note 1

Notes:

1. Total dimensions not used in estimating model.
2. Allocation based on Urenco decommissioning experience.
3. Total dimensions provided.

Table D-1.1C
Number and Dimensions of Facility Components

Technical Services Building

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes	None	NA	NA
Fume Cupboards	18	Standard laboratory fume cupboards, approx 6.5 – 8 feet high x 5 feet wide	see Note 1
Lab Benches	25	Various sizes of lab and workshop benches ranging from 6.5 – 13 feet long by 2.5 feet wide	see Note 1
Sinks	12	Standard laboratory sinks and hand wash basins plus larger sinks for laundry	see Note 1
Drains	12	Standard Laboratory type drains plus larger laundry drain	see Note 1
Floors	see Note 3	Floor area covers all Workshops and Labs in the Technical Services Bldg that may be exposed to contamination	26,340 ft^2
Walls	see Note 3	Wall area covers all Workshops and Labs in the Technical Services Bldg that may be exposed to contamination	40,074 ft^2
Ceilings	see Note 3	Ceiling area covers all Workshops and Labs in the Technical Services Bldg that may be exposed to contamination	26,340 ft^2
Ventilation ductwork	see Note 3	Various pieces of equipment including, filter banks, extractor fans, vent stack, dampers and approx 2,034 feet of large and small ductwork	2,034 feet
Hot Cells	None	NA	NA
Equipment/Materials	57	Various pieces of equipment including, mass spectrometers, washing machines, hydraulic lift tables, cleaning cabinets	see Note 1
Soil Plots	None	NA	NA
Storage Tanks	1	Waste oil storage tank (53 gal)	see Note 1
Storage Areas	2	Storage area for product removal, dirty pumps	see Note 1
Radwaste Areas	None	NA	NA
Scrap Recovery Areas	None	NA	NA
Maintenance Shop	None	NA	NA
Equipment Decontamination Areas	None	NA	NA
Other	1 Lot (see Note 2)	Hand tools and consumables that become contaminated while carrying out dismantling/decontamination work, unmeasured work and scaffolding	see Note 1

Notes: 1. Total dimensions not used in estimating model.
2. Allocation based on Urenco decommissioning experience.
3. Total dimensions provided.

Table D-1.1D
Number and Dimensions of Facility Components

Gaseous Effluent Vent System Throughout Plant

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes	None	NA	NA
Fume Cupboards	None	NA	NA
Lab Benches	None	NA	NA
Sinks	None	NA	NA
Drains	None	NA	NA
Floors	None	NA	NA
Walls	None	NA	NA
Ceilings	None	NA	NA
Ventilation/Ductwork	see Note 3	Various sizes of ductwork ranging from 3 to 18 inches plus dampers, valves and flexibles	5,656 feet
Hot Cells	None	NA	NA
Equipment/Materials	None	NA	NA
Soil Plots	None	NA	NA
Storage Tanks	None	NA	NA
Storage Areas	None	NA	NA
RadWaste Areas	None	NA	NA
Scrap Recovery Areas	None	NA	NA
Maintenance Shop	None	NA	NA
Equipment Decontamination Areas	None	NA	NA
Other	1 Lot (see Note 2)	Hand tools and consumables that become contaminated while carrying out dismantling/decontamination work, unmeasured work and scaffolding	see Note 1

Notes:
1. Total dimensions not used in estimating model.
2. Allocation based on Urenco decommissioning experience.
3. Total dimensions provided.

Table D-1.1E
Number and Dimensions of Facility Components

Blending & Sampling

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes	None	NA	NA
Fume Cupboards	None	NA	NA
Lab Benches	None	NA	NA
Sinks	None	NA	NA
Drains	None	NA	NA
Floors	None (see Note 4)	NA	NA
Walls	None (see Note 4)	NA	NA
Ceilings	None (see Note 4)	NA	NA
Ventilation/Ductwork	Covered in GEV system estimate	Covered in GEV system estimate	Covered in GEV system estimate
Hot Cells	None	NA	NA
Equipment/Materials	see Note 3	Various sizes of pipe-work ranging from DN25 to DN65	2,461 feet
	38 Valves	Various types of valve ranging from 0.6 to 2.5 inches and manual to control	see Note 1
	12	Various pieces of equipment including hot boxes and traps	see Note 1
Soil Plots	None	NA	NA
Storage Tanks	None	NA	NA
Storage Areas	None	NA	NA
Radwaste Areas	None	NA	NA
Scrap Recovery Areas	None	NA	NA
Maintenance Shop	None	NA	NA
Equipment Decontamination Areas	None	NA	NA
Other	1 Lot (see Note 2)	Hand tools and consumables that become contaminated while carrying out dismantling/decontamination work, unmeasured work and scaffolding	see Note 1

Notes:
1. Total dimensions not used in estimating model.
2. Allocation based on Urenco decommissioning experience.
3. Total dimensions provided.
4. No floors, walls or ceilings are anticipated needing decontamination.

Table D-1.1F
Number and Dimensions of Facility Components

Test & Post Mortem

Component	Number of Components	Dimensions of Components	Total Dimensions
Glove Boxes	None	NA	NA
Fume Cupboards	None	NA	NA
Lab Benches	4	Various sizes of lab and workshop benches ranging from 6.5 – 13 feet long by 2.5 feet wide	see Note 1
Sinks	2	Standard laboratory sinks and hand wash basins plus larger sinks for laundry	see Note 1
Drains	2	Standard laboratory type drains plus larger laundry drain	see Note 1
Floors	None (see Note 4)	NA	NA
Walls	None (see Note 4)	NA	NA
Ceilings	None (see Note 4)	NA	NA
Ventilation/Ductwork	None	NA	NA
Hot Cells	None	NA	NA
Equipment/Materials	see Note 3	Various sizes of pipe-work ranging from DN16 to DN40	164 feet
	56 Valves	Various types of valve ranging from 0.6 to 1.6 inches and manual to control	see Note 1
	7	Various pieces of equipment including feed take off vessels and traps	see Note 1
Soil Plots	None	NA	NA
Storage Tanks	None	NA	NA
Storage Areas	None	NA	NA
Radwaste Areas	None	NA	NA
Scrap Recovery Areas	None	NA	NA
Maintenance Shop	None	NA	NA
Equipment Decontamination Areas	None	NA	NA
Other	1 Lot (see Note 2)	Hand tools and consumables that become contaminated while carrying out dismantling/decontamination work, unmeasured work and scaffolding	see Note 1

Notes:
1. Total dimensions not used in estimating model.
2. Allocation based on Urenco decommissioning experience.
3. Total dimensions provided.
4. No floors, walls or ceilings are anticipated needing decontamination.

Table D-1.2
Planning and Preparation (Note 1)

Activity	Costs ($000)	Activity Duration (Months)	Notes
Project Plan & Schedule	100	4	–
Site Characterization Plan	200	4	–
Site Characterization	300	4	–
Decommissioning Plan	350	6	–
NRC Review Period	50	12	–
Site Services Specifications	100	2	–
Project Procedures	100	4	–
TOTAL	1,200	36	see Note 2

Notes:

1. Deviates from NUREG-1757 because costs are derived from activity durations based on recent commercial decommissioning cost estimates.
2. Some activities will be conducted in parallel to achieve a 24 month time frame.

Table D-1.3
Decontamination or Dismantling of Radioactive Components (Man Hours) – Other Buildings (Note 1)

Component	Decon Method (Note 4)	Craftsman	Supervision (Note 2)	Project Man (Note 5)	HP&S/Chem (Notes 3 & 5)
Glove Boxes		0	0		
Fume Cupboards		312	62		
Lab Benches		324	64		
Sinks		101	20		
Drains		102	20		
Floors		647	129		
Walls		422	84		
Ceilings		275	55		
Ventilation/Ductwork		8,468	1,693		
Hot Cells		0	0		
Equipment/Materials		1,533	307		
Soil Plots		0	0		
Storage Tanks		14	3		
Storage Areas		110	22		
Radwaste Areas		0	0		
Scrap Recovery Areas		0	0		
Maintenance Shop		0	0		
Equipment Decontamination Areas		0	0		
Other		1,913	382		
TOTAL Hours	–	14,221	2,841	2,430	2,990

Notes:

(1) Includes the Decontamination Facility, Technical Services Building, Gaseous Effluent Vent System Throughout Plant, Blending & Sampling, and Test & Post Mortem Facilities.
(2) Supervision @ 20%.
(3) Supply ongoing monitoring & analysis service for dismantling teams.
(4) Decontamination method not defined at this time.
(5) Total hours allocated based on Urenco decommissioning experience.

Table D-1.4
Restoration of Contaminated Areas on Facility Grounds (Work Days)

Activity	Labor Category	Labor Category	Labor Category	Labor Category	Labor Category	Labor Category
Backfill and Restore Site (see Note 1)						
TOTAL						

Note:
1. Deviates from NUREG-1757 because cost is based on volume and unit cost associated with removal and disposal of liners and earthen covers of the facility evaporative basins. The cost (see Table D-1.14) assumes transport and disposal of approximately 33,000 ft^3 of contaminated soil and basin membrane. Other contaminated areas outside of the plant buildings are not expected.

Table D-1.5
Final Radiation Survey (Note 1)

Activity	Costs ($000)	Activity Duration (Months)	Notes
Prepare Survey Plans & Grid Areas	500	8	--
Collect Survey Readings & Analyze Data	1,400	16	--
Final Status Survey Report & NRC Review	300	8	--
Confirmatory Survey & Report	200	6	--
Terminate Site License	100	2	--
TOTAL	2,500	40	see Note 2

Notes:
1. Deviates from NUREG-1757 because cost are derived from activity durations based on recent commercial decommissioning cost estimates.
2. Some activities will be conducted in parallel to achieve a 36 month time frame.

Table D-1.6
Site Stabilization and Long-Term Surveillance (Work Days)

Activity	Labor Category	Labor Category	Labor Category	Labor Category	Labor Category	Labor Category
(see Note 1)						

Note:
1. Site stabilization and long-term surveillance will not be required.

Table D-1.7
Total Work Days by Labor Category (Based on a 7.5 Hr Working Day)

Task	Shift- worker (multi-functional)	Craftsman	Supervision	Project Man	HP&S	Cleaner
Planning and Preparation (see Table D-1.2)						
Decontamination and/or Dismantling of Radioactive Facility Components	56,067	1,896	6,156	1,478	1,828	2,897
Restoration of Contaminated Ares on Facility Grounds (see Table D-1.4)						
Final Radiation Survey (see Table D-1.5)						
Site Stabilization and Long-Term Surveillance (see Table D-1.6)						

Table D-1.8
Worker Unit Cost Schedule

Labor Cost Component	Shift- worker (multi-functional)	Craftsman	Supervision	Project Man	HP&S	Cleaner
Salary & Fringe ($/year)	73,006	65,184	96,000	120,000	96,000	73,006
Overhead Rate (%)	excluded	excluded	excluded	excluded	excluded	excluded
Total Cost Per Year ($)	73,006	65,184	96,000	120,000	96,000	73,006
Total Cost Per Work Day* ($/day)	342	306	450	563	450	342

*Based on 213.33 work days per year @ 7.5 Hrs per day (1600 Hrs per year)

Table D-1.9
Total Labor Costs by Major Decommissioning Task ($000)

Task	Shift-worker (multi-functional)	Craftsman	Supervision	Project Man	HP&S	Cleaner
Planning and Preparation (see Table D-1.2)						
Decontamination and/or Dismantling of Radioactive Facility Components	19,175	579	2,770	832	823	991
Restoration of Contaminated Ares on Facility Grounds (see Table D-1.4)						
Final Radiation Survey (see Table D-1.5)						
Site Stabilization and Long-Term Surveillance (see Table D-1.6)						

Table D-1.10
Packaging, Shipping and Disposal of Radioactive Wastes (Excluding Labor Costs)

(a) Packing Material Costs (see Note 1)

Waste Type	Volume m³ (ft³)	Number of Containers	Type of Container	Unit Cost of Container	Total Packaging Costs ($000)
TOTAL					

Note:
1. Included in waste disposal costs.

(b) Shipping Costs (see Note 1)

Waste Type	Number of Truckloads	Unit Cost per truckload ($/truck)	Surcharges ($/mile)	Overweight Charges ($/mile)	Distance Shipped (miles)	Total Shipping Costs ($000)
TOTAL						

Note:
1. Included in waste disposal costs.

(c) Waste Disposal Costs (includes packaging & shipping costs)

Waste Type	Disposal Volume m³ (ft³)	Unit Cost ($/ft³)	# of drums	Total Disposal Costs ($000)
Other Buildings :				
Miscellaneous low level waste	83 (2.930)	150	400	440
Separation Modules:				
Solidified Liquid Wastes	432 (15,251)	100	2,159	1,525
Centrifuge Components, Piping and Other Parts	1,036 (36,595)	100	5,180	3,659
Aluminum	3,602 (127,200)	100	NA	12,720
TOTAL	5,153 (181,976)	--	7,739	18,344

(d) Processing Costs

Materials	Disposal Weight (tons)	Unit Cost ($/lb)	Total Disposal Costs ($000)
Aluminum	10,177	0.14	2,860
Other materials	155	2.67	830
Total	10,332	--	3,690

Table D-1.11
Equipment & Supply Costs (Excluded Containers)

a) Equipment

Equipment	Quantity	Unit Cost ($/unit)	Total Cost Equipment ($000)
Separation Modules			
Building:			
Dismantling & decontamination building	45210 ft^2	1,545	6,490
Special floor & vent system	45210 ft^2	294	1,240
Plant equipment			
Basic decontamination equipment	lot (see Note 1)	600,000	600
Decontamination line equipment	2 units	3,908,850	7,820
Evaporation installation	lot (see Note 1)	390,000	390
Radiation and control equipment	lot (see Note 1)	410,000	410
Electrical & Instrumentation			
Electrical system	lot (see Note 1)	500,000	500
Instrumentation	lot (see Note 1)	590,000	590
Design & Engineering			
Building	-	20% (see Note 1)	1,550
Plant & equipment	-	15% (see Note 1)	1,400
Electrical & Instrumentation	-	25% (see Note 1)	270
Other Buildings:			
Dismantling/Cleaning Tools, Equipment and Consumables	lot (see Note 1)	100,000	100
Total	--	--	21,360

Notes: 1. Allocation based on Urenco decommissioning experience.

b) Supply

Equipment	Quantity	Unit Cost ($/$ft^3$)	Total Cost Equipment ($000)
Electricity kwh	2,910,344	0.062	180
Gas ft^3	16,900,000	0.004	75
Water ft^3	86,300	0.035	3
Materials	lot (see Note 1)		653
Total	--	--	910

Notes: 1. Allocation based on Urenco decommissioning experience.

Table D-1.12
Laboratory Costs

Activity	Quantity	Unit Cost ($)	Total Costs ($000)
Analysis of samples	931	934	870
Total	--	--	870

Table D-1.13
Period Dependent Costs

Cost Item	Total Cost ($000)
License Fees	See Note 1
Insurance	See Note 1
Taxes	See Note 1
Other	See Note 1
TOTAL	10,000

Note 1: Period Dependent Costs include management, insurance, taxes, and other costs for the period beginning with the termination of operations of Separations Building Module 3 and the remaining plant facilities. This assumes $2,000,000 per year for each of the five years at the end of the project. It has been assumed that the period dependent decommissioning costs incurred during concurrent enrichment operations will be funded from operating plant funding and not the decommissioning trust fund.

Table D-1.14
Total Decommissioning Costs (see Note 7) - Page 1 of 2

Task/Components	Costs ($000)		Total ($000)	Percentage	Notes
	Separations Modules	Other Buildings			
Planning and Preparation (see Table D-1.2)	1,200	0	1,200	1%	1
Decontamination and Dismantling of Radioactive Facility Components (see Table D-1.9)	24,060	1,110	25,170	30%	8
Restoration of Contamination Areas on Facility Grounds (see Table D-1.4)	1,000	0	1,000	1%	2
Final Radiation Survey (see Table D-1.5)	2,500	0	2,500	3%	3
Site Stabilization and Long-term Surveillance	0	0	0	0%	4
Waste Processing Costs (see Table D-1.10)	3,690	0	3,690	4%	5
Waste Disposal Costs (see Table D-1.10)	17,904	440	18,344	22%	6
Equipment Costs (see Table D-1.11)	21,260	100	21,360	25%	–
Supply Costs (see Table D-1.11)	910	0	910	1%	–
Laboratory Costs (see Table D-1.12)	870	0	870	1%	–
Period Dependent Costs (see Table D-1.13)	10,000	0	10,000	12%	–
SUBTOTAL	83,394	1,650	85,044		–
Contingency (25%)	20,849	413	21,262		–
TOTAL	104,243	2,063	106,306		–
Tails Disposition	0	0	731,181		9
GRAND TOTAL	–	–	837,487		10

Table D-1.14
Total Decommissioning Costs - Page 2 of 2

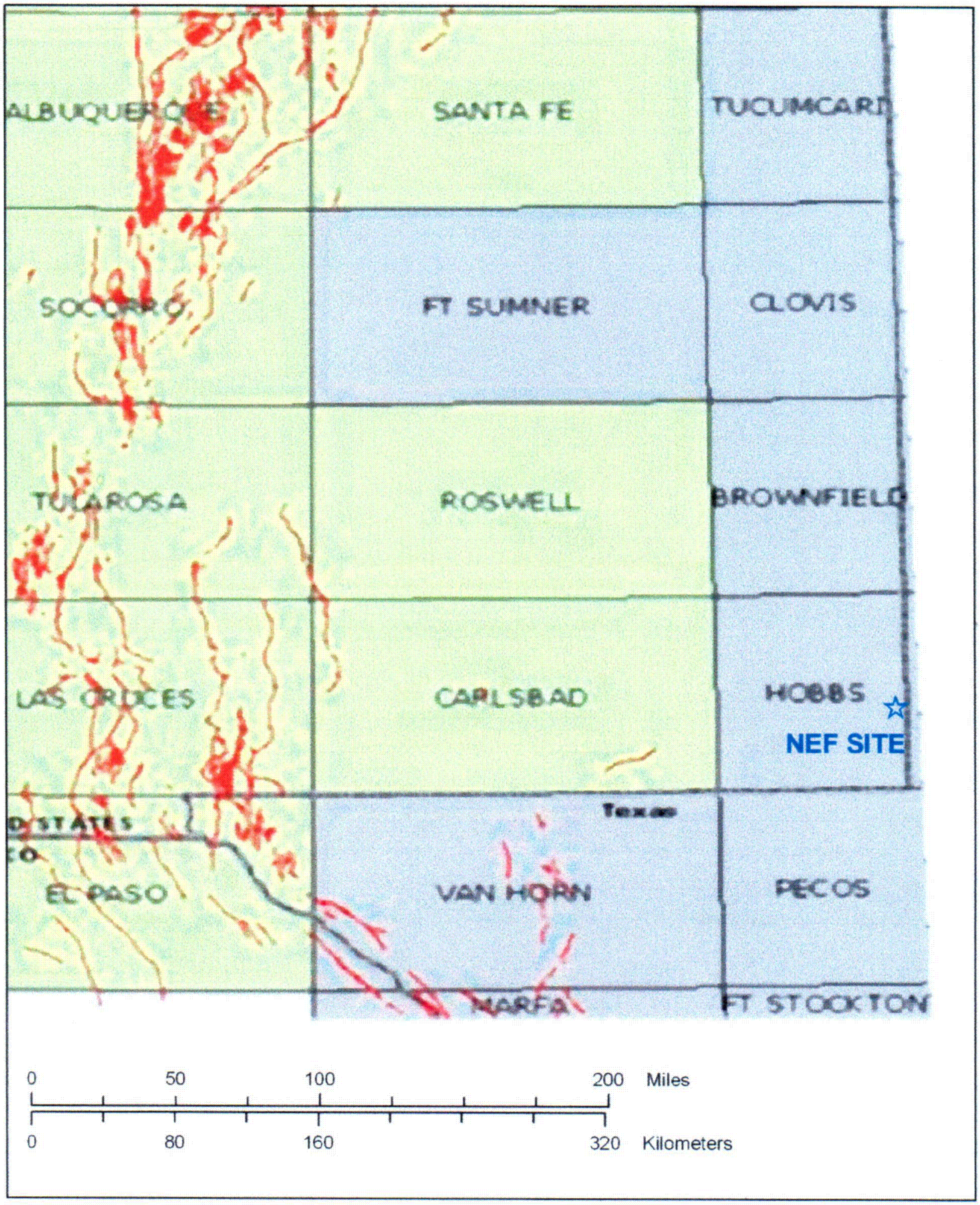
Notes:

1. The $1,200 includes planning, site characterization, Decommissioning Plan preparation, and NRC review for the entire plant.
2. Cost provided is for removal and disposal of liners and earthen covers of the facility evaporative basins. The cost assumes transport and disposal of approximately 33,000 ft^3 of contaminated soil and basin membrane at recent commercial rates. Other contaminated areas outside of the plant buildings are not expected.
3. The $2,500 includes the Final Radiation Survey, NRC review, confirmatory surveys and license termination for the entire plant.
4. Site stabilization and long-term surveillance will not be required.
5. Waste processing costs are based on commercial metal melting equipment and unit rates obtained from Urenco experience in Europe.
6. Includes waste packaging and shipping costs. Waste disposal costs for Other Buildings are based on a $150 per cubic foot unit rate which includes packaging, shipping and disposal at Envirocare in Utah.
7. More than 97% of the decommissioning costs for the facility are attributed to the dismantling, decontamination, processing, and disposal of centrifuges and other equipment in the Separations Building Modules, which are considered classified. Given the classified nature of these buildings, the data presented in these Tables have been structured to meet the applicable NUREG-1757 recommendations, to the extent practicable. However, specific information such as numbers of components and unit rates has been intentionally excluded to protect the classified nature of the data. The remaining 3% of the decommissioning costs are for the remaining systems and components in Other Buildings.
8. The $1,110 for Other Buildings includes the decontamination and dismantling of contaminated equipment in the TBS, Blending and Liquid Sampling Area, Centrifuge Test and Post Mortem Facilities, and Gaseous Effluent Vent System.
9. Refer to Section 10.3, for Tails Disposition discussion.
10. Combined total for both decommissioning and tails disposition.

ATTACHMENT 3

Louisiana Energy Services
Response to April 19, 2004
Request for Additional Information

Figures Referenced from Responses



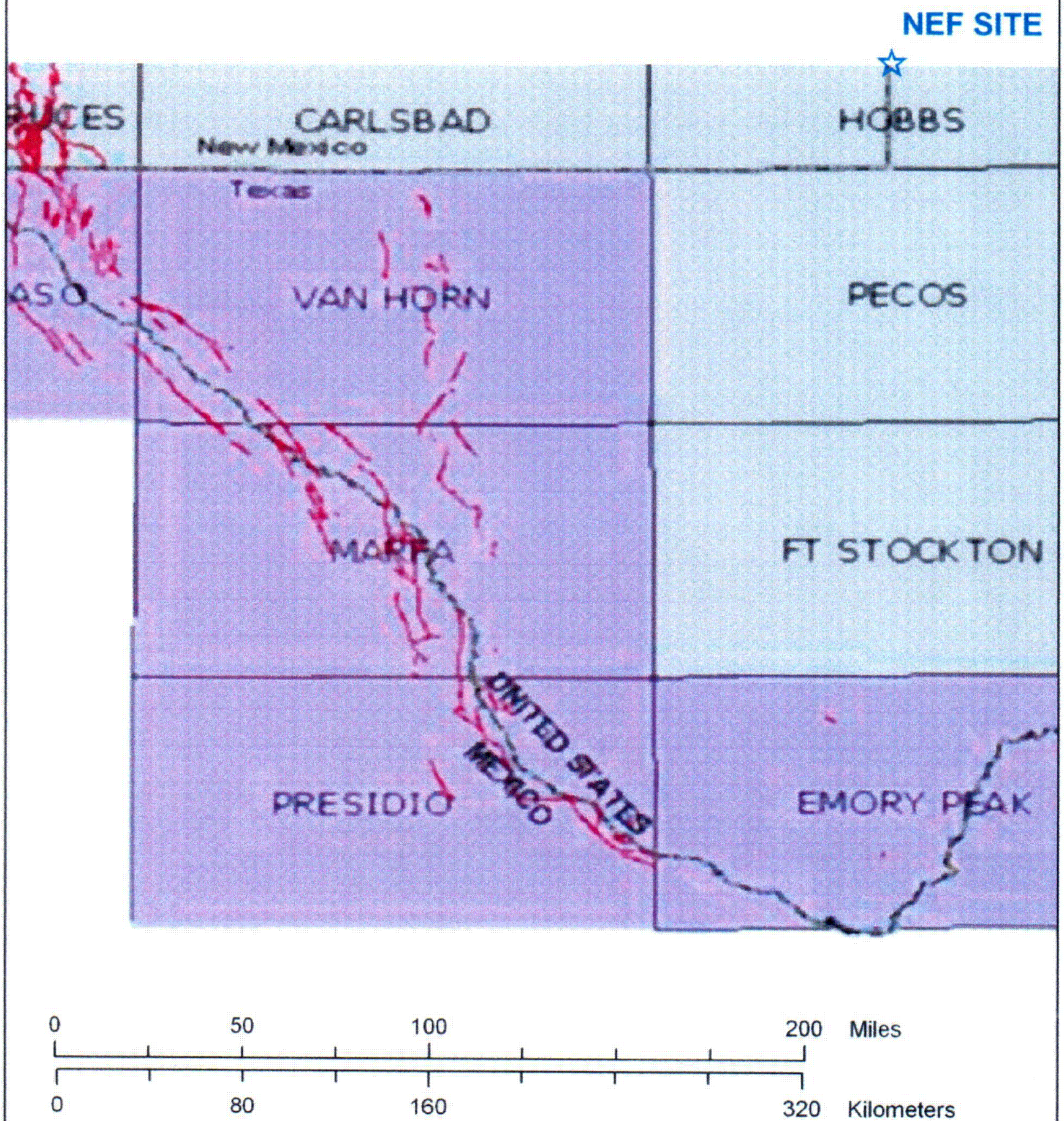
SOURCE: Earthquake Hazards Program Quaternary Fault and Fold Database (USGS, 2004)

Figure ISA-9.1

DATE: 5 / 19 / 04

QUATERNARY FAULTS IN NEW MEXICO

C01



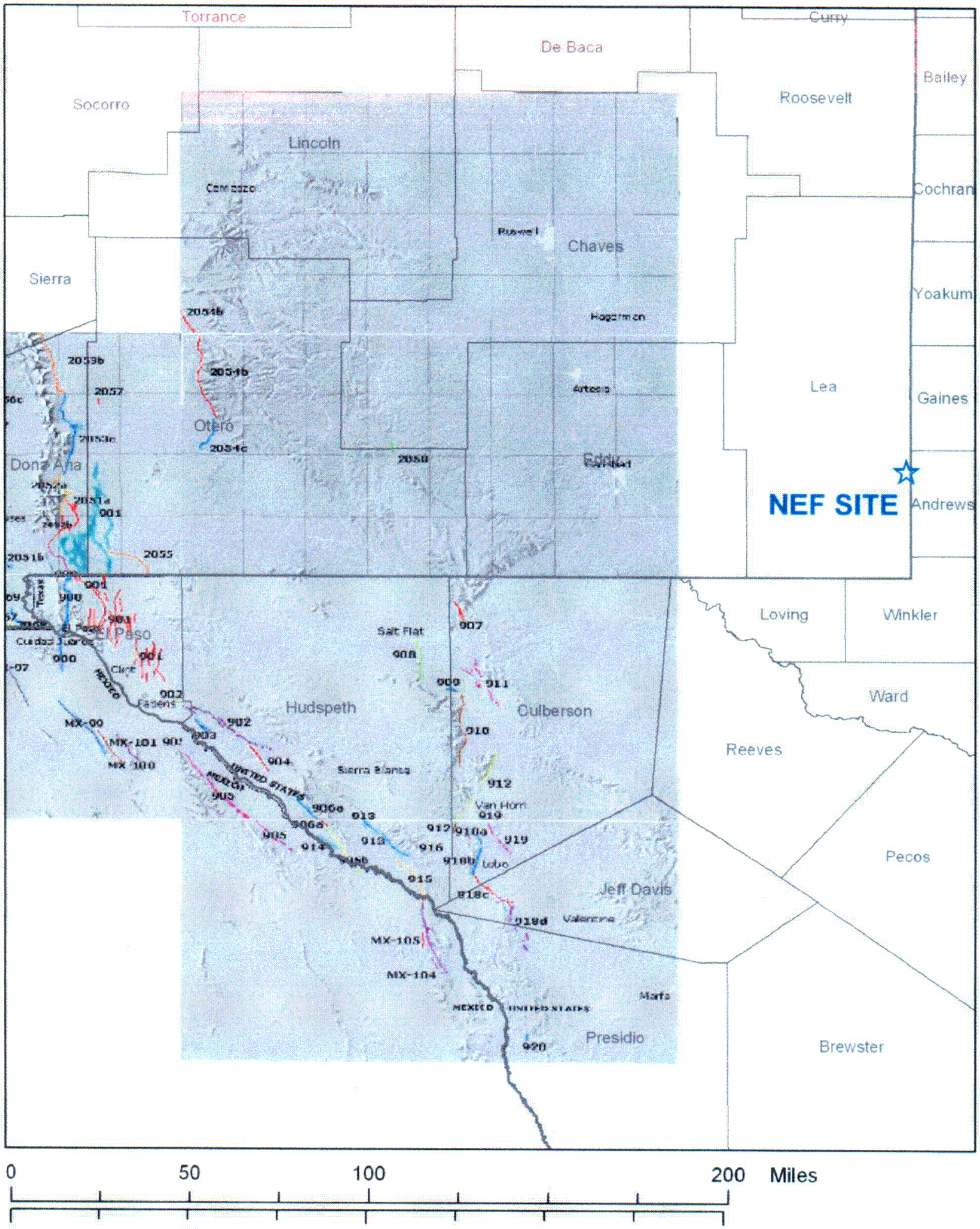
SOURCE: Earthquake Hazards Program
Quaternary Fault and Fold Database
(USGS, 2004)

Figure ISA-9.2

DATE: 5 / 19 / 04

QUATERNARY FAULTS IN TEXAS

C02



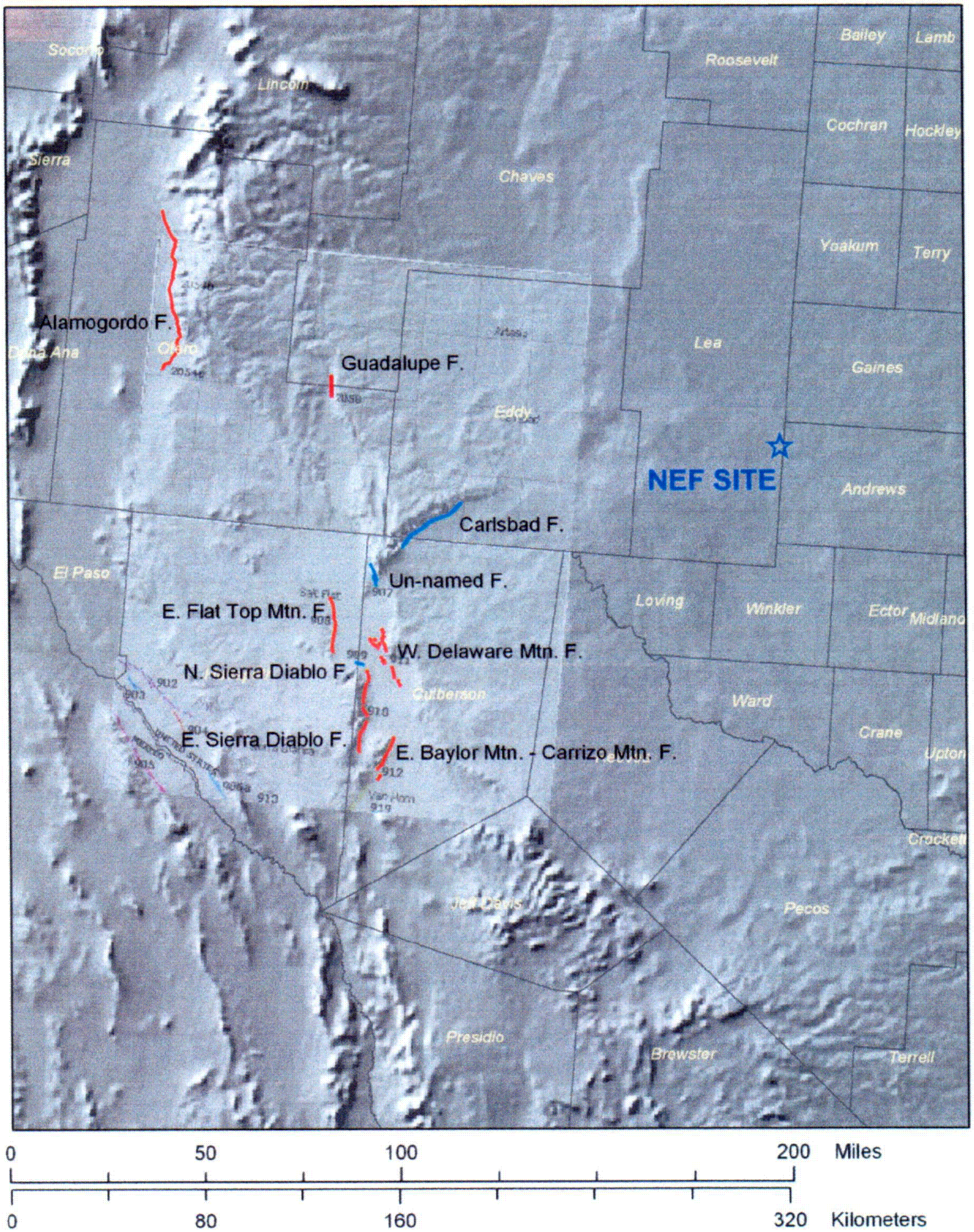
SOURCE: USGS, 2004; Machette, 2000

Figure ISA-9.3

DATE: 5 / 19 / 04

QUATERNARY FAULTS WITHIN 322 km (200 mi) OF NEF SITE

C03



NOTE: Locations of nearest capable faults (red traces) and older faults (blue traces)

Figure ISA-9.4

DATE: 5 / 19 / 04

LOCATIONS OF NEAREST FAULTS TO THE NEF SITE

C04

Figure removed under 10 CFR 2.390.

Figure removed under 10 CFR 2.390.

Figure removed under 10 CFR 2.390.

ATTACHMENT 4

Louisiana Energy Services
Response to April 19, 2004
Request for Additional Information

LES Procedure
DP-ISA-1.1, "IROFS Boundary Definition"

Title: IROFS Boundary Definition			
	[signature] 5/17/04 Approval Date Functional Manager		[signature] 5/17/04 Approval Date Vice President, Licensing, Safety, and Nuclear Engineering
Procedure No. DP-ISA-1.1	Revision No. 0	Effective Date	Page No. 1 of 49

Training Level	B

Procedure Change History

- Initial Issue

Impact of Change

- No Impact

1.0 PURPOSE

This procedure provides instructions for preparation of IROFS boundaries packages and defines the interface with the National Enrichment Facility Project Team Managers/Staff responsible for NEF design.

2.0 SCOPE

This procedure is applicable to all IROFS.

3.0 DEFINITIONS

Item(s) Relied on for Safety (IROFS) means structures, systems, equipment, components, and activities of personnel that are relied on to prevent potential accidents at a facility that could exceed the performance requirements in 10CFR70.61 (b), (c) and (d) or to mitigate their potential consequences.

4.0 RESPONSIBILITY

Note: Designee will meet or exceed the qualifications of the position for which he/she is being designated (applies to all designees in this procedure).

4.1 Vice President, Licensing, Safety, and Nuclear Engineering or Designee – Implementation of DP-ISA-1.1, "IROFS Boundary Definition."

4.2 ISA Manager or Designee – Approves completed IROFS Boundary Definition Packages. Ensures that IROFS boundaries are defined and IROFS documentation is provided to the ISA Team member for incorporation into the IROFS Boundary Definition Packages.

4.3 ISA Team Member – Develops, and updates IROFS Boundary Definition Packages for NEF IROFS using IROFS documentation provided by National Enrichment Facility Project Team Managers/Staff. A second ISA Team Member peer reviews prepared IROFS Boundary Definition Packages.

4.4 National Enrichment Facility Project Team Managers/Staff – Provide applicable IROFS information to the ISA Team Member for the purpose of developing and updating the IROFS Boundary Definition Packages for each NEF IROFS.

5.0 PROCEDURE

5.1 Establishing IROFS Boundaries

5.1.1 Using the following guidance for establishing IROFS boundaries, the NEF IROFS shall be placed into separate bins based on type. The bins are as follows:

- Bin 1, Hardware Only IROFS (No operator actions are needed to satisfy the intended safety function of the IROFS)
- Bin 2, Administrative IROFS Requiring Use of Components (The intended IROFS function is fulfilled through operator intervention combined with a Component)
- Bin 3, Administrative IROFS for Parameter Monitoring (The intended IROFS function is fulfilled through operator monitoring of a specific parameter to ensure the parameter is within specified limit(s))
- Bin 4, Administrative Only IROFS (The intended IROFS function is fulfilled through implementation of procedural requirements not involving the use of components

5.1.2 The ISA Team Member shall develop the IROFS Boundary Package for each IROFS using the applicable Enclosure, currently available ISA information, and input from National Enrichment Facility Project Team Managers/Staff. During the process of developing the IROFS Boundary Package, the ISA Team Member shall interface with the National Enrichment Facility Project Team Managers/Staff to facilitate obtaining sufficient detail to meet the criteria specified in the applicable Enclosure for the existing level of design of the IROFS and items that may affect the function of IROFS. Upon completing the applicable Enclosure for the existing level of design of the IROFS and items that may affect the function of IROFS, the ISA Team Member shall sign and date the applicable cover sheet as preparer.

5.1.3 A second ISA Team Member shall peer review the prepared IROFS Boundary Package, comment and after concurrence with the preparer due to any required changes shall sign and date the IROFS Boundary Package cover sheet as reviewer.

5.1.4 The ISA Manager or Designee shall approve and shall sign and date the completed and reviewed IROFS Boundary Package to indicate approval.

5.1.5 Documents generated by this procedure shall be maintained in accordance with AP-QA-17.1, "Records."

5.1.6 A copy of the approved completed IROFS Boundary Package shall be forwarded to the National Enrichment Facility Project Team Managers/Staff responsible for design of the NEF, for use in developing a detail design of the IROFS boundary.

5.2 Updating IROFS Boundaries

5.2.1 As updated information is made available to the ISA Manager as required by the change control procedure (e.g., design change procedure, procedure change procedure, etc.), the IROFS Boundary Packages shall be updated to reflect the current IROFS design.

5.2.2 The ISA Team Member shall update the IROFS Boundary Package using information provided by National Enrichment Facility Project Team Managers/Staff. During the process of updating the IROFS Boundary Package, the ISA Team Member shall interface with the National Enrichment Facility Project Team Managers/Staff to facilitate obtaining sufficient detail to meet the criteria specified in the applicable Enclosure for the updated level of design of the IROFS and items that may affect the function of IROFS. Upon completion of updating the IROFS Boundary Package, the ISA Team Member shall sign and date the applicable cover sheet as preparer.

5.2.3 A second ISA Team Member shall peer review the prepared updated IROFS Boundary Package, comment and after concurrence with the preparer due to any required changes shall sign and date the IROFS Boundary Package cover sheet as reviewer.

5.2.4 The ISA Manager or Designee shall approve and shall sign and date the completed and reviewed IROFS Boundary Package to indicate approval.

5.2.5 Documents generated by this update shall be maintained in accordance with AP-QA-17.1.

5.2.6 A copy of the approved updated IROFS Boundary Package shall be forward to the National Enrichment Facility Project Team Managers/Staff responsible for design of the NEF, for further developing a detail design of the IROFS boundary.

6.0 REFERENCES

6.1 Title 10 to the Code of Federal Regulations, Part 70, "Domestic Licensing of Special Nuclear Material"

6.6 NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," Rev. 0

7.0 ENCLOSURES

7.1 Enclosure 1, "BIN 1 - Hardware only IROFS Boundary Definition"

7.2 Enclosure 2, "BIN 2 - Administrative IROFS Requiring Use of Components"

7.3 Enclosure 3, "BIN 3 - Administrative IROFS for Parameter Monitoring"

7.4 Enclosure 4, "BIN 4 - Administrative only IROFS"

ENCLOSURE 1

BIN 1 - Hardware only IROFS Boundary Definition

IROFS Number:

Preparer/Date:

Reviewer/Date:

Approver/Date:

BIN 1 - Hardware only IROFS Boundary Definition

BIN 1 - Hardware only IROFS (No operator actions are needed to satisfy the intended safety function of the IROFS)

The basis for establishing the boundaries for the hardware only IROFS is the Standard Technical Specification definition of OPERABLE-OPERABILITY used for nuclear power reactors. This definition is as follows.

> The system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specific safety function(s) are also capable of performing their related support function(s).

For the purpose of the IROFS boundary definition, the OPERABLE-OPERABILITY definition wording of "system, subsystem, division, component, or device," is replaced with the term "IROFS."

1. IROFS SAFETY FUNCTION IDENTIFICATION

   From the SAR description of the IROFS, list the IROFS number and identify its specified safety function, e.g., trip of power to a component, etc.

2. IROFS ATTRIBUTES

   Using the IROFS safety function description in item 1 as the starting point, consider and address each of the following attributes.

   SAFETY FUNCTION(S)

   a. Separation from other redundant or diverse IROFS

      If separation is required, then define train/channel orientation: (1) Channels are defined as an arrangement of components and modules as required to generate a single action signal when required by a plant condition. A channel loses it identity where single action signals are combined. (2) A train is defined as a given system or set of components that enables the establishment and

maintenance of physical, electrical, and functional independence from other redundant sets of components

If separation is not required, provide the basis/justification and identify the components of the channel or train of the IROFS.

b. Redundancy/Diversity/Independence

If redundancy, diversity, or independence is required, then identify the method and components that provide redundancy, diversity, or independence, as applicable.

If redundancy, diversity, and independence are not required, provide the basis/justification.

c. Electrical Separation (Isolation)

If electrical separation (isolation) is required, then identify the device used to provide isolation.

If electrical separation is not required, provide the basis/justification.

d. Fail-Safe or Highly Reliable

Each hardware only IROFS shall be designed to be fail-safe or highly reliable. An example of an IROFS that is considered to be fail-safe is one that is designed such, that on a power failure, the IROFS fails to the safe state and performs the IROFS safety function. An example of an IROFS that is considered to be highly reliable is one that is designed such that, if electrical power is required for the IROFS to function, two diverse sources of reliable electrical power are provided.

If the IROFS is designed to be fail-safe, then identify the method or device used to satisfy the fail-safe criteria.

BIN 1 - Hardware only IROFS Boundary Definition

If the IROFS is designed to be highly reliable, then identify the methods and components used to achieve high reliability. Also identify the conditions that cause the safe failure (e.g., loss of power, loss of air, etc.) and how the condition is reset.

e. Leak-tightness

If leak-tightness is required, then identify the boundaries of the structure, system or component that must be leak-tight and the amount of leakage that is allowed.

If leak-tightness is not required, provide the basis/justification.

f. Piping and (Pressure) Vessel Pressure Integrity

If piping or (pressure) vessel pressure integrity is required, then identify the boundaries of the piping or pressure vessel for which pressure integrity is required.

If piping or (pressure) vessel pressure integrity is not required, provide the basis/justification.

g. Software Validation and Verification (V&V)

If computer software is required by the IROFS to function, then a V&V shall be successfully completed for the associated software. Identify the associated software.

If computer software is not required by the IROFS to function, provide the basis/justification.

BIN 1 - Hardware only IROFS Boundary Definition

h. Setpoints

If automatic actuation of the IROFS must occur when a specific parameter value (setpoint) is reached, then NEF-specific Allowable Values and Trip Setpoints shall be provided using NRC approved setpoint calculation methodology, instrument loop design and as-built details. In addition, the frequency specified for calibration of the associated instrumentation shall be based on the magnitude of instrument drift assumed in the setpoint calculation. Identify the associated setpoint calculation.

If setpoint calculations are not required, provide the basis/justification.

i. IROFS Reset Capability

IROFS automatic actuation should continue to completion when actuated. The IROFS actuation should not be terminated or cleared on automatic reset of the actuation signal (i.e., the signal shall be sealed-in and manual action is required to reset the actuation). Identify the associated components.

If automatic reset of the IROFS actuation is provided, provide basis/justification.

j. Equipment Qualification

IROFS components should be qualified for the environment in which they are required to operate (assuming no HVAC is running in the area).

Identify any additional components necessary to achieve qualification.

BIN 1 - Hardware only IROFS Boundary Definition

k. Seismic Qualification

IROFS components whose safety function is required during or after a design basis seismic event shall be seismically qualified. Identify any additional components necessary to achieve qualification.

If seismic qualification is not required, provide basis/justification.

l. Normal Electrical Power Supply Voltage and Frequency Variations (including power surges)

For IROFS provided with normal electrical power, the design shall be such that normal electrical power supply voltage and frequency variations do not adversely impact the IROFS safety function. Identify design features provided to preclude adverse impact.

If design features are not required to preclude normal electrical power supply voltage and frequency variations from adversely impacting IROFS safety function, provide basis/justification.

m. Electromagnetic Interference (EMI) / Radio-Frequency Interference (RFI)

IROFS shall be designed such that EMI and RFI do not adversely impact IROFS safety function. Identify design features or controls provided to preclude adverse impact.

If design features or controls are not required to ensure EMI and RFI do not adversely impact IROFS safety function, provide basis/justification.

n. Fire Protection

IROFS components whose safety function is required during or after a fire shall be protected to ensure the IROFS safety function is

not adversely impacted. Identify design features provided to preclude adverse impact.

If fire protection is not required, provide basis/justification.

o. Lightning Protection

IROFS shall be designed such that lightning does not adversely impact IROFS safety function. Identify design features provided to preclude adverse impact.

If design features are not required to ensure lightning does not adversely impact IROFS safety function, provide basis/justification.

p. Internal Flooding

IROFS shall be designed such that internal flooding (from either internal or external sources) does not adversely impact IROFS safety function. Identify design features provided to preclude adverse impact.

If design features are not required to ensure internal flooding does not adversely impact IROFS safety function, provide basis/justification.

q. Load Drop/Impact

IROFS shall be designed such that load drops/impacts do not adversely impact IROFS safety function. Identify design features provided to preclude adverse impact.

If design features are not required to ensure load drops/impacts do not adversely impact IROFS safety function, provide basis/justification.

BIN 1 - Hardware only IROFS Boundary Definition

r. Non-QA Level 1 Component Interactions with QA Level 1 Components

IROFS shall be designed such that interactions between non-QA Level 1 components and QA Level 1 components do not adversely impact IROFS safety function. Identify design features provided to preclude adverse impact.

If design features are not required to ensure that interactions between non-QA Level 1 components and QA Level 1 components do not adversely impact IROFS safety function, provide basis/justification.

SUPPORT FUNCTION(S)

s. Instrumentation

If instrumentation is required to support the performance of the IROFS safety function, then identify the associated instrumentation.

If instrumentation is not required to support the IROFS safety function, provide the basis/justification.

t. Controls

If controls are required to support the performance of the IROFS safety function, then identify the associated controls.

If controls are not required to support the IROFS safety function, provide the basis/justification.

u. Heat Tracing

If heat tracing is required to support the performance of the IROFS safety function, then identify the associated heat tracing.

If heat tracing is not required to support the IROFS safety function, provide the basis/justification.

v. Instrument/Control Air System

If instrument/control air is required to support the performance of the IROFS safety function, then identify the associated instrument/control air system.

If instrument/control air is not required to support the IROFS safety function, provide the basis/justification.

w. Cooling System(s) (Air and/or Water)

If air cooling and/or water cooling are required to support the performance of the IROFS safety function, then identify the associated cooling system(s).

If air cooling and/or water cooling are not required to support the IROFS safety function, provide the basis/justification.

x. Lubrication System(s)

If lubrication system(s) are required to support the performance of the IROFS safety function, then identify the associated lubrication system(s).

If lubrication system(s) are not required to support the IROFS safety function, provide the basis/justification.

y. Seal System(s)

If seal system(s) are required to support the performance of the IROFS safety function, then identify the associated seal system(s).

If seal system(s) are not required to support the IROFS safety function, provide the basis/justification.

z. Electrical Power (Normal/Emergency) System(s)

If electrical power system(s) are required to support the performance of the IROFS safety function, then identify the associated electrical power system(s).

If electrical power system(s) are not required to support the IROFS safety function, provide the basis/justification.

3. MANAGEMENT MEASURES/RELIABILITY

IROFS management measures ensure compliance with the performance requirements assumed in the ISA documentation. The measures are applied to particular structures, systems, equipment, components, and activities of personnel and may be graded commensurate with the reduction of the risk attributable to that IROFS. The IROFS management measures shall ensure that these structures, systems, equipment, components, and activities of personnel within the identified IROFS boundary are designed, implemented, and maintained, as necessary, to comply with the performance requirements in the ISA documentation.

a. Configuration Management

Configuration management of IROFS, and any items that may affect the function of IROFS, shall be applied to all engineered or administrative controls identified within the scope of the IROFS boundary.

b. Maintenance

ENCLOSURE 1 (continued)

BIN 1 - Hardware only IROFS Boundary Definition

Maintenance of IROFS, and any item that may affect the function of IROFS, encompasses planned surveillance testing and maintenance/preventative maintenance, as well as unplanned corrective maintenance.

1. Planned Maintenance/Preventive Maintenance

   Planned Maintenance/Preventive Maintenance is important for ensuring that the IROFS is maintained reliable. Required planned maintenance/preventive maintenance, including type and frequency, shall be identified for each hardware only IROFS.

   If planned maintenance/preventive maintenance is not required to ensure the IROFS is maintained reliable, provide the basis/justification.

2. Functional Testing (Preoperational and Operational)/Inspection

   Functional testing/inspection ensure that the IROFS is capable of performing its required safety function. Required functional testing (both preoperational and operational)/inspection, including type and frequency, shall be identified for each hardware only IROFS.

   If functional testing/inspection are not required to ensure the IROFS is capable of performing its required safety function, provide the basis/justification.

3. Calibration

   Calibration ensures that IROFS instrumentation is capable of performing its required safety function and responds to the measured parameter within the necessary range and accuracy. Required calibration (both preoperational and operational), including method and frequency, shall be identified for each hardware only IROFS. The frequency

shall be based on the magnitude of instrument drift assumed in the associated setpoint calculation.

If calibration is not required to ensure the IROFS is capable of performing its required safety function and responding to the measured parameter within the necessary range and accuracy, provide the basis/justification.

4. Response Time Testing

Response time testing ensures that the IROFS is capable of performing its required safety function within the time period assumed in the safety analysis. Required response time testing (both preoperational and operational), including the method and frequency, shall be identified for each hardware only IROFS for which a response time is assumed in the safety analysis.

If response time testing is not required to ensure the IROFS is capable of performing its required safety function, provide the basis/justification.

c. Training and Qualifications

IROFS, and any items that may affect the function of IROFS, require that personnel involved at each level (from design through and including any assumed process implementation steps or actions) have and maintain the appropriate training and qualifications. Unique training and qualifications for each hardware only IROFS shall be identified.

d. Procedures

All activities involving IROFS, and any items that may affect the function of IROFS, are conducted in accordance with approved procedures. Each of the other IROFS management measures (e.g., configuration management, maintenance, and training) is implemented via approved procedures. The associated

implementation procedures (maintenance, training, functional testing, calibration, response time testing, etc.) for each IROFS shall also be identified.

e. Audits and Assessments

Audits shall be conducted to verify compliance with regulatory and procedural requirements and licensing commitments. Assessments shall be conducted to ensure that IROFS are reliable and are available to perform their intended safety functions as documented in the ISA.

f. Incident Investigations

Incident investigations shall be conducted within the corrective action process (CAP) for incidents associated with IROFS, and any items that may affect the function of IROFS.

g. Record Management

All records associated with IROFS, and any items that may affect the function of IROFS, shall be managed in a controlled and systematic manner in order to provide identifiable and retrievable documentation. Applicable design specifications, procurement documents, or other documents specify the QA records to be generated by, supplied to, or held, in accordance with approved procedures are included.

h. Other Quality Assurance Elements

Chapter 11 of the SAR identifies specifics of various other quality assurance elements. Any other quality assurance element associated with IROFS, or any items that may affect the function of IROFS, that is required to ensure the IROFS is available and reliable to perform the function when needed to comply with the performance requirements assumed in the ISA documentation, are listed in SAR Table 3.8-1, discussed in SAR Section 3.8.3. Using this information, incorporated any other quality assurance element associated with each IROFS in items 3.b, c, or d above.

BIN 1 - Hardware only IROFS Boundary Definition

4. DOCUMENTATION OF IROFS BOUNDARY

   a. Using the information developed in item 2 above, the following documents shall be annotated to indicate IROFS boundary for each hardware only IROFS.

      1. Piping and Instrument Diagrams (P&IDs)
      2. Process Flow Diagrams
      3. System Trip and Alarm List
      4. Electrical Schematics
      5. Control Logic Definition
      6. Calibration Data Sheets
      7 Setpoint Calculations

   b. Using the information developed in item 2 above, for each hardware only IROFS, a list shall be developed and maintained of all components (e.g., detectors, sensors, electronics, wiring, cabling, piping, valves, tanks, etc.) required to satisfy the IROFS safety function, including the required IROFS support functions. The components shall be identified by tag number, unique ID number, etc.

   c. For each hardware only IROFS, the responses to items 1, 2, and 3 above shall be documented.

ENCLOSURE 2

BIN 2 - Administrative IROFS Requiring Use of Components

IROFS Number:

Preparer/Date:

Reviewer/Date:

Approver/Date:

ENCLOSURE 2 (continued)

BIN 2 - Administrative IROFS Requiring Use of Components

BIN 2 - Administrative IROFS Requiring Use of Components (The intended IROFS function is fulfilled through operator intervention combined with a component)

Since the safety function of this type of IROFS is satisfied using both procedures and hardware, the basis for establishing the IROFS boundaries involves the methods used for hardware only IROFS (Bin 1, Enclosure 1) and administrative only IROFS (Bin 4, Enclosure 4).

1. IROFS SAFETY FUNCTION IDENTIFICATION

From the SAR description of the IROFS, list the IROFS number and identify its specified safety function, e.g., Administrative control and associated training of personnel on the movement/interaction of components containing fissile material and the criticality safety concerns associated with the movement, etc.

2. IROFS (HARDWARE PORTION) ATTRIBUTES

Using the IROFS safety function description in item 1 as the starting point, consider and address each of the following attributes.

SAFETY FUNCTION(S)

a. Separation from other redundant or diverse IROFS

If separation is required, then define train/channel orientation: (1) Channels are defined as an arrangement of components and modules as required to generate a single action signal when required by a plant condition. A channel loses it identity where single action signals are combined. (2) A train is defined as a given system or set of components that enables the establishment and maintenance of physical, electrical, and functional independence from other redundant sets of components

If separation is not required, provide the basis/justification and identify the components of the channel or train of the IROFS.

BIN 2 - Administrative IROFS Requiring Use of Components

b. Redundancy/Diversity/Independence

If redundancy, diversity, or independence is required, then identify the method and components that provide redundancy, diversity, or independence, as applicable.

If redundancy, diversity, and independence are not required, provide the basis/justification.

c. Electrical Separation (Isolation)

If electrical separation (isolation) is required, then identify the device used to provide isolation.

If electrical separation is not required, provide the basis/justification.

d. Fail-Safe or Highly Reliable

Each hardware only IROFS shall be designed to be fail-safe or highly reliable. An example of an IROFS that is considered to be fail-safe is one that is designed such, that on a power failure, the IROFS fails to the safe state and performs the IROFS safety function. An example of an IROFS that is considered to be highly reliable is one that is designed such that, if electrical power is required for the IROFS to function, two diverse sources of reliable electrical power are provided.

If the IROFS is designed to be fail-safe, then identify the method or device used to satisfy the fail-safe criteria. Also identify the conditions that cause the safe failure (e.g., loss of power, loss of air, etc.) and how the condition is reset.

If the IROFS is designed to be highly reliable, then identify the methods and components used to achieve high reliability.

BIN 2 - Administrative IROFS Requiring Use of Components

e. Leak-tightness

If leak-tightness is required, then identify the boundaries of the structure, system or component that must be leak-tight and the amount of leakage that is allowed.

If leak-tightness is not required, provide the basis/justification.

f. Piping and (Pressure) Vessel Pressure Integrity

If piping or (pressure) vessel pressure integrity is required, then identify the boundaries of the piping or pressure vessel for which pressure integrity is required.

If piping or (pressure) vessel pressure integrity is not required, provide the basis/justification.

g. Software Validation and Verification (V&V)

If computer software is required by the IROFS to function, then a V&V shall be successfully completed for the associated software. Identify the associated software.

If computer software is not required by the IROFS to function, provide the basis/justification.

h. Setpoints

If automatic actuation of the IROFS must occur when a specific parameter value (setpoint) is reached, then NEF-specific Allowable Values and Trip Setpoints shall be provided using NRC approved setpoint calculation methodology, instrument loop design and as-built details. In addition, the frequency specified for calibration of the associated instrumentation shall be based on the magnitude of instrument drift assumed in the setpoint calculation. Identify the associated setpoint calculation.

BIN 2 - Administrative IROFS Requiring Use of Components

If setpoint calculations are not required, provide the basis/justification.

i. IROFS Reset Capability

IROFS automatic actuation should continue to completion when actuated. The IROFS actuation should not be terminated or cleared on automatic reset of the actuation signal (i.e., the signal shall be sealed-in and manual action is required to reset the actuation). Identify the associated components.

If automatic reset of the IROFS actuation is provided, provide basis/justification.

j. Equipment Qualification

IROFS components should be qualified for the environment in which they are required to operate (assuming no HVAC is running in the area).

Identify any additional components necessary to achieve qualification.

k. Seismic Qualification

IROFS components whose safety function is required during or after a design basis seismic event shall be seismically qualified. Identify any additional components necessary to achieve qualification.

If seismic qualification is not required, provide basis/justification.

ENCLOSURE 2 (continued)

BIN 2 - Administrative IROFS Requiring Use of Components

l. Normal Electrical Power Supply Voltage and Frequency Variations (including power surges)

For IROFS provided with normal electrical power, the design shall be such that normal electrical power supply voltage and frequency variations do not adversely impact the IROFS safety function. Identify design features provided to preclude adverse impact.

If design features are not required to preclude normal electrical power supply voltage and frequency variations from adversely impacting IROFS safety function, provide basis/justification.

m. Electromagnetic Interference (EMI) / Radio-Frequency Interference (RFI)

IROFS shall be designed such that EMI and RFI do not adversely impact IROFS safety function. Identify design features or controls provided to preclude adverse impact.

If design features or controls are not required to ensure EMI and RFI do not adversely impact IROFS safety function, provide basis/justification.

p. Fire Protection

IROFS components whose safety function is required during or after a fire shall be protected to ensure the IROFS safety function is not adversely impacted. Identify design features provided to preclude adverse impact.

If fire protection is not required, provide basis/justification.

ENCLOSURE 2 (continued)

BIN 2 - Administrative IROFS Requiring Use of Components

q. Lightning Protection

IROFS shall be designed such that lightning does not adversely impact IROFS safety function. Identify design features provided to preclude adverse impact.

If design features are not required to ensure lightning does not adversely impact IROFS safety function, provide basis/justification.

p. Internal Flooding

IROFS shall be designed such that internal flooding (from either internal or external sources) does not adversely impact IROFS safety function. Identify design features provided to preclude adverse impact.

If design features are not required to ensure internal flooding does not adversely impact IROFS safety function, provide basis/justification.

q. Load Drop/Impact

IROFS shall be designed such that load drops/impacts do not adversely impact IROFS safety function. Identify design features provided to preclude adverse impact.

If design features are not required to ensure load drops/impacts do not adversely impact IROFS safety function, provide basis/justification.

r. Non-QA Level 1 Component Interactions with QA Level 1 Components

IROFS shall be designed such that interactions between non-QA Level 1 components and QA Level 1 components do not adversely

impact IROFS safety function. Identify design features provided to preclude adverse impact.

If design features are not required to ensure that interactions between non-QA Level 1 components and QA Level 1 components do not adversely impact IROFS safety function, provide basis/justification.

SUPPORT FUNCTION(S)

s. Instrumentation

If instrumentation is required to support the performance of the IROFS safety function, then identify the associated instrumentation.

If instrumentation is not required to support the IROFS safety function, provide the basis/justification.

t. Controls

If controls are required to support the performance of the IROFS safety function, then identify the associated controls.

If controls are not required to support the IROFS safety function, provide the basis/justification.

u. Heat Tracing

If heat tracing is required to support the performance of the IROFS safety function, then identify the associated heat tracing.

If heat tracing is not required to support the IROFS safety function, provide the basis/justification.

BIN 2 - Administrative IROFS Requiring Use of Components

v. Instrument/Control Air System

If instrument/control air is required to support the performance of the IROFS safety function, then identify the associated instrument/control air system.

If instrument/control air is not required to support the IROFS safety function, provide the basis/justification.

w. Cooling System(s) (Air and/or Water)

If air cooling and/or water cooling are required to support the performance of the IROFS safety function, then identify the associated cooling system(s).

If air cooling and/or water cooling are not required to support the IROFS safety function, provide the basis/justification.

x. Lubrication System(s)

If lubrication system(s) are required to support the performance of the IROFS safety function, then identify the associated lubrication system(s).

If lubrication system(s) are not required to support the IROFS safety function, provide the basis/justification.

y. Seal System(s)

If seal system(s) are required to support the performance of the IROFS safety function, then identify the associated seal system(s).

If seal system(s) are not required to support the IROFS safety function, provide the basis/justification.

z. Electrical Power (Normal/Emergency) System(s)

If electrical power system(s) are required to support the performance of the IROFS safety function, then identify the associated electrical power system(s).

If electrical power system(s) are not required to support the IROFS safety function, provide the basis/justification.

3. IROFS (PROCEDURAL PORTION) ATTRIBUTES

Using the IROFS safety function description in item 1 as the starting point, consider and address each of the following attributes

a. Procedures

Procedures shall be developed, implemented and maintained. The procedures shall be implemented by qualified and trained personnel. Identify the procedures used to fulfill the IROFS safety function. Identify the associated Training Lesson Plans

b. Enhanced IROFS

If the IROFS is classified as "enhanced," then identify the measure used to enhance it, e.g., independent verification of the operator action, etc.

Independent verification of each applicable IROFS safety function 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 has been performed correctly (e.g., verify a condition) or re-perform the task (e.g., collect and analyze a sample). 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 or process (as applicable) involved and its relationship to facility safety.

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."

c. Accessibility

Identify any accessibility limitations and mitigative features required due to plant conditions (toxic chemical, physical access, sufficient lighting, requirements for plant security, radiological exposure).

d. Personnel Staffing

Identify personnel requirements and limitations including the qualification and number of personnel required for performance of the IROFS implementation procedure.

e. Implementation Frequency

If an implementation frequency is required, then identify the frequency.

4. MANAGEMENT MEASURES/RELIABILITY

IROFS management measures ensure compliance with the performance requirements assumed in the ISA documentation. The measures are applied to particular structures, systems, equipment, components, and activities of personnel and may be graded commensurate with the

BIN 2 - Administrative IROFS Requiring Use of Components

reduction of the risk attributable to that IROFS. The IROFS management measures shall ensure that these structures, systems, equipment, components, and activities of personnel within the identified IROFS boundary are designed, implemented, and maintained, as necessary, to comply with the performance requirements in the ISA documentation.

a. Configuration Management

Configuration management of IROFS, and any items that may affect the function of IROFS, shall be applied to all engineered or administrative controls identified within the scope of the IROFS boundary.

b. Maintenance

Maintenance of IROFS, and any item that may affect the function of IROFS, encompasses planned surveillance testing and maintenance/preventative maintenance, as well as unplanned corrective maintenance.

1. Planned Maintenance/Preventive Maintenance

Planned Maintenance/Preventive Maintenance is important for ensuring that the IROFS is maintained reliable. Required planned maintenance/preventive maintenance, including type and frequency, shall be identified for each IROFS.

If planned maintenance/preventive maintenance is not required to ensure the IROFS is maintained reliable, provide the basis/justification.

2. Functional Testing (Preoperational and Operational)/Inspection

Functional testing/inspection ensure that the IROFS is capable of performing its required safety function. Required functional testing (both preoperational and operational)/inspection, including type and frequency, shall be identified for each IROFS.

BIN 2 - Administrative IROFS Requiring Use of Components

If functional testing/inspection are not required to ensure the IROFS is capable of performing its required safety function, provide the basis/justification.

3. Calibration

Calibration ensures that IROFS instrumentation is capable of performing its required safety function and responds to the measured parameter within the necessary range and accuracy. Required calibration (both preoperational and operational), including method and frequency, shall be identified for each IROFS. The frequency shall be based on the magnitude of instrument drift assumed in the associated setpoint calculation.

If calibration is not required to ensure the IROFS is capable of performing its required safety function and responding to the measured parameter within the necessary range and accuracy, provide the basis/justification.

4. Response Time Testing

Response time testing ensures that the IROFS is capable of performing its required safety function within the time period assumed in the safety analysis. Required response time testing (both preoperational and operational), including the method and frequency, shall be identified for each IROFS for which a response time is assumed in the safety analysis.

If response time testing is not required to ensure the IROFS is capable of performing its required safety function, provide the basis/justification.

c. Training and Qualifications

IROFS, and any items that may affect the function of IROFS, require that personnel involved at each level (from design through

and including any assumed process implementation steps or actions) have and maintain the appropriate training and qualifications. Unique training and qualifications for each IROFS shall be identified.

d. Procedures

All activities involving IROFS, and any items that may affect the function of IROFS, are conducted in accordance with approved procedures. Each of the other IROFS management measures (e.g., configuration management, maintenance, and training) is implemented via approved procedures. The associated implementation procedures (maintenance, training, functional testing, calibration, response time testing, etc.) for each IROFS shall also be identified.

e. Audits and Assessments

Audits shall be conducted to verify compliance with regulatory and procedural requirements and licensing commitments. Assessments shall be conducted to ensure that IROFS are reliable and are available to perform their intended safety functions as documented in the ISA.

f. Incident Investigations

Incident investigations shall be conducted within the corrective action process (CAP) for incidents associated with IROFS, and any items that may affect the function of IROFS.

g. Record Management

All records associated with IROFS, and any items that may affect the function of IROFS, shall be managed in a controlled and systematic manner in order to provide identifiable and retrievable documentation. Applicable design specifications, procurement documents, or other documents specify the QA records to be generated by, supplied to, or held, in accordance with approved procedures are included.

h. Other Quality Assurance Elements

Chapter 11 of the SAR identifies specifics of various other quality assurance elements. Any other quality assurance element associated with IROFS, or any items that may affect the function of IROFS, that is required to ensure the IROFS is available and reliable to perform the function when needed to comply with the performance requirements assumed in the ISA documentation, are listed in SAR Table 3.8-1, discussed in SAR Section 3.8.3. Using this information, incorporated any other quality assurance element associated with each IROFS in items 3.b, c, or d above.

5. DOCUMENTATION OF IROFS BOUNDARY

a. Using the information developed in item 2 above, the following documents shall be annotated to indicate IROFS boundary for each IROFS (hardware portion).

   1. Piping and Instrument Diagrams (P&IDs)

   2. Process Flow Diagrams

   3. System Trip and Alarm List

   4. Electrical Schematics

   5. Control Logic Definition

   6. Calibration Data Sheets

   7 Setpoint Calculations

b. Using the information developed in item 2 above, for each IROFS (hardware portion), a list shall be developed and maintained of all components (e.g., detectors, sensors, electronics, wiring, cabling, piping, valves, tanks, etc.) required to satisfy the IROFS safety function, including the required IROFS support functions. The components shall be identified by tag number, unique ID number, etc.

c. Using the responses to items 1, 3 and 4 above, develop the following documents which shall be annotated to indicate the IROFS (administrative portion) boundary (i.e., identify which documents or portions of documents, are used to satisfy requirements of the

BIN 2 - Administrative IROFS Requiring Use of Components

associated administrative IROFS requiring use of components) for each administrative IROFS requiring use of components.

1. Procedures

2. Lesson Plans

d. Using the information developed in item 3 above, for each administrative IROFS, a list shall be developed and maintained of all procedures and lesson plans required to satisfy the IROFS safety function, including the required IROFS support functions. The procedures and lesson plans shall be identified by a unique ID number.

e. The response to items 1, 2, 3 and 4 above shall be documented for each Administrative IROFS requiring use of components.

ENCLOSURE 3

BIN 3 - Administrative IROFS for Parameter Monitoring

IROFS Number:

Preparer/Date:

Reviewer/Date:

Approver/Date:

BIN 3 - Administrative IROFS for Parameter Monitoring

BIN 3 - Administrative IROFS for Parameter Monitoring (The intended IROFS function is fulfilled through operator monitoring of a specific parameter to ensure the parameter value is within specified limit(s))

Since the safety function of this type of IROFS is satisfied using both procedures and monitoring devices, the basis for establishing the IROFS boundaries is derived from the requirements of the LES QAPD requirements for Measuring and Test Equipment (M&TE) and the method used for administrative only IROFS (Bin 4, Enclosure 4).

1. IROFS SAFETY FUNCTION IDENTIFICATION

   From the SAR description of the IROFS, list the IROFS number and identify its specified safety function, e.g., procedures and training to administratively control cylinder over fill by verifying that cylinder weight is within specified limits.

2. IROFS ATTRIBUTES

   Using the IROFS safety function description in item 1 as the starting point, consider and address each of the following attributes.

   a. Parameter Monitoring Instrumentation/Device(s)

      The instrumentation or device(s) used to measure specific parameter(s) identified in the Administrative IROFS for Parameter Monitoring shall meet the QA Level 1 M&TE requirements. As such, these instrument(s) or device(s) shall meet the requirements specified in the LES QAPD for M&TE. The type, range, accuracy, and tolerance associated with the M&TE shall be sufficient for determining that the parameter(s) required to be monitored by the IROFS are within specified limits. In accordance with the LES QAPD, the M&TE shall be calibrated and adjusted, as needed, at specified periods to maintain accuracy within necessary limits. In addition, M&TE shall be properly handled and stored to maintain accuracy.

      Identify the M&TE to be used for monitoring the IROFS parameter(s).

BIN 3 - Administrative IROFS for Parameter Monitoring

b. Procedures

Procedures shall be developed, implemented and maintained. The procedures shall be implemented by qualified and trained personnel. The associated procedures shall identify the M&TE to be used during the performance of the monitoring. The associated procedures shall include acceptance criteria, e.g., acceptable ranges or values of the monitored parameter(s).

Identity the procedures used to fulfill the IROFS safety function. Identify the associated Training Lesson Plans.

c. Enhanced IROFS

If the IROFS is classified as "enhanced," then identify the measure used to enhance it, e.g., independent verification of the operator action, etc.

If the IROFS is classified as "enhanced," then identify the measure used to enhance it, e.g., independent verification of the operator action, etc.

Independent verification of each applicable IROFS safety function 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 has been performed correctly (e.g., verify a condition) or re-perform the task (e.g., collect and analyze a sample). 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 or process (as applicable) involved and its relationship to facility safety.

The requirements for independent verification are consistent with the applicable guidance provided in ANSI/ANS-3.2-1994,

BIN 3 - Administrative IROFS for Parameter Monitoring

"Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants."

d. Accessibility

Identify any accessibility limitations and mitigative features required due to plant conditions (toxic chemical, physical access, sufficient lighting, requirements for plant security, radiological exposure).

e. Personnel Staffing

Identify personnel requirements and limitations including the qualification and number of personnel required for performance of the IROFS implementation procedure.

f. Implementation Frequency

If an implementation frequency is required, then identify the frequency.

3. MANAGEMENT MEASURES

IROFS management measures ensure compliance with the performance requirements assumed in the ISA documentation. The measures are applied to particular structures, systems, equipment, components, and activities of personnel and may be graded commensurate with the reduction of the risk attributable to that IROFS. The IROFS management measures shall ensure that these structures, systems, equipment, components, and activities of personnel within the identified IROFS boundary are designed, implemented, and maintained, as necessary, to comply with the performance requirements in the ISA documentation.

a. Configuration Management

Configuration management of IROFS, and any items that may affect the function of IROFS, shall be applied to all engineered or

administrative controls identified within the scope of the IROFS boundary.

b. Maintenance

Maintenance of IROFS, and any item that may affect the function of IROFS, encompasses planned surveillance testing and maintenance/preventative maintenance, as well as unplanned corrective maintenance.

1. Planned Maintenance/Preventive Maintenance

Planned Maintenance/Preventive Maintenance is important for ensuring that the IROFS is maintained reliable. Required planned maintenance/preventive maintenance, including type and frequency, shall be identified for each IROFS.

If planned maintenance/preventive maintenance is not required to ensure the IROFS is maintained reliable, provide the basis/justification.

2. Functional Testing (Preoperational and Operational)/Inspection

Functional testing/inspection ensure that the IROFS is capable of performing its required safety function. Required functional testing (both preoperational and operational)/inspection, including type and frequency, shall be identified for each IROFS.

If functional testing/inspection are not required to ensure the IROFS is capable of performing its required safety function, provide the basis/justification.

3. Calibration

Calibration ensures that IROFS instrumentation is capable of performing its required safety function and responds to the

measured parameter within the necessary range and accuracy. Required calibration (both preoperational and operational), including method and frequency, shall be identified for each IROFS. The frequency shall be based on the magnitude of instrument drift assumed in the associated setpoint calculation.

If calibration is not required to ensure the IROFS is capable of performing its required safety function and responding to the measured parameter within the necessary range and accuracy, provide the basis/justification.

4. Response Time Testing

Response time testing ensures that the IROFS is capable of performing its required safety function within the time period assumed in the safety analysis. Required response time testing (both preoperational and operational), including the method and frequency, shall be identified for each IROFS for which a response time is assumed in the safety analysis.

If response time testing is not required to ensure the IROFS is capable of performing its required safety function, provide the basis/justification.

c. Training and Qualifications

IROFS, and any items that may affect the function of IROFS, require that personnel involved at each level (from design through and including any assumed process implementation steps or actions) have and maintain the appropriate training and qualifications. Unique training and qualifications for each IROFS shall be identified.

d. Procedures

All activities involving IROFS, and any items that may affect the function of IROFS, are conducted in accordance with approved

procedures. Each of the other IROFS management measures (e.g., configuration management, maintenance, and training) is implemented via approved procedures. The associated implementation procedures (maintenance, training, functional testing, calibration, response time testing, etc.) for each IROFS shall also be identified.

e. Audits and Assessments

Audits shall be conducted to verify compliance with regulatory and procedural requirements and licensing commitments. Assessments shall be conducted to ensure that IROFS are reliable and are available to perform their intended safety functions as documented in the ISA.

f. Incident Investigations

Incident investigations shall be conducted within the corrective action process (CAP) for incidents associated with IROFS, and any items that may affect the function of IROFS.

g. Record Management

All records associated with IROFS, and any items that may affect the function of IROFS, shall be managed in a controlled and systematic manner in order to provide identifiable and retrievable documentation. Applicable design specifications, procurement documents, or other documents specify the QA records to be generated by, supplied to, or held, in accordance with approved procedures are included.

h. Other Quality Assurance Elements

Chapter 11 of the SAR identifies specifics of various other quality assurance elements. Any other quality assurance element associated with IROFS, or any items that may affect the function of IROFS, that is required to ensure the IROFS is available and reliable to perform the function when needed to comply with the performance requirements assumed in the ISA documentation, are listed in SAR Table 3.8-1, discussed in SAR Section 3.8.3. Using

BIN 3 - Administrative IROFS for Parameter Monitoring

this information, incorporated any other quality assurance element associated with each IROFS in items 3.b, c, or d above.

Procedures are controlled in accordance with the applicable provisions of the LES QAPD.

4. DOCUMENTATION OF IROFS BOUNDARY

   a. Using the responses to items 1 2 and 3 above, develop the following documents which shall be annotated to indicate the IROFS boundary (i.e., identify which documents or portions of documents, are used to satisfy requirements of the associated IROFS) for each Administrative IROFS for Parameter Monitoring.

      1. Procedures

      2. Lesson Plans

   b. Using the information developed in item 2 and 3 above, for each Administrative IROFS for Parameter Monitoring, a list shall be developed and maintained of all procedures and lesson plans required to satisfy the IROFS safety function, including the M&TE calibration procedures. The procedures and lesson plans shall be identified by a unique ID number.

   c. Using the response to item 2 above, a list shall be developed and maintained for all instrument(s) or device(s) that satisfy LES QAPD requirements for M&TE for each Administrative IROFS for Parameter Monitoring. The instrumentation or device(s) shall be identified by the tag number, unique ID number, etc.

   d. The response to items 1, 2 and 3 above shall be documented for each Administrative IROFS for Parameter Monitoring.

ENCLOSURE 4

BIN 4 - Administrative only IROFS

IROFS Number:

Preparer/Date:

Reviewer/Date:

Approver/Date:

BIN 4 - Administrative only IROFS

BIN 4 - Administrative only IROFS (The intended IROFS function is fulfilled through implementation of procedural requirements, not involving the use of components)

The basis for establishing the boundaries for the administrative only IROFS is derived from NEF SAR Chapter 11 requirements for procedures. Specifically, all activities involving IROFS are required to be 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. These procedures are also required to be sufficiently detailed such that qualified individuals can perform the required functions without direct supervision.

1. IROFS SAFETY FUNCTION IDENTIFICATION

   From the SAR description of the IROFS, list the IROFS number and identify its specified safety function, e.g., control, through the use of procedures and training, of the type of oil used in the process vacuum pumps.

2. IROFS ATTRIBUTES

   Using the IROFS safety function description in item 1 as the starting point, consider and address each of the following attributes.

   a. Procedures

      Procedures shall be developed, implemented and maintained. The procedures shall be implemented by qualified and trained personnel. Identify the procedures used to fulfill the IROFS safety function. Identify the associated Training Lesson Plans

   b. Enhanced IROFS

      If the IROFS is classified as "enhanced," then identify the measure used to enhance it, e.g., independent verification of the operator action, etc.

BIN 4 - Administrative only IROFS

If the IROFS is classified as "enhanced," then identify the measure used to enhance it, e.g., independent verification of the operator action, etc.

Independent verification of each applicable IROFS safety function 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 has been performed correctly (e.g., verify a condition) or re-perform the task (e.g., collect and analyze a sample). 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 or process (as applicable) involved and its relationship to facility safety.

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."

c. Accessibility

Identify any accessibility limitations and mitigative features required due to plant conditions (toxic chemical, physical access, sufficient lighting, requirements for plant security, radiological exposure).

d. Personnel Staffing

Identify personnel requirements and limitations including the qualification and number of personnel required for performance of the IROFS implementation procedure.

e. Implementation Frequency

If an implementation frequency is required, then identify the frequency.

3. MANAGEMENT MEASURES

IROFS management measures ensure compliance with the performance requirements assumed in the ISA documentation. The measures are applied to particular structures, systems, equipment, components, and activities of personnel and may be graded commensurate with the reduction of the risk attributable to that IROFS. The IROFS management measures shall ensure that these structures, systems, equipment, components, and activities of personnel within the identified IROFS boundary are designed, implemented, and maintained, as necessary, to comply with the performance requirements in the ISA documentation.

a. Configuration Management

Configuration management of IROFS, and any items that may affect the function of IROFS, shall be applied to all engineered or administrative controls identified within the scope of the IROFS boundary.

b. Maintenance

Planned Maintenance/Preventive Maintenance is important for ensuring that the IROFS is maintained reliable. Required planned maintenance/preventive maintenance, including type and frequency, shall be identified for each IROFS.

If planned maintenance/preventive maintenance is not required to ensure the IROFS is maintained reliable, provide the basis/justification.

c. Training and Qualifications

IROFS, and any items that may affect the function of IROFS, require that personnel involved at each level (from design through

and including any assumed process implementation steps or actions) have and maintain the appropriate training and qualifications. Unique training and qualifications for each IROFS shall be identified.

d. Procedures

All activities involving IROFS, and any items that may affect the function of IROFS, are conducted in accordance with approved procedures. Each of the other IROFS management measures (e.g., configuration management, maintenance, and training) is implemented via approved procedures. The associated implementation procedures (maintenance, training, etc.) for each IROFS shall also be identified.

e. Audits and Assessments

Audits shall be conducted to verify compliance with regulatory and procedural requirements and licensing commitments. Assessments shall be conducted to ensure that IROFS are reliable and are available to perform their intended safety functions as documented in the ISA.

f. Incident Investigations

Incident investigations shall be conducted within the corrective action process (CAP) for incidents associated with IROFS, and any items that may affect the function of IROFS.

g. Record Management

All records associated with IROFS, and any items that may affect the function of IROFS, shall be managed in a controlled and systematic manner in order to provide identifiable and retrievable documentation. Applicable design specifications, procurement documents, or other documents specify the QA records to be generated by, supplied to, or held, in accordance with approved procedures are included.

ENCLOSURE 4 (continued)

BIN 4 - Administrative only IROFS

h. Other Quality Assurance Elements

Chapter 11 of the SAR identifies specifics of various other quality assurance elements. Any other quality assurance element associated with IROFS, or any items that may affect the function of IROFS, that is required to ensure the IROFS is available and reliable to perform the function when needed to comply with the performance requirements assumed in the ISA documentation, are listed in SAR Table 3.8-1, discussed in SAR Section 3.8.3. Using this information, incorporated any other quality assurance element associated with each IROFS in items 3.b, c, or d above.

4. DOCUMENTATION OF IROFS BOUNDARY

a. Using the responses to items 1, 2 and 3 above, develop the following documents which shall be annotated to indicate the IROFS boundary (i.e., identify which documents or portions of documents, are used to satisfy requirements of the associated administrative only IROFS) for each Administrative only IROFS.

   1. Procedures

   2. Lesson Plans

b. The response to items 1 and 2 above shall be documented for each Administrative only IROFS.

c. Using the information developed in item 2 and 3 above, for each administrative only IROFS, a list shall be developed and maintained of all procedures and lesson plans required to satisfy the IROFS safety function. The procedures and lesson plans shall be identified by a unique ID number