DUKE COGEMA
STONE & WEBSTER

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

31 August 2001
DCS-NRC-000060

Subject: Docket Number 070-03098
Duke Cogema Stone & Webster
Mixed Oxide Fuel Fabrication Facility
Response to Request for Additional Information

References: (1) J. G. Giitter (NRC) letter to P. Hastings (DCS), dated 21 June 2001, "Mixed Oxide Fuel Fabrication Facility Construction Authorization - Request For Additional Information"

(2) R. H. Ihde (DCS) letter to W. F. Kane (NRC), DCS-NRC-000038, dated 28 February 2001, "Construction Authorization Request"

This letter provides the Duke Cogema Stone & Webster (DCS), LLC, response to the NRC's request for additional information (RAI, Reference 1) concerning the Mixed Oxide (MOX) Fuel Fabrication Facility Construction Authorization Request (CAR, Reference 2).

Some of this information is the privileged (proprietary) information of DCS. DCS requests that the proprietary information be withheld from public disclosure pursuant to 10 CFR 2.790(b).

Enclosure 1 provides the affidavit attesting to the privileged information nature of the proprietary portion of this submittal. Enclosure 2 is a redacted, non-proprietary version of the response to the RAI. Enclosure 2 may be disclosed to the public since the proprietary information has been removed from the redacted copy. Enclosure 3 is the proprietary portion of the response to the RAI. Enclosure 3 should be withheld from public disclosure; the proprietary information is denoted by brackets.

For each question in the RAI, a response and action is provided; the *actions* refer to changes to the CAR necessitated by the response to the question. DCS is not revising the CAR at this time, but expects to revise it later this year to incorporate these changes.

Notable among the questions in the RAI were those associated with the design basis of the MFFF. As NRC's authorization of construction of the MFFF is predicated in part on the acceptability of the safety assessment of the design bases of principal structures, systems, and components (SSCs), DCS is particularly interested in resolving any questions on the part of the NRC Staff with regard to this issue. Consequently, DCS has included in these responses a substantial amount of design information that extends beyond what is considered "design basis" as defined in 10 CFR 50.2.

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DCS trusts that this information will facilitate review of the CAR, and is interested in meeting with the Staff to resolve any outstanding questions and make additional information available for review as necessary.

Please note that DCS has requested clarification from the NRC regarding RAI Question 40.

If I can provide any additional information, please feel free to contact me at (704) 373-7820.

Sincerely,

Peter S. Hastings, P.E.
Licensing Manager

Enclosures:
(1) Affidavit Pursuant To 10 CFR 2.790(b)(1)
(2) Non-Proprietary Response to Request for Additional Information
(3) Proprietary Portion of Response to Request for Additional Information
(4) Non-Proprietary Reference Material Accompanying Response to Request for Additional Information
(5) Non-Proprietary Data Disk Accompanying Response to RAI Question 46

Distribution:

(with all enclosures):

Andrew Persinko, USNRC/HQ (including 5 copies of enclosures 3, 4, and 5)
Donald J. Silverman, Esq., DCS
PRA/EDMS: Corresp\Outgoing\NRC\Licensing\DCS-NRC-000060

(with letter and enclosures 1 and 2):

David Alberstein, USDOE/HQ
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(without enclosures):

Marc Arslan, DCS
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Eric J. Leeds, USNRC/HQ
John E. Matheson, DCS
J. David Nulton, USDOE/MD
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Luis A. Reyes, USNRC/RII
Michael F. Weber, USNRC/HQ

Enclosure (1)

Affidavit Pursuant To 10 CFR 2.790(b)(1)

AFFIDAVIT PURSUANT TO 10 CFR 2.790(b)(1)

1. I am President and Chief Executive Officer of Duke Cogema Stone and Webster, LLC; ("DCS") and as such have the responsibility for reviewing information sought to be withheld from public disclosure in connection with design and licensing of the Mixed Oxide Fuel Fabrication Facility (the "MFFF"); and am authorized on the part of DCS to apply for this withholding.

2. I am making this affidavit in conformance with the provisions of 10CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with DCS's application for withholding, which accompanies this affidavit.

3. I have knowledge of the criteria used by DCS in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by DCS, its partners, and/or affiliates, and has been held in confidence by the same.

   (ii) The information is of a type that would customarily be held in confidence by DCS, its partners, and/or affiliates. The information consists of design details and processing methods and mechanisms relative to a method of processing that provides a competitive advantage to DCS, its partners, and/or affiliates.

   (iii) The information was transmitted to the NRC in confidence and under the provisions of 10CFR 2.790, it is to be received in confidence by the NRC.

   (iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

   (v) The proprietary information sought to be withheld in this submittal is that which is marked in the proprietary version of the enclosure to the accompanying DCS letter, and omitted from the non-proprietary version. This information describes DCS' design for the MFFF. This information enables DCS, its partners, and/or affiliates to support license amendment applications for the MFFF.

   (vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to DCS, its partners, and/or affiliates.

      (a) It allows DCS to reduce vendor and consultant expenses associated with supporting the licensing of fuel fabrication plants.

(b) DCS may sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the licensing of fuel fabrication plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by DCS, its partners, and/or affiliates.

5. Public disclosure of this information is likely to cause harm to DCS, its partners, and/or affiliates because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring commensurate expense or allowing DCS, its partners, and/or affiliates to recoup a portion of its expenditures or benefit from the sale of the information.

R. H. Ihde, being duly sworn, states that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth within are true and correct to the best of his knowledge.

Robert H. Ihde

R. H. Ihde, President and CEO

Subscribed and sworn to on this 31 day of August, 2001

Teresa A. Blackmon
Notary Public

My Commission Expires:

My Commission Expires December 28, 2005

SEAL

Enclosure (2)

Non-Proprietary Response to Request for Additional Information

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION FOR THE DUKE COGEMA STONE & WEBSTER (DCS) MIXED OXIDE (MOX) FUEL FABRICATION FACILITY (MFFF) CONSTRUCTION AUTHORIZATION REQUEST (CAR)

CHAPTER 1, GENERAL INFORMATION

1. Section 1.1.2.1, pp. 1.1-1 thru 1.1-2

Revise the description of Savannah River Site workers who are outside the mixed oxide fuel fabrication facility (MFFF) restricted area but within the controlled area boundary that is provided in the second paragraph of Section 1.1.2.1 to state that these workers are deemed to be "members of the public."

The description in Section 1.1.2.1 should reflect the NRC's policy that individuals who are not closely and frequently connected to the licensed activity should be considered members of the public. This policy is described in an NRC Staff Requirements Memorandum on a similar and related issue at the U.S. Department of Energy's (DOE's) Hanford site (SECY-98-038, "Hanford Tank Waste Remediation System Privatization Co-located Worker Standards.")

Response:

DCS has reviewed SECY-98-038[1] and other related NRC and DOE documents pertaining to whether DOE co-located workers (CLWs) should be deemed "members of the public." DCS has concluded that referring to Savannah River Site (SRS) workers who are outside of the MFFF restricted area but inside the controlled area boundary as "members of the public" would be incorrect and inconsistent with 10 CFR Part 70, and that the policy set forth in SECY-98-038 has been superseded by the amendments to Part 70. In addition, applying this label to the SRS workers is not necessary to ensure that dose limits outside of the MFFF restricted area are within the Part 20, non-occupational dose limits during normal operations. DCS' current design and licensing approach for the MFFF will ensure the requirements of 10 CFR Part 20 and Part 70 are fully met. This includes maintaining normal-operation doses at the MFFF restricted area boundary to below 100 mrem/yr. The discussion below provides the basis for DCS' conclusion.

Purpose of SECY-98-038

SECY-98-038 was written to inform the Commission of a new issue presented by a difference between the DOE and the NRC with regard to the specification of accident design values for co-located workers. The SECY document applied to DOE's Hanford site as part of the Tank Waste Remediation System Privatization (TWRS-P) effort. The SECY document approved by the Commission clearly recognized that for normal operations, DOE's requirements would ensure occupational and non-occupational doses would be within Part 20 limits. The SECY document

[1] *Hanford Tank Remediation System Privatization Co-Located Worker Standards*, SECY-98-038, U.S. Nuclear Regulatory Commission, Washington, D.C., March 4, 1998.

concluded "NRC and DOE differences with respect to CLWs are significant only with respect to off-normal or accident conditions (i.e., referred to in TWRS-P contracts as anticipated events, unlikely events, and extremely unlikely events), and in how the related accident analyses can affect facility design choices." Although in the SECY document it was stated that the NRC staff interpreted that "committing to meet Part 835 implies considering CLWs as members of the general public during normal operations of the facility," this staff interpretation was clearly not the focus or purpose of SECY-98-038. This is further illustrated in the concluding paragraph of SECY-98-038, which states that CLWs would be treated as members of the public because the current approach at Hanford was not consistent with NRC regulations and the NRC regulatory approach "for protection of members of the public during accidents."

SECY-98-038 (accident standards for co-located workers) is not applicable to the MFFF. 10 CFR 70.61(f) specifies the requirements that pertain to individuals who are not MFFF workers and will be performing ongoing activities in the controlled area. During development of the revised Part 70 rule, SECY-98-038 and SECY-98-185[2], were consistent with respect to the co-located worker. However, SECY-99-147[3] provided a subsequent draft of the proposed Part 70 rule in which treatment of site workers not related to licensed operations was substantially changed. The final rule for 10 CFR Part 70 specifically addressed and clarified the co-located worker issue.

> "The licensee can set the controlled area at any location around its facility as long as it maintains control of that area as specified in Part 20 and retains the authority to exclude or remove personnel and property from the area. If the controlled area included the nearby Department of Energy (DOE) facilities, then NRC would consider the personnel working at those facilities to be "workers" for the purposes of the performance requirements of Sec. 70.61, provided the conditions of Sec. 70.61(f)(2) are met[4]."

<u>NRC Task Force Position on CLWs</u>

An NRC task force that focused on external regulation of DOE nuclear facilities identified the topic of DOE co-located workers as an additional issue that would have to be addressed.[5] The NRC task force summarized the issue as follows.

> "In performing an accident analysis for an individual facility at a DOE site, DOE has traditionally treated all workers at that facility and workers at other facilities within the site (co-located workers) as general employees who may receive an occupational exposure. NRC would treat all workers outside the facility as members of the public, unless they have been given appropriate radiation safety

[2] *Proposed Rulemaking - Revised Requirements for the Domestic Licensing of Special Nuclear Material,* SECY-98-185, U.S. Nuclear Regulatory Commission, Washington, D.C., July 30, 1998.
[3] *Proposed Rulemaking - Revised Requirements for the Domestic Licensing of Special Nuclear Material,* SECY-99-147, U.S. Nuclear Regulatory Commission, Washington, D.C., June 2, 1999.
[4] 65 FR 56211, September 18, 2000.
[5] *External Regulation of Department of Energy Nuclear Facilities: A Pilot Program,* NUREG-1708, U.S. Nuclear Regulatory Commission, Washington, D.C., July 1999.

training (10 CFR Part 19, which is similar to the training required by DOE regulation 10 CFR Part 835 for general employees). DOE has expressed a concern that if NRC treated co-located workers as subject to the public dose limit, it might need to substantially revise many of its accident analyses.

"However, if DOE complies with Part 835 by giving radiation training to its general employees, including its co-located workers, and making provisions to evacuate visiting members of the public, there should be no difference between NRC and DOE positions."

The NRC staff stated that the concern was focused on treating CLWs as subject to the public dose limits when performing safety analyses. The task force also came to a similar conclusion as SECY-98-038, that if DOE complies with Part 835, there should be no difference between NRC and DOE positions.

DNFSB Position on CLWs

The Defense Nuclear Facilities Safety Board (DNFSB) has also addressed the issue of co-located workers at DOE sites[6]. The DNFSB stated that the issue is the protection of CLWs from the possible harmful effects of nuclear radiation at DOE facilities. The report recognized that despite the ongoing discussions on the CLW issue, the DOE system that has been in effect has been successful. The report states: "There is no pattern of personal injury from nuclear radiation among co-located workers at DOE's defense nuclear facilities and sites, nor is there such a pattern among other groups of individuals at or near the sites." The report describes the integrated safety management approach that ensures the protection of all individuals, including the facility workers, the co-located workers, and the public. Although the report focuses on DOE's integrated, protective system, it is also directly applicable to the safety process prescribed by NRC regulations.

Additional Considerations

As NRC stated in publishing the final rule for Part 70, "[t]he licensee can set the controlled area at any location around its facility as long as it maintains control of that area as specified in Part 20 and retains the authority to exclude or remove personnel and property from the area. If the controlled area included the nearby Department of Energy (DOE) facilities, then NRC would consider the personnel working at those facilities to be "workers" for the purposes of the performance requirements of Sec. 70.61, provided the conditions of Sec. 70.61(f)(2) are met." If the NRC staff makes an interpretation under Part 20, that during normal operations the personnel working at nearby DOE facilities are "members of the public", there is an inconsistency in terminology and application of the two regulations (Part 70 and Part 20).

Treating SRS workers as "members of the public" during normal operations results in a conflict between federal agencies. As discussed earlier, the DOE does not consider co-located workers or

[6] *Protection of Collocated Workers at the Department of Energy's Defense Nuclear Facilities and Sites*, DNFSB/TECH-20, Defense Nuclear Facilities Safety Board, Washington, D.C. February 1999.

general site workers to be "members of the public." This situation would result in inconsistent interpretation of analogous (virtually identical) language in the two regulations, Part 20 and Part 835. This inconsistency between NRC and DOE would also result in different reporting requirements with regard to public exposure. The Savannah River Site currently reports public exposure at the site boundary (MFFF controlled area boundary). An interpretation by the NRC staff that "members of the public" (i.e., site workers) are inside the SRS site boundary would result in inaccurate and inappropriately skewed MFFF public exposure reporting. DCS believes it makes regulatory sense for the NRC and the DOE to have a similar reference for measuring and reporting public dose. The SRS site boundary is also well recognized by other federal and state agencies. For example, the Environmental Protection Agency (EPA) applies the requirements of the Clean Air Act at the boundaries of the SRS. EPA and the state and county emergency preparedness agencies also apply emergency preparedness requirements at the SRS site boundary for protective actions for members of the public. Finally, the different interpretations of where the members of the public actually reside, obscures the fact that the same degree of protection is provided to MFFF workers, SRS workers, and the public under both NRC and DOE regulatory regimes. This situation could potentially erode stakeholder perceptions associated with nuclear safety.

Summary

The preceding paragraphs describe various NRC and DOE documents that support DCS' licensing approach that SRS workers who are outside the MFFF restricted area, but within the controlled area boundary are appropriately deemed to be "workers." Labeling these individuals as "members of the public" is inconsistent with Part 70 and would be a source of potential confusion to the many stakeholders involved with the licensing of the MFFF or regulation of the Savannah River Site. Finally, the documents referenced above all acknowledge that both the DOE and NRC radiological practices ensure the safety of facility workers, site workers (co-located workers), and the public. DCS fully intends to meet the requirements of Part 20 and Part 70.

Action:

None

2. Section 1.1.2.1, pp. 1.1-1 thru 1.1-2

Revise the description of the controlled area boundary to include only those areas to which Duke Cogema Stone & Webster (DCS) can limit access for any reason.

Section 70.61(f) states that each licensee must establish a controlled area for which they retain the authority to exclude or remove personnel and property. The area that is defined by DCS in Section 1.1.2.1 includes areas within the Savannah River Site that the DOE does not currently control access by physical structures, such as gates, barriers or fences. This includes, for example, the area north of South Carolina Route 278 and the area southwest of South Carolina Route 125.

Response:

DCS believes the controlled area boundary as described in the CAR is consistent with the regulations. The draft rule contained in SECY-98-185, *Proposed Rulemaking - Revised Requirements for the Domestic Licensing of Special Nuclear Material* (dated July 30, 1998), did include the definition for a physical barrier for (what was then called) the Controlled Site Boundary. However, the final rule (65 FR 56211) specifically did not include such a requirement. Rather, it included the provisions of 10 CFR §70.61(f), with which DCS intends to comply through the implementation of §70.61(f)(2). The subject of Controlled Site Boundary and Controlled Area was discussed in a March 24, 1999, public meeting concerning the Part 70 rulemaking. The transcript reveals that the NRC distinguished between a Controlled Site Boundary (i.e., consisting of physical barriers) and a Controlled Area. The deletion of the Controlled Site Boundary (requiring a physical barrier) occurred in SECY-99-147 (June 2, 1999) and the wording in the final rule is essentially the same as that in the June 1999 draft.

Limited access to certain areas of the Savannah River Site is provided to non-workers and members of the public; the controls associated with such access are as described below. The SRS boundary established by the Department of Energy is recognized by other federal and state regulatory agencies as a controlled boundary. The DOE either directly controls access to all areas of the site via fences and checkpoints, or indirectly through procedures in place for those limited situations where access is provided to non-badged personnel. The protocol described in Section 1.1.2.1 of the CAR will provide for MFFF integration with the existing SRS programs that ensure protection of the general public (non-badged personnel) during these times of site access.

The general public has limited access to certain areas of the site through travel on four roadways, controlled using appropriate warnings and advisories, use of barricades when necessary, and procedures for the SRS security contractor and offsite law enforcement authorities to close down roads and/or implement immediate access control measures, if necessary.

DOE conducts annual hunts to reduce the site's deer population for approximately 30 days per year. These hunts are strictly controlled: hunters are briefed on how they will be notified in an emergency, transported to the hunt site by SRS personnel, placed in pre-determined locations (where they remain for the duration of the hunt), notified and collected in the event of an emergency.

The South Carolina Department of Natural Resources (SCDNR) manages the Crackerneck Wildlife Management Area and Ecological Reserve, a tract of land south of Jackson off SC125, bounded by Three Runs Creek, the Savannah River, and the site boundary. Access roads are equipped with unmanned barricades, limiting travel from SC125. For 45 days each year, SCDNR opens Crackerneck to hunters, and on Saturdays, the area is also opened for recreational use (e.g., hikers, bikers, etc.). On the days that Crackerneck is open, SCDNR staffs the entrance, requires visitors to sign in and out, and provides safety and emergency alerting information. SCDNR officials carry remote worker pagers while at SRS. In addition, the SRS security contractor would dispatch vehicles across the Crackerneck road network, using a combination of a hi-lo siren and public address system announcements to alert visitors to an emergency.

The CSX has right-of-way for two rail routes that run through the site – a line between Florence, SC, and Augusta, GA; and a main line between Yemassee, SC, and Augusta. The two lines join within the site boundary. Early in 1989, CSX discontinued service on the line from this SRS junction to Florence. The Yemassee-Augusta line remains active with U.S. Department of Transportation-regulated commercial rail shipments traversing the site regularly. The CSX Switching Yard in Augusta maintains radio communications with trains in their jurisdiction, and will divert rail traffic before it enters the site in the event of an emergency. CSX would also be able to identify trains that are on the site at the time of the emergency to verify rapid transit.

When site visitors are issued temporary badges, they receive printed materials that include instructions on the alerting signals used onsite and appropriate actions to take; SRS hosts are responsible for the visitors in their care.

Non-DOE workers who conduct regular business onsite are required to take annual training analogous to that received by DOE employees and contractors. Such workers are afforded unescorted access only to the general site and specific work sites. These badged workers include the D-Area Powerhouse Staff, University of Georgia Ecology Laboratory Staff, U.S. Forest Service, Three Rivers Landfill Staff, vendors (e.g., food service, vending machines, pest control, etc.), and various state and county agency representatives.

Action:

None

3. Section 1.1.2.1, pp. 1.1-1 thru 1.1-2

Complete the description of the 10 CFR 70.61(f) protocol described in Section 1.1.2.1 to include members of the public who are in the controlled area and outside the Savannah River Site F-Area.

Section 70.61(f) describes a method of complying with the 10 CFR 70.61 performance requirements when non-workers are expected to perform ongoing activities in the controlled area. It is NRC policy that individuals who are not closely and frequently connected to the licensed activity should be considered members of the public. Therefore, though Section 1.1.2.1 describes a protocol that the applicant intends to use to justify the application of performance requirements for workers to individuals in the Savannah River Site F-Area, the applicant should also describe whether or not performance requirements applicable to individuals outside the controlled area will be applied to non-workers on the Savannah River Site outside F-Area.

Response:

The response to Question 1 provides the basis for DCS' position that SRS workers are not "members of the public." The response to Question 2 identifies the instances where the general public may have limited access to certain areas of the Savannah River Site.

The protocol described in Section 1.1.2.1 will provide for integration with the existing SRS programs that ensure protection of the general public (non-badged personnel) during these times of site access.

Action:

Revise the description of the DCS/DOE protocol described in Section 1.1.2.1 to state that the protocol will provide for MFFF integration with existing SRS controls established for general public access to the site.

4. **Section 1.2.1, pp. 1.2-1 thru 1.2-2**

Provide information on the presence and operations of any other companies on the site. If no such companies or operations are applicable, the application should so state.

NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility," (SRP) Section 1.2.3.A.vi indicates that information of the presence and operations of any other companies on the site is needed to review institutional arrangements. The application has not provided information on the presence and operations of other companies.

Response:

During construction, DCS anticipates approximately eight major subcontractors will be on site. During operation, it is anticipated that some of the support services (e.g., security, laundry) will be subcontracted; however, DCS will be the operator of the facility. No subcontracts have been issued yet; therefore, the identity of companies that will be on site is not yet known. Please refer to Section 4.2 of the CAR for an explanation of the arrangements that will be in place with subcontractors.

Action:

None

5. Section 1.1.3.5.3, p. 1.1-6

Specify the location where radioactive solid waste is sorted, packaged or stored until transferred to DOE facilities.

Sections 7.4.3.2.C and 7.4.3.2.F of the SRP recommend that information be provided on fire resistivity and combustibility of radioactive waste and radioactive material storage facilities. The application does not describe fire protection measures for the temporary storage of radioactive solid waste.

Response:

Radioactive waste sorting, counting, and storage are performed in the Waste Air Lock (Room B-255), Counting Room (Room B-253), and Waste Storage Room (B-254) located together in the MOX Processing Area. Packaged radioactive 55-gallon waste drums from the process rooms are sent to the Waste Air Lock. The waste drums are screened in the Waste Air Lock for criticality safety as part of the sorting process. Full pallets (containing four waste drums) or individual drums can be sent from the Waste Air Lock to the nuclear counting unit in the Counting Room for neutron counting and gamma spectrometry. After counting, the waste drums are returned to the Waste Air Lock in order to be sorted by the nature of the waste contained and placed on segregated pallets. The drums remain in the Waste Air Lock until a pallet is full. After waste pallets are full, they are sent to the Waste Storage Room where they are stored prior to shipment to DOE.

The combustible solid radioactive waste in the waste drums is primarily composed of discarded gloves, disassembled HEPA filters, and HP sampling media (air filters and smears). The Waste Air Lock, Counting Room, and Waste Storage Room are separate fire areas. Each of these fire areas is protected by automatic non-halogenated clean agent suppression systems. The passive fire protection features in these waste rooms, which have a minimum fire rating of two hours, include walls, floors, ceilings, penetrations, and doors separating each of these rooms from other fire areas.

Action:

The information presented in this response will be incorporated into the next amendment of the CAR.

6. **Section 1.2.4.1, p. 1.2-3**

Address the requirements in 10 CFR 70.17 in an exemption request applicable to the submittal of a decommissioning funding plan.

Under 10 CFR 70.25(a), a applicant for a specific license for possession and use of unsealed special nuclear material having the possession limits requested must provide a decommissioning funding plan. In the application, DCS requested an exemption stating that the facility will be turned over to DOE at the conclusion of the contract and DOE will assume responsibility for decommissioning.

Response:

DCS will submit under separate cover a request for a license condition as stated in CAR Section 1.2.4.1, along with a request for exemption from the requirements of 10 CFR § 70.38 and 10 CFR § 70.25(a)-(f) relating to the responsibility for decommissioning and maintaining decommissioning funding. (Note: the exemption is not being sought from the record keeping requirements of 10 CFR § 70.25(g)).

Action:

DCS will submit an exemption request under separate cover no later than submittal of its license application for the possession and use of special nuclear material.

7. **Section 1.2.4.2, p. 1.2-3**

Address the requirements in 10 CFR 70.17 in an exemption request applicable to using DOE authority under the Price Anderson Act for liability coverage.

Under 10 CFR 140.13a, applicants for plutonium processing and fuel fabrication plants must provide protection in the form specified in 10 CFR 140.14 in the amount of $200,000,000. In the application, DCS is proposing to request an exemption from the regulations to use the DOE provisions of the Price Anderson Act for liability coverage. In Section 2.5 of the application, the applicant describes the intended arrangements with DOE for indemnification.

Response:

As discussed in CAR Section 1.2.4.2, DCS will submit, pursuant to 10 CFR 140.8, a request for exemption from the requirements of 10 CFR 140. Since the Price Anderson liability coverage is for nuclear liability, it is not required for construction authorization. (Based on 10 CFR 140.2(a)(3), Part 140 applies after receipt of a possession and use license.)

Action:

DCS will submit an exemption request under separate cover no later than submittal of its license application for the possession and use of special nuclear material.

DCS will revise CAR Section 1.2.4.2 in the next CAR update to delete the phrase "If the NRC determines that the financial protection requirements of 10 CFR 140 apply to the MFFF."

8. Section 1.3.2.1.2, p. 1.3.2-2

Revise the statement in Section 1.3.2.1.2 which states "There are no facilities or populations within 5 mi (8 km) of the MFFF site that are not part of the Savannah River Site complex," to include the 1,400 acre Three Rivers Solid Waste Authority landfill.

Section 1.3 of the SRP requests that the applicant provide a description, distance, and direction to nearby public facilities. The Three Rivers Solid Waste Authority landfill is located north of Upper Three Runs creek and east of South Carolina Route 125.

Response:

The Three Rivers Landfill is operated by the Three Rivers Solid Waste Authority, but is still controlled by DOE. It is approximately 4 miles west of the MFFF site. The Three Rivers Landfill is part of the Savannah River Site complex and access is controlled consistent with SRS procedures. Therefore, the statement in CAR Section 1.3.2.1.2 is considered correct without change.

Action:

None

9. **Section 1.3.3.4.1, p. 1.3.3-3**

Provide the maximum rotational speed (V_{rot}), the maximum translational speed (V_{tr}), the radius (R_m) of maximum rotational speed in the Table 1.3.3-7, and the equations for atmospheric pressure change (APC) and the rate of APC in the Table 1.3.3-8.

Regulatory Guide 1.76 provides design basis tornado characteristics of maximum wind speed, maximum rotational speed, maximum translational speed, radius of maximum rotational speed, pressure drop, and rate of pressure drop. However, the Table 1.3.3-8 did not provide these characteristic and equations for pressure drop and rate of pressure drop. The staff could not verify the 70 psf at 31 psf/sec for PC-3 and 150 psf at 55 psf/sec for PC-4 without V_{rot}, V_{tr}, R_m, and corresponding equations.

Response:

The parameters are based on the methodology used in "Assessment of Tornado and Straight Wind Risks at the Savannah River Site" (McDonald, 1982) and "Natural Phenomena Hazards Modeling Project: Extreme Wind/Tornado Hazards Models for Department of Energy Sites" (Coats and Murray, 1985) . The values of the parameters for the PC-3 and PC-4 design basis tornado (DBT) speeds of 180 and 240 miles per hour (mph) are given below.

Parameter	**PC-3**	**PC-4**
V_{max} = DBT, mph	180	240
V_t = Maximum Translational Speed, mph	50	50
V_{ro} = Maximum Rotational Speed, mph $= V_{max} - V_t$	130	190
V_θ = Maximum Tangential Speed, mph $= 0.89 \times V_{ro}$	115.7	169.1
APC = Maximum Atmospheric Pressure Change, psf $= 0.00238 \times (V_\theta) \times (5{,}280/3{,}600)^2$ rounded up to	68.5 70	146.4 150
R_{max} = Maximum Radius, ft	165	195
Rate of Pressure Drop, psf/sec $= APC \times (V_t / R_{max}) \times (5{,}280/3{,}600)$	31	55

The tornado characteristics in Regulatory Guide 1.76 are based on an annual exceedance probability of approximately 1E-07. Since Regulatory Guide 1.76 was issued (in 1974), much has been learned regarding tornado effects (from an engineering perspective) that suggests adoption of larger exceedance probabilities. Some increase over 1E-07 is justified, particularly for facilities that pose substantially smaller risks than commercial nuclear power plants.

As noted in CAR Section 1.3.3.4.1, the recurrence frequency for tornado events for the MFFF is selected to be 2E-06, comparable to the PC-4 criteria described above. This recurrence frequency is more consistent with later NRC sponsored studies such as NUREG CR-4461, *Tornado Climatology of the Contiguous United States*. In addition, this recurrence interval is comparable to that used for DOE reactor facilities. Design for this magnitude of tornado event ensures that high consequence tornado-induced events will be highly unlikely.

References:

Coats, D. W. and R. C. Murray, 1985, "Natural Phenomena Hazards Modeling Project: Extreme Wind/Tornado Hazards Models for Department of Energy Sites", UCRL-53526, Rev. 1, Lawrence Livermore National Laboratory, University of California, August 1985.

McDonald, J.R., 1982, "Assessment of Tornado and Straight Wind Risks at the Savannah River Site", October 1982.

NUREG CR-4461, *Tornado Climatology of the Contiguous United States*

Action:

None

10. Section 1.3.3.4.1, p. 1.3.3-3

Provide the justification for the missile criteria in Table 1.3.3-8.

For example, the impacting velocity for an automobile in Table 1.3.3-8 is about 10 percent of design basis tornado windspeed. Section 3.5.1.4 of SRP recommends all plants be designed to protect safety-related equipment against damage from missiles which might be generated by the design basis tornado for that plant. They should include at least three objects: a massive high kinetic energy missile (4000 lb. automobile) which deforms on impact, a rigid missile (125 kg 8" armor piercing artillery shell) to test penetration resistance, and a small rigid missile (1" solid steel pipe) of a size sufficient to just pass through any openings in protective barriers, all impacting at 35 percent of the maximum horizontal windspeed of the design basis tornado. They are assumed to have the same velocity in all directions. The automobile impact should be considered at elevations up to 30 ft above all grade levels within 0.5 mile of the facility structures, and the other missiles at all elevations.

Response:

The design basis straight-line wind and tornado missiles, except for the PC-3 tornadic automobile missile, are based on the DOE-STD-1020 (DOE, 1996) requirements for PC-3 and PC-4 DOE sites. McDonald (McDonald, 1999) provides a rationale for the PC-3 and PC-4 design basis missiles criteria for DOE sites in relation to the observed data for tornadoes and results of simulation studies. Sections 2.7 and 3.8.4 of the McDonald report (McDonald, 1999) summarize the basis for the DOE missiles and for the corresponding speeds, respectively. As noted in the response to Question 9, tornado criteria for the MFFF have been selected based on a recurrence frequency of 2E-06/yr.

In the comprehensive report (McDonald, 1999), McDonald observes that the design basis missile criteria for DOE sites are different from the U.S. NRC criteria for nuclear power plants. Section 1.4 of the report (McDonald, 1999) addresses specifically NUREG-0800 3.5.1.4 criteria. The DOE criteria are more appropriate, the report (McDonald, 1999) states, because the missiles are selected based on field observations, and the level of risk is different from nuclear power plants. Similarly, the design basis tornado speeds are lower than those in the U.S. NRC criteria for reactors.

The PC-3 tornadic missile of an automobile weighing 3,000 pounds, not given in DOE-STD-1020 (DOE, 1996), represents large heavy missiles that roll and tumble along the ground (McDonald, 1999). Its design basis speed is based on the recommendation of McDonald-Mehta (McDonald-Mehta, 1997).

The missile criteria in Table 1.3.3-8 are appropriate for SRS and adequately represent the three types of missiles identified in NUREG-0800. These are a massive high kinetic energy missile that deforms on impact, a rigid missile to test penetration resistance, and a small rigid missile of a size sufficient to just pass through any openings in protective barriers.

References:

DOE, 1996, US Department of Energy, "Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities", DOE-STD-1020-94, Change Notice 1, January 1996.

McDonald, J.R., 1999, "Rationale for Wind-Borne Missile Criteria for DOE Facilities", UCRL-CR-135687, S/C B505188, September 1999.

McDonald-Mehta, 1997, "Tornado Hazard Assessment", Memorandum of November 7, 1997 from McDonald-Mehta Engineers to B. J. Gutierrez of DOE/SR related to the Accelerator Production of Tritium (APT) project.

Action:

None

11. Section 1.3.3.4.2, pp. 1.3.3-3 and 1.3.3-4

Justify the straight-line wind speeds in Table 1.3.3-7.

The Section 3.3.1 of SRP recommends the design wind speed be 100-year return period fastest mile of wind. American Society of Civil Engineers (ACSE) 7-98 provides map of basic wind speeds of 50-year return period fastest mile of wind for U.S., which have to be converted to 100-year return period fastest mile of wind. For example, the ACSE 7-98 shows about 100 mph with annual probability of 0.02 for MFFF location. With annual probability of 0.01, the ASCE 7-98 wind speed might be 115 mph (i.e., 1.15 x 100 mph). However, the Table 1.3.3-7 shows 88 mph with annual probability of 0.01. Therefore, the adequacy of the Table 1.3.3-7 is questionable.

Response:

The values for the estimated maximum 3-second gust straight wind speeds in Table 1.3.3-7 are based on the site-specific investigation reported in "Tornado, Maximum Wind Gust, and Extreme Rainfall Event Recurrence Frequencies at the Savannah River Site" (Weber et al., 1998). The estimates provided in the table for straight wind are generated from a Fisher-Tippet Type I extreme value distribution function using historical wind speed (gust) data from Savannah River Site and the surrounding region (Weber et al., 1998). The methodology (Weber et al., 1998) is consistent with the Department of Energy criteria for site hazard characterization (DOE, 1996).

ASCE 7 maps are based on a similar methodology. However, their grid may not be as finely divided and likely does not include as much local data as the Savannah River Site report (Weber et al., 1998). Commentary to ASCE 7 states, in part "There was insufficient variation in 50-year speeds over the ¾ of the lower 48 states to justify contours. The division between the 90 and 85 mph ... regions, which follows state lines, was sufficiently close to 85 mph contour..." Section 6.5.4.2 of ASCE 7 allows estimation of basic wind speeds from regional climatic data subject to the cautions indicated in its commentary on the section.

References:

DOE, 1996, US Department of Energy, "Natural Phenomena Hazards Site Characterization Criteria", DOE-STD-1022-94, March 1994, Change Notice 1, January 1996.

Weber, A.H., et al, 1998, "Tornado, Maximum Wind Gust, and Extreme Rainfall Event Recurrence Frequencies at the Savannah River Site", WSRC-TR-98-00329, Westinghouse Savannah River Company, Aiken, SC.

Action:

None

12. **<u>Section 1.3.6, pp. 1.3.6-1 thru 1.3.6-86</u>**

Provide the following items for the Probabilistic Seismic Hazard Assessment (PSHA):

a. PSHA inputs (e.g., logic tree or what other input was used to define the input distribution parameters).

b. Integrated PSHA hazard curves at several important frequencies (e.g., Peak Ground Acceleration, 1, 2.5, 5 and 10 Hz) for both rock and soil surfaces.

c. Magnitude-distance desegregation results.

d. Uniform hazard spectra for both rock and soil surfaces.

The application states that, "An SRS-specific PSHA was developed using bedrock outcrop EPRI and LLNL hazards and SRS site-specific properties including soil column thickness..." yet the relevant inputs and hazard results of these original probabilistic seismic hazard analyses were not provided. These items are essential for evaluating the adequacy of the PSHA, PSHA results, and whether these results are applicable to the seismic design of a particular site, as delineated in Position 3 of Regulatory Guide 1.165 (U.S. Regulatory Commission, 1997b), Section 2.5.2 (Vibratory Ground Motion) of NUREG-0800 (U.S. Regulatory Commission, 1997c), and Section 7 (Guidance on Documenting the PSHA Process and Results) of NUREG/CR-6372 (U.S. Nuclear Regulatory Commission, 1997a).

Response:

Response to this Question is provided in two parts. The first portion directly responds to the technical questions posed in items 12a through 12d above. In these responses, DCS describes the PSHA that has been performed to develop criteria for site-wide application with DOE facilities at SRS. The responses to Question 14 and Question 15 demonstrate that these criteria are also appropriate for the MFFF site, and that a further MFFF site-specific PSHA is not required.

The second portion of the response to this question addresses the issue raised in the Question's commentary about lack of clarity in explaining PSHA methodology and approaches. In response to this issue, a new CAR section is proposed to supplement information already presented. This new section provides a "roadmap" to the PSHA process, and draft CAR text is provided after the References following Response 12d below.

Response 12a.

The Electric Power Research Institute (EPRI) and the Lawrence Livermore National Laboratory (LLNL) seismic hazard studies conducted for the eastern U.S. from the mid 1980s to early 1990s were the PSHAs used to develop design spectra at the SRS (WSRC, 1997). The basis for the use of the arithmetic mean of the two PSHAs is contained in USDOE (1996). This response and the response to Question 19 will point to particular sections as necessary in the EPRI and LLNL documentation rather than repeating or duplicating material. Where requested information is not in the original reports and is available, this information is provided herein. Where requested information is not available, it is so noted. The development of design spectra at the SRS is discussed in the response to Question 19d.

The discussions of the PSHA inputs are treated separately for EPRI and then LLNL. Note that in the development of the SRS design spectra, the DOE contracted separately to Jack Benjamin and Associates and the LLNL to provide hard rock results for the SRS. SRS central site coordinates were used for the computation of EPRI and LLNL bedrock PSHAs. While the SRS results are not specifically documented in the EPRI and LLNL reports, the PSHA for the deep soil site plant Vogtle site (located approximately 19 km west of the center of the SRS) are contained in both the EPRI and LLNL documentation. Because of the close proximity of the Vogtle site to the SRS, references to data summaries and results for site Vogtle will be used.

EPRI (1988)

The EPRI (1988) PSHA for the Eastern U.S. is documented by volume by the six consulting companies independently contracted to conduct the study. The specific volume containing the seismogenic models for each company is as follows:

Team #	Company	EPRI (1988) Reference Vol #
1	Weston Geophysical	5
2	Dames and Moore	6
3	Law Engineering	7
4	Woodward-Clyde	8
5	Bechtel	9
6	Rondout	10

Each of these volumes, corresponding to a consulting company team contains the seismogenic maps (source zones), alternate source zones, magnitude distributions, and rates of seismicity distributions for each zone. Although "logic trees" were not illustrated in these reports, the reports contain much of the basis for decision-making logic tree. Details of the rates of seismicity and the maximum magnitude used by each team are contained in the volumes identified above. Seismic source breakdown and magnitude distribution for the Vogtle site are listed in Vol. 4, Table 6-6e. Sensitivity of the Vogtle hazard to source combination by team is illustrated in Vol. 4, Figures 6-6e through 6-6j. Sensitivity to maximum magnitude at Vogtle is illustrated in Vol. 4, Figures 6-6k through 6-6p. Sensitivity to seismicity option is illustrated in Vol. 4, Figures 6-6q through 6-6v. Note that the surface hazard results for the Vogtle site were completed using a generic deep soil site response. The generic site response was not site-specific (to Vogtle or the SRS) and is unconservative at some oscillator frequencies. For this reason, hard rock PSHAs were used and site-specific corrections were applied for the SRS.

In the development of the seismogenic zones, all six teams confined the "Charleston" source zone to the Charleston area and/or to the central South Carolina coastal area for their base-case seismic zonation maps. This effectively limits the exposure of higher seismic hazard to the coastal and central areas of the state of South Carolina.

LLNL (1989, 1993)

The LLNL study was funded by the NRC to conduct an expert opinion based PSHA for the eastern U.S. nuclear power plants. Because of significant differences in the uncertainties portrayed by the results, the DOE funded an additional analysis to update the attenuation and seismicity models. This update also used a generic soil-site response model. Subsequently, the SRS-specific hard-rock PSHA was completed in 1996.

The LLNL hazard methodology and input data used for the SRS is reported in Savy et al. (1993), will hereafter be referred to as LLNL93. LLNL93 documents the ground motion and seismicity elicitation workshops and updates the previous LLNL report (Bernreuter et al., 1989) (hereafter referred to as LLNL89).

Section 2 of LLNL89 provides a complete description of the methodology used in the calculation of PSHA, including a description of how uncertainty and expert opinion are treated. The PSHA inputs are described in Section 3. Section 3 documents the elicitation, including questionnaires, ground motion (Table 3-2) and seismicity experts (Table 3-1), seismic zonation maps (Appendix B), and rates of seismicity (Appendix B). Section 4 makes comparisons to earlier LLNL studies but does not include the Plant Vogtle site. The Plant Vogtle soil surface results are given in section 2.16 of Volume 3. The seismicity models and ground motion models contained in LLNL89 were updated in LLNL93; consequently, the results contained in LLNL89 were superseded by LLNL93. However, much of the methodology and the seismic zonation maps contained in LLNL89 were used unchanged in LLNL93.

Of the 11 seismicity experts used in LLNL89, all experts except Nos. 6 and 11 confined the "Charleston" zone to the Charleston, SC area and/or the central South Carolina coastal area for their base seismic zonation maps. Expert 6 used a considerably larger area for the "Charleston" zone that covered a significant portion of the state of South Carolina. Expert 11 used a "Charleston" zone that covered most of the Coastal Plain from Alabama to Virginia.

The seismicity elicitations revised in LLNL93 are contained in Appendix B of that report. Tables A1 through A13 contain the expert seismicity and magnitude distributions. Soil surface PGA hazard for the Plant Vogtle site is contained in Appendix E.

Response 12b.

EPRI hard-rock mean and fractile hazard for the SRS are illustrated in the attached Figures 1a, 1b, 1c, 1d, and 1e corresponding to 1-, 2.5-, 5-, 10-Hz and PGA, respectively.

LLNL hard-rock mean and fractile hazard for the SRS are illustrated in the attached Figures 2a, 2b, 2c, 2d, 2e, and 2f corresponding to 1-, 2.5-, 5-, 10-, 25-Hz and PGA, respectively.

Corresponding EPRI and LLNL SRS hazard curves for generic soil are currently unavailable. Efforts are underway to collect this information. Note that this information was not used in the development of SRS design criteria. WSRC (1997) describes development of PC-3 and PC-4 design spectra.

Response 12c.

EPRI hard-rock mean hazard deaggregation for the SRS are illustrated in the attached Figures 3a, 3b, 3c, 3d, and 3e corresponding to 1-, 2.5-, 5-, 10-Hz and PGA respectively. The deaggregations are at the annual probability of exceedance of approximately 10^{-4}.

LLNL hard-rock mean hazard deaggregation for the SRS are illustrated in the attached Figures 4a, 4b, 4c, and 4d corresponding to 1-, 2.5-, 5- and 10-Hz respectively. The deaggregations are at the annual probability of exceedances of approximately 2x10-4.

Corresponding EPRI and LLNL deaggregations for generic SRS soil are currently unavailable. Efforts are underway to collect this information. Note that this information was not used in the development of SRS design criteria. WSRC (1997) describes development of PC-3 and PC-4 design spectra.

Response 12d.

EPRI mean and fractile hard-rock uniform hazard spectra (UHS) for the SRS are illustrated in the attached Figure 5 for an approximate annual probability of exceedance of 10-4.

LLNL mean and fractile hard-rock UHS for the SRS are illustrated in the attached Figure 6 for an approximate annual probability of exceedance of 10-4.

Combined EPRI and LLNL mean hard-rock UHS are given in WSRC (1997).

Corresponding EPRI and LLNL UHS for generic soil are currently unavailable. Efforts are underway to collect this information. Note that this information was not used in the development of SRS design criteria. WSRC (1997) describes development of PC-3 and PC-4 design spectra.

References:

Bernreuter, D.L., J.B. Savy, R.W. Mensing and J.C. Chen, 1989. Seismic Hazard Characterization of 69 Nuclear Power Plant Sites East of the Rocky Mountains, NUREG/CR-5250, UCID-21517.

Electric Power Research Institute, 1988. Seismic Hazard Methodology for the Central and Eastern United States, Vol. 1-10, NP-4726.

Savy, J.B., A.C. Boissonnade, R.W. Mensing and C.M. Short, 1993. Eastern U.S. Seismic Hazard Characterization Update, Lawrence Livermore National Laboratory, July 20, 1993.

U.S. Department of Energy, 1996. DOE Standard: Natural Phenomena Hazards Assessment Criteria, DOE-STD-1023-95, Change Notice #1, Washington, D.C., January 1996.

WSRC, 1997. SRS Seismic Response Analysis and Design Basis Guidelines, WSRC-TR-97-0085, Rev. 0.

Draft New CAR Section

To clarify the process used in developing the site-wide PSHA and design criteria, a new subsection will be added to the CAR (in Section 1.3.6). Content of this new section will be structured similar to the text presented below.

1.0 Introduction

WSRC has used a disciplined, systematic approach to develop the PC-3 and PC-4 site-wide design spectra. This approach has included the contributions of national and international consultants, oversight groups and panels that validated the WSRC procedure(s) and results. It is this baseline data that has been made available to the MFFF project for selection of design bases for the MFFF. The following sections provide the various elements of the PSHA process used to develop the PC-3 and PC-4 seismic design spectra and the regulatory approach to select the MFFF seismic design spectra.

2.0 Roles and Responsibilities of Participants and Consultants

2.1 **General** - This section provides insight into the participants and consultants that contributed to the development of the SRS PC-3 and PC-4 seismic design spectra that form the technical basis for selecting the MFFF Design Earthquake. This section highlights the input of participants and consultants used to characterize the SRS bedrock geology and soils, seismic sources, attenuation relationships and static and dynamic properties of the Savannah River Site (SRS) soil profile. The development of the PC-3 and PC-4 Seismic Design Spectra is documented in WSRC, 1997 prepared by the Site Geotechnical Services (SGS) Department of Westinghouse Savannah River Company (WSRC).

2.2 **Introduction** - SGS was formed in 1992 to provide geological, seismological, geotechnical (GSG) and geo-environmental services for the SRS under a single authority, the Site Chief Geotechnical Engineer. Through centralization of all geoscience data into a single database, SGS has ensured that facility safety operations, long-term environmental restoration and remediation, and development of new missions are always based on the most modern, comprehensive, accurate GSG data available. General areas of SGS responsibility included:

- Develop GSG codes, standards, procedures, policies and practices;
- Manage and control of funds for sitewide efforts such as the development of seismic design spectra;
- Define and arbitrate the final official SRS position on GSG issues; and
- Perform geotechnical and environmental engineering consulting services for the various operating divisions at SRS.

SGS consists of approximately sixty (60) multi-disciplined professionals with degrees in such areas of specialty as geology, seismology, geotechnical engineering, biology, ecology and environmental engineering.

2.3 **Tier 1 Documentation** – This documentation includes the reports and its appendices prepared by WSRC's SGS in response to sitewide ground motion initiatives and in support of critical mission facilities. WSRC, 1997 is an example of a report prepared in support of a sitewide initiative to develop seismic design spectra using a Probabilistic Seismic Hazard Analysis (PSHA) approach with a deterministic historical check. Other reports related to ground motion include WSRC, 1998 and WSRC, 1999. Supporting the preparation of reports by SGS were experts with national and international reputations in such areas as geology, seismology and geotechnical engineering. Table 2-1 presents a list of consultants used to prepare major SRS reports with their areas of expertise. The acknowledgement section in a report indicates which consultants, if any, contributed to the preparation of the report. The names on the report and the approval signature(s) indicate those who are professionally responsible for the report.

2.4 **Tier 2 Documentation** – This documentation consists of the much larger body of background information that comprises the analysis documentation and the results of reviews by various oversight groups and panels. This second–tier material is maintained by SGS and the Site Chief Geotechnical Engineer in an appropriately accessible, usable and (if appropriate) auditable form. In accordance with WSRC procedures, calculations are prepared and checked as part of WSRC report preparation procedures. The calculation (WSRC, 2001) submitted to demonstrate that the soil properties at the MFFF site fall within the range used to develop the SRS PC-3 and PC-4 seismic design spectra is an example of Tier 2 Documentation. Tier 2 documentation also includes the results of reviews by oversight groups and panels.

The Defense Nuclear Facility Safety Board (DNFSB) and its staff and consultants have been active in the oversight of seismic design activities that led to the development of the PC-3 and PC-4 seismic design spectra. Records of DNFSB activities are available as Tier 2 documentation.

The Department of Energy (DOE) and WSRC, when appropriate, has also assembled expert panels to support the design efforts associated with critical mission facilities. Since 1992, DOE and WSRC assembled three expert panels to support the efforts of the Replacement Tritium Facility (RTF) Geotechnical Investigation and In-Tank Precipitation Facility (ITP) and H-Tank Farm Geotechnical Report. W.F. Marcuson III and J. Mitchell formed the 1993 geotechnical panel for RTF and W.F. Marcuson III, J. Mitchell, and G.F. Sowers formed the 1995 geotechnical panel for ITP. WSRC also assembled a Senior Seismic Advisory Panel for the ITP project in 1994. R.J. Budnitz (Chairman), J. Carl Stepp, J.W. Reed, J.M. Roesset, and J.H. Schmertmann were members of the Senior Seismic Advisory Panel. The contributions of these expert panels are documented and are similarly available as part of the Tier 2 documentation for the RTF and ITP projects. While these expert panels did not directly contribute to the development of the PC-3 and PC-4 spectra, they assisted WSRC in the development of the approach used for geotechnical investigations and seismic design of structures.

2.5 **Summary** – In recognition of the uncertainty in estimating annual frequencies of earthquake-caused ground motions, WSRC's SGS developed and implemented a process that allowed for multi-disciplinary evaluation and the development of a seismic design bases that has kept pace and contributed to current industry technologies and practices. It is this baseline data that has been made available to DCS and the MFFF project.

TABLE 2-1

LIST OF CONSULTANTS

Consultant	Area of Expertise
Gail Atkinson	Engineering models of earthquake ground motion using semi-empirical models
Dave Boore	Engineering models of earthquake ground motion using semi-empirical models
Kenneth Campbell	Engineering models of earthquake ground motion using empirical data
Allin Cornell	Seismic hazard assessments, engineering seismology
Joe Fletcher	Site response and seismic instrumentation
Robert Pyke	Assessments of soil site dynamic characteristics and response
S. Rouhani	Statistical characterization of seismic velocity profiles
Walter Silva	Engineering models of earthquake ground motion and response spectra
Carl Stepp	Seismic hazard assessments
Kenneth Stokoe	Geophysical characterization of soil and rock sites; laboratory measurements of dynamic properties of soil and rock samples
Pradeep Talwani	Characterization of seismic sources
Gabe Toro	Statistical characterization of site properties for site response analysis
Robert Youngs	Site characterization for seismic response
John Clark	Seismic Reflection and Borehole Geophysics
Allan Dennis	Structural Geology
Robert D. Hatcher	Structural Geology
Ernie Majer	Geophysics
William Domoracki	Geophysics
Carl Costantino *	Geotechnical engineering and site response
Jeff Kimball*	Seismic hazard assessments, engineering seismology, DOE Guidance
Brent Gutierrez *	Natural hazards, DOE guidance

* DOE Reviewers

3.0 **Comparisons with Other PSHA Studies** – Section 4.0 NUREG/CR-5250 (Bernreuter et al., 1989) compares the results of NUREG/CR-5250 with previous results from LLNL and previous studies by others. The comparisons show good agreement. Section 4.0 of NUREG/CR-5250 provides the details.

In 1999, WSRC initiated a study to evaluate the differences between the building code hazard assessment (National Earthquake Hazard Reduction Program) and the site-specific hazard evaluations used for SRS building code design. Also, WSRC compared the SRS site-specific bedrock hazard with the USGS hazard, corrected to account for SRS conditions, (Frankel, 1999). Details regarding the WSRC study are contained in the Tier 1 Document, WSRC, 1999. For ease of reference, the results of this study are summarized in the paragraphs that follow.

3.1 **Summary of Findings** – Regarding development of SRS site-specific hazard from USGS National Map input, WSRC, 1999 concluded:

- The hard-rock PSHA is consistent with the earthquake source definition and recurrence rates contained in the National Map but results in different hazard because of differences in the assumed bedrock conditions. (The crustal model used for the USGS National Map contained a low-speed gradient that is significantly slower than the observed bedrock shear waves at the SRS);
- The USGS hard-rock hazard is close to LLNL at 2.5 and 5-Hz, but is greater than LLNL and EPRI for 1, 10-Hz and PGA;
- The USGS hard-rock hazard is generally more conservative then either EPRI or LLNL hazard because of a highly energetic source assumed for the Charleston zone. (The USGS National Map available at the time for this study employed a large magnitude earthquake (Mw=7.3) having a short return period (650 years) that occurs in an area source zone as close as 80 km to the SRS);
- The computed USGS soil surface hazard is less than the NEHRP spectrum recommended for shallow SRS soils;
- The computed USGS soil surface hazard is greater than the SRS design basis at 1 Hz and
- The methodology of WSRC (1998) is useful to derive site-specific soil surface hazard from hard-rock hazard disaggregation.

4.0 **Internal Quality Control and Review** – WSRC performed the work in support of the MFFF Construction Application Request (CAR) in accordance with the WSRC Quality Assurance (QA) program and Criterion 1-6 and 15-18 of ASME/NQA-1-1989. DCS has approved WSRC as a supplier of services. Section 15 in the CAR provides additional details regarding quality control and review.

5.0 **PSHA Methodology**

5.1 **General** – A PSHA incorporates the source zone definition and ground motion prediction assessments required for a deterministic approach, but also considers the estimated rates of occurrence of earthquakes, and explicitly incorporates the uncertainties in all parameters. This approach predicts the probability of exceeding a particular ground motion value at a location during a specified period of time. This approach is essential for hazard mitigation of spatially distributed facilities having different risk factors. The current DOE criteria are probabilistic based. Details of PSHA methodology is provided in WSRC, 1997 and WSRC, 1998. However, for ease of reference the following paragraphs are provided below that discuss the DOE STD-1023-95 and the SRS Site-Specific PSHA.

5.2 **DOE STD-1023-95** – This standard provides guidelines for developing site-specific probabilistic seismic hazard assessments, and criteria for determining ground motion parameters for the design earthquakes. It also provides criteria for determination of design response spectra. Five performance categories are specified, from Performance Category 0 (PC0) for SSCs that require no hazard evaluation, to design of PC4, a desired performance level comparable to commercial nuclear power plants. These criteria address weaknesses in prior guidance by specifying Uniform Hazard Spectrum (UHS) controlling frequencies, requiring a site-specific spectral shape and a historic earthquake check, to assure that the Design Basis Earthquake (DBE) contains sufficient breadth to accommodate anticipated motions from historic earthquakes above moment magnitude (Mw) 6.

The fundamental elements of the criteria for higher hazard nuclear facilities (PC3 and PC4) are as follows:

1. A probabilistic seismic hazard assessment (PSHA) must be conducted for the site (or use an existing PSHA that is less than 10 years old).
2. A target DBE response spectrum is defined by the mean UHS.
3. Mean UHS shapes are checked by median site-specific spectral shapes, which are derived from de-aggregated PSHA earthquake source parameters. The median site-specific spectral shapes are scaled to the UHS at two specific frequencies (average 1-2.5, and 5-10 Hz).
4. Estimated site-specific ground motions from historical earthquakes (significant felt or instrumental with Mw > 6) are developed using best estimate magnitude and distance.
5. Spectral shapes are adjusted until DBE response spectra have a smooth site-specific shape.
6. Probabilistic assessment of ground failure should be applied if necessary (i.e., wherever there may be instances of liquefaction or slope failure).

Recently, NEHRP-97 criteria have been adopted by WSRC and DOE for evaluation of spectra for PC1 and PC2 facilities and structures. DOE-STD-1023-95 allows the use of building codes and/or alternate design criteria for PC1 and PC2 design. The NEHRP design criteria is defined as 2/3 of the maximum considered earthquake ground motion (i.e., 2/3 of the 2500 year UHS). WSRC, 1999 discussed in Section 3.0, provides a comparison of the uniform hazard spectrum (UHS) derived from the computed site-specific hazard (referred to as USGS soil surface hazard) to the NEHRP (1997) spectrum for the SRS. This comparison was of particular interest for deep-soil eastern U.S. sites, because it compared a building code design spectrum to a site-specific spectrum using the same hazard model and identical criteria.

5.3 **SRS-Specific Probabilistic Seismic Hazard Assessments** – An SRS-specific probabilistic seismic hazard assessment (PSHA) is critically dependent upon the local geological and geotechnical properties at the site or facility location. Past PSHAs, specifically those conducted by EPRI (NEI, 1994) and LLNL (Bernreuter, 1997; Savy, 1996) for the SRS, did not incorporate these detailed site properties and consequently, those soil hazard results were not appropriate for use at the SRS. An SRS-specific PSHA should account for soil properties derived from site geological, geophysical, geotechnical and seismic investigations (WSRC, 1997). An SRS-specific PSHA was developed using bedrock outcrop EPRI and LLNL hazard and SRS site properties including soil column thickness, soil and bedrock shear-wave velocity, and dynamic properties (WSRC, 1998).

The bedrock seismic hazard evaluations used for the SRS-specific soil surface hazard were the EPRI and LLNL results for bedrock for the SRS and vicinity (a later evaluation was completed using the U.S. National Map bedrock seismic hazard (WSRC, 1999; Frankel et al., 1996). These evaluations did not revise or confirm in any way the experts' evaluations of activity rates, seismic source zonation, or the decay of ground motion with distance used in EPRI and LLNL seismic hazard assessments. The analysis results in an SRS-specific hazard evaluation for a soil site by continuing the hazard from bedrock to the soil surface using detailed soil response functions. Earthquake magnitude and ground motion level dependence of the site response is accommodated by applying site response functions consistent with the distribution of earthquake magnitude and ground motion levels obtained from dissaggregating the bedrock uniform hazard spectrum.

Frequency and ground motion level dependent soil amplification functions (SAFs) developed in WSRC (1997) were used to account for the observed variations in properties throughout the SRS including: soil column thickness, stratigraphy, shear-wave velocity, and material dynamic properties, as well as basement properties. SAFs (frequency dependent ratio of soil response to bedrock input) were derived in WSRC (1997) by performing a statistical analysis of the response of bedrock spectra through realizable soil columns bounded by the observed variations in soil-column properties over the SRS. Ground motion level dependent distributions of SAFs were derived for each of 6 soil categories: three on crystalline basement and three on Triassic basement. Those SAF distributions were used to compute soil surface hazard.

The methodology to compute soil surface hazard was formalized by Cornell (1997). The technique is to difference the bedrock hazard disaggregation for a suite of bedrock motions and sum the probability of exceedance (POE) of surface motions using the appropriate magnitude and ground motion level-dependent soil/rock transfer functions. The approach yields soil surface hazard that would be obtained from correctly applying local site soil transfer functions to the ground motion attenuation model used in a PSHA. The analysis is repeated at the oscillator frequencies available in the bedrock hazard disaggregation and for each soil column thickness and bedrock type. The envelope of the hazard curves is taken from the soil and bedrock categories.

6.0 Results

6.1 PC-3 And PC-4 Site-Wide Design Spectra - The PC3 and PC4 site-wide design spectra fully implement DOE-STD-1023-95. DOE-STD-1023-95 specified a broadened mean-based UHS representing a specified annual probability of exceedance (for an SSC performance category) and a historical earthquake deterministic spectrum that ensures breadth of the UHS. For the SRS, the deterministic spectrum is represented by a repeat of the 1886 Charleston earthquake. The development of the SRS design basis spectra uses a statistical methodology to verify that a mean-based response is achieved at the soil free surface.

The EPRI and LLNL bedrock level uniform hazard spectra were averaged and broadened per DOE-STD-1023-95. Available SRS soil data were used to parameterize the soil shear-wave velocity profile. The parameterization was used to establish statistics on site response for ranges of soil column thickness present at the SRS. The mean soil UHS was obtained by scaling the bedrock UHS by the ground motion dependent mean site amplification functions.

The soil data used to develop the sitewide spectra incorporate the available SRS velocity and dynamic property database available to about mid-1996. The spectra are based on soil properties and stratigraphy from specific locations at the SRS, and are parameterized to represent the variability in measured properties. Because of the potential for variation of soil properties in excess of what have been measured at the SRS, the design basis spectra are issued as "committed" in accordance with the WSRC Quality Assurance Manual 1Q. The open item is the soil column variability used in the calculations. To eliminate the open item and upgrade the design basis spectrum to "confirmed," the soil parameters available at the specific site or facility where it is being used must be reviewed and determined to be consistent with the data parameterized in the study. The results of this review for the MFFF site are provided in the Tier 2 document cited in Section 2.4 (WSRC, 2001).

The current PC-3 and PC-4 sitewide spectra are based on the WSRC analysis developed in 1997, and incorporates variability in soil properties and soil column thickness. The design basis spectra for PC3 and PC4 are given in CAR Figure 1.3.6-13 and CAR Figure 1.3.6-14, respectively. Following the development of PC3 and PC4 design basis spectra and the PC1 and PC2 design basis spectra, the DNFSB had several interactions with the DOE and WSRC on seismic design spectra. As a result, additional conservatisms were applied to the PC3 spectral shape at high and intermediate frequencies. The shape change was incorporated in the Site Engineering Standard. The shape change, illustrated in CAR Figure 1.3.6-20, increased the low-frequency (0.1-0.5 Hz) portion of the PC-3 spectrum and also increased intermediate frequencies (1.6-13 Hz) of the design basis spectrum.

7.0 **External Peer Review** – In addition to reviews conducted in development of the SRS site-wide criteria, DCS has also initiated a series of peer reviews of appropriate technical topics during the development of the MFFF design. The MFFF Structural Consulting Board (SCB) was formed and chartered to provide senior oversight for overall MFFF design approaches and to perform periodic reviews of in-process results. Members of the SCB have included recognized industry experts such as Robert Kennedy, Carl Costantino, and Thomas Houston, as well as subject matter experts from within the DCS companies. SCB members have been involved in the selection of the design bases for the MFFF, and have concurred in their selection. Similarly, before its completion, the MFFF Site Geotechnical Report was the subject of a detailed peer review by a panel including other industry experts including Brent Gutierrez, Phillip Kasik, Jeffrey Kimball, Michael Lewis, and Bhasker Tripathi.

8.0 **Regulatory Approach** - Section 1.3.6.6 of the CAR describes selection of the Design Earthquake for the MFFF. Soil surface hazard relationships (acceleration versus mean annual probability of exceedance) presented in WSRC, 1998 are used to evaluate the relative probability of exceedance of the PC-3 and PC-4 accelerations, and the accelerations of intermediate spectra. A Regulatory Guide 1.60 horizontal spectrum scaled to 0.2g peak ground acceleration is demonstrated to lie between the PC-3 and PC-4 soil surface spectra. The spectral ordinates for the Regulatory Guide 1.60 spectrum are determined to have return periods greater than 10,000-years for frequencies of significant structural interest (2.5 Hz, 5 Hz, and 10 Hz).

After review of SRS regional hazards, DCS has determined that though near-field earthquakes are not dominant, their contribution to the MFFF site hazard is potentially significant. Therefore, rather than 2/3 the horizontal accelerations, the MFFF vertical component of earthquake motion at the soil surface is selected as the Regulatory Guide 1.60 vertical spectrum scaled to 0.2g peak ground acceleration. As discussed in CAR Section 1.3.6.6 (to be updated - Question 18), this results in vertical and horizontal spectra that are consistent with the guidance in ASCE 4-98 and Regulatory Guide 1.60, and appropriately consider the potential effects of near-field earthquakes.

9.0 References

Bernreuter, D.L., Savy, J.B., Mensing, R.W. and Chen, J.C., January 1989. "Seismic Hazard Characterization of 69 Nuclear Plant Sites East of the Rocky Mountains," LLNL NUREG/CR-5250, Volume 1.

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WSRC, 1998. Soil Surface Seismic Hazard and Design Basis Guidelines for Performance Category 1 & 2 SRS Facilities, by R.C. Lee, WSRC-TR-98-00263, Rev. 0.

WSRC, 1999. Computation of USGS Soil UHS and Comparison to NEHRP and PC1 Seismic Response Spectra for the SRS, by R.C. Lee, WSRC-TR-99-00271, Rev. 0.

WSRC, 2001. Applicability of SRS Site-Wide Spectra to the MFFF Site. Calculation Number K-CLC-F-00049, Rev. 0.

Action:

Incorporate, as appropriate, a new section in the CAR similar to that of **Draft New CAR Section** above.

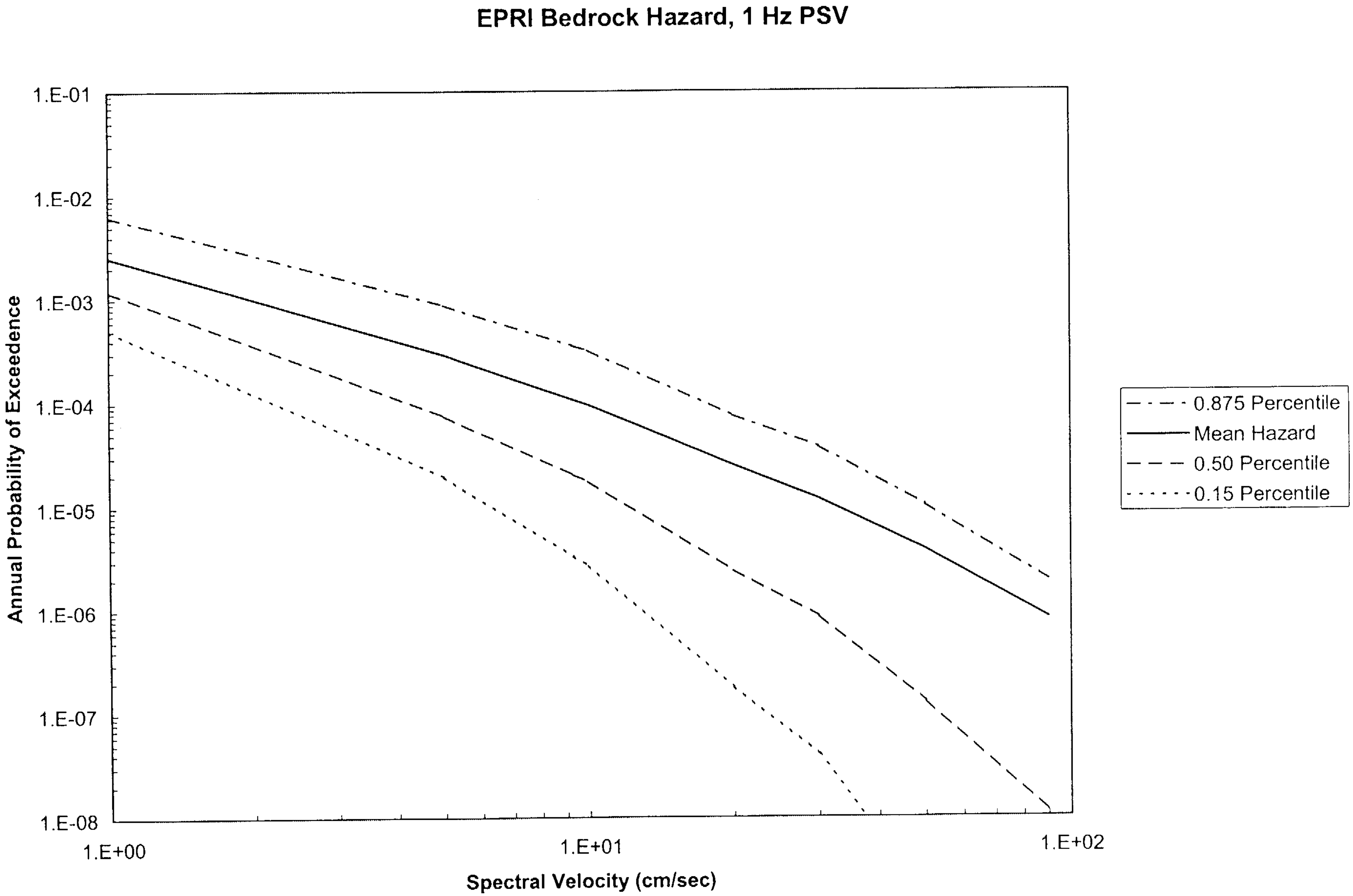


Figure RAI-012-1a. EPRI mean and fractile hazard for SRS hard-rock site conditions for a 1-Hz oscillator frequency.

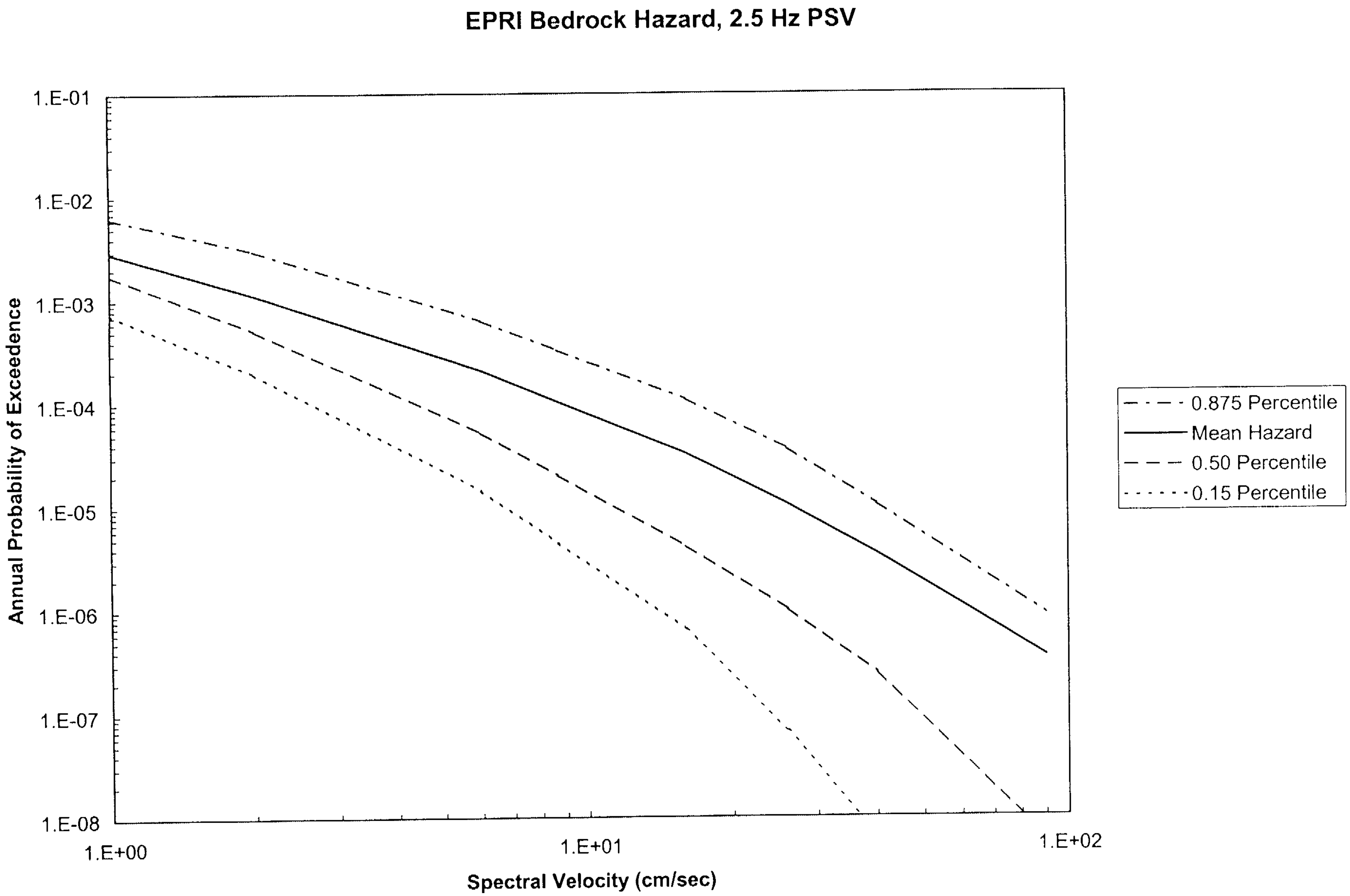


Figure RAI-012-1b. EPRI mean and fractile hazard for SRS hard-rock site conditions for a 2.5-Hz oscillator frequency.

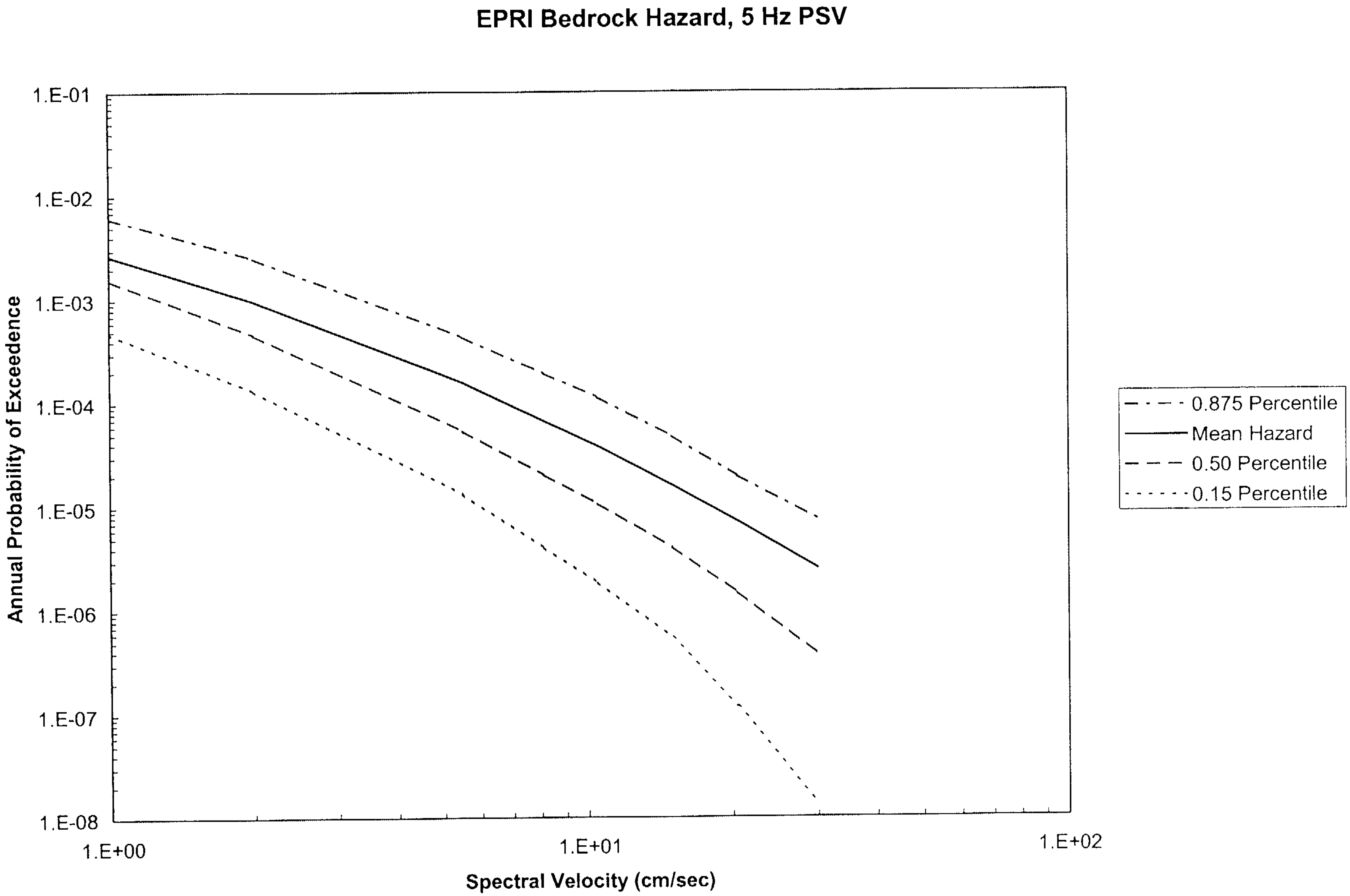


Figure RAI-012-1c. EPRI mean and fractile hazard for SRS hard-rock site conditions for a 5-Hz oscillator frequency.

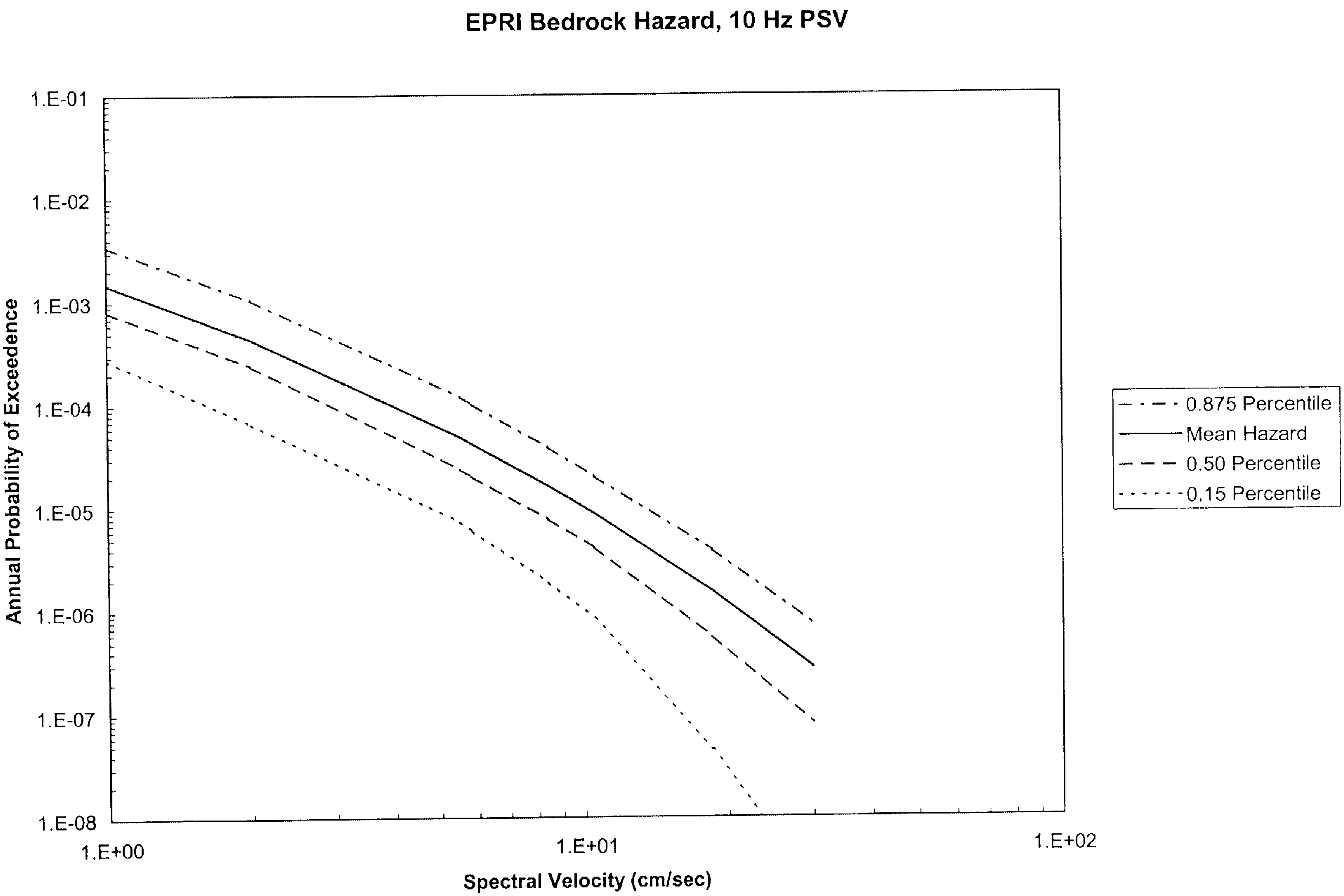


Figure RAI-012-1d. EPRI mean and fractile hazard for SRS hard-rock site conditions for a 10-Hz oscillator frequency.

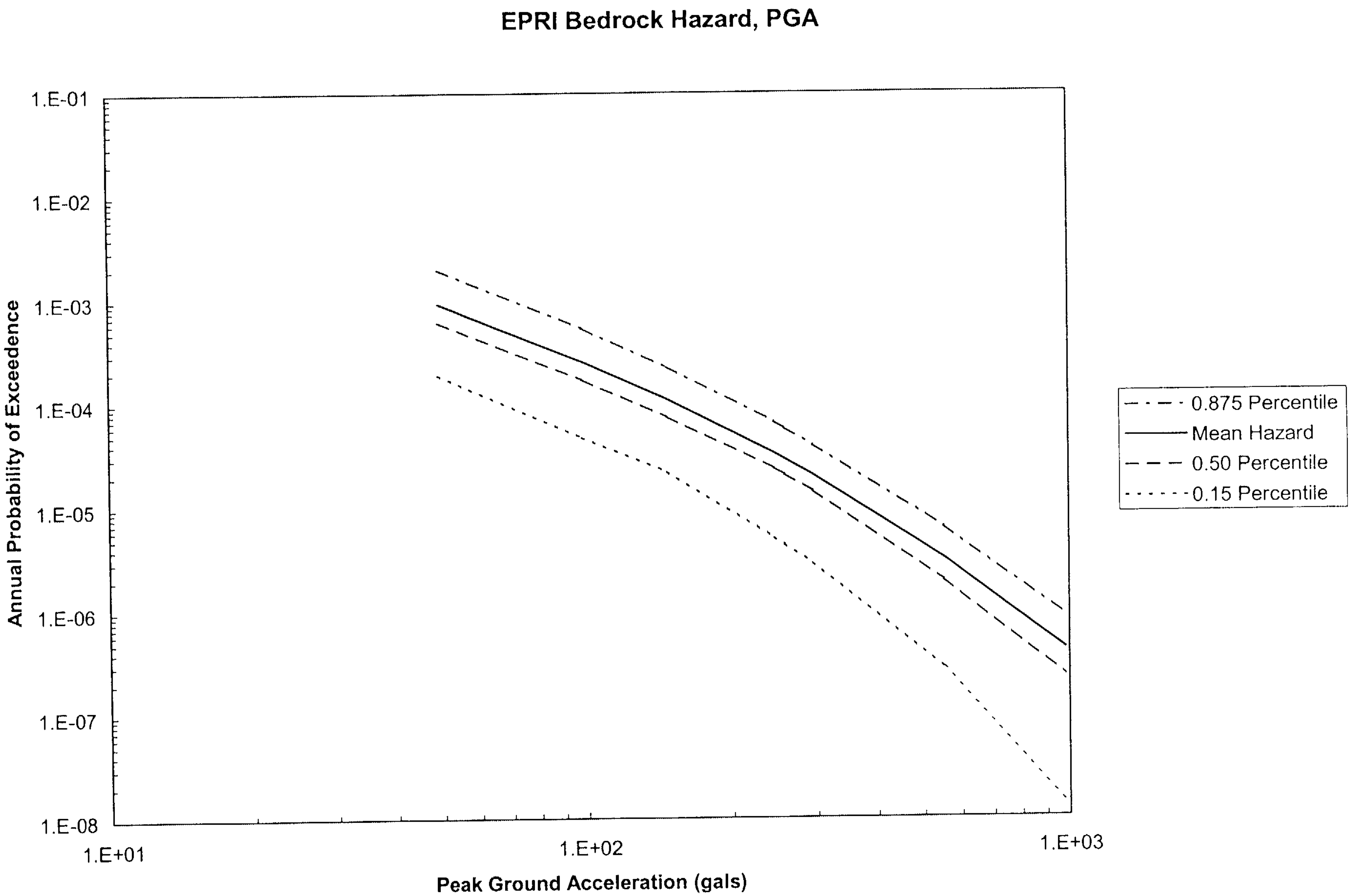


Figure RAI-012-1e. EPRI mean and fractile hazard for SRS hard-rock site conditions for PGA.

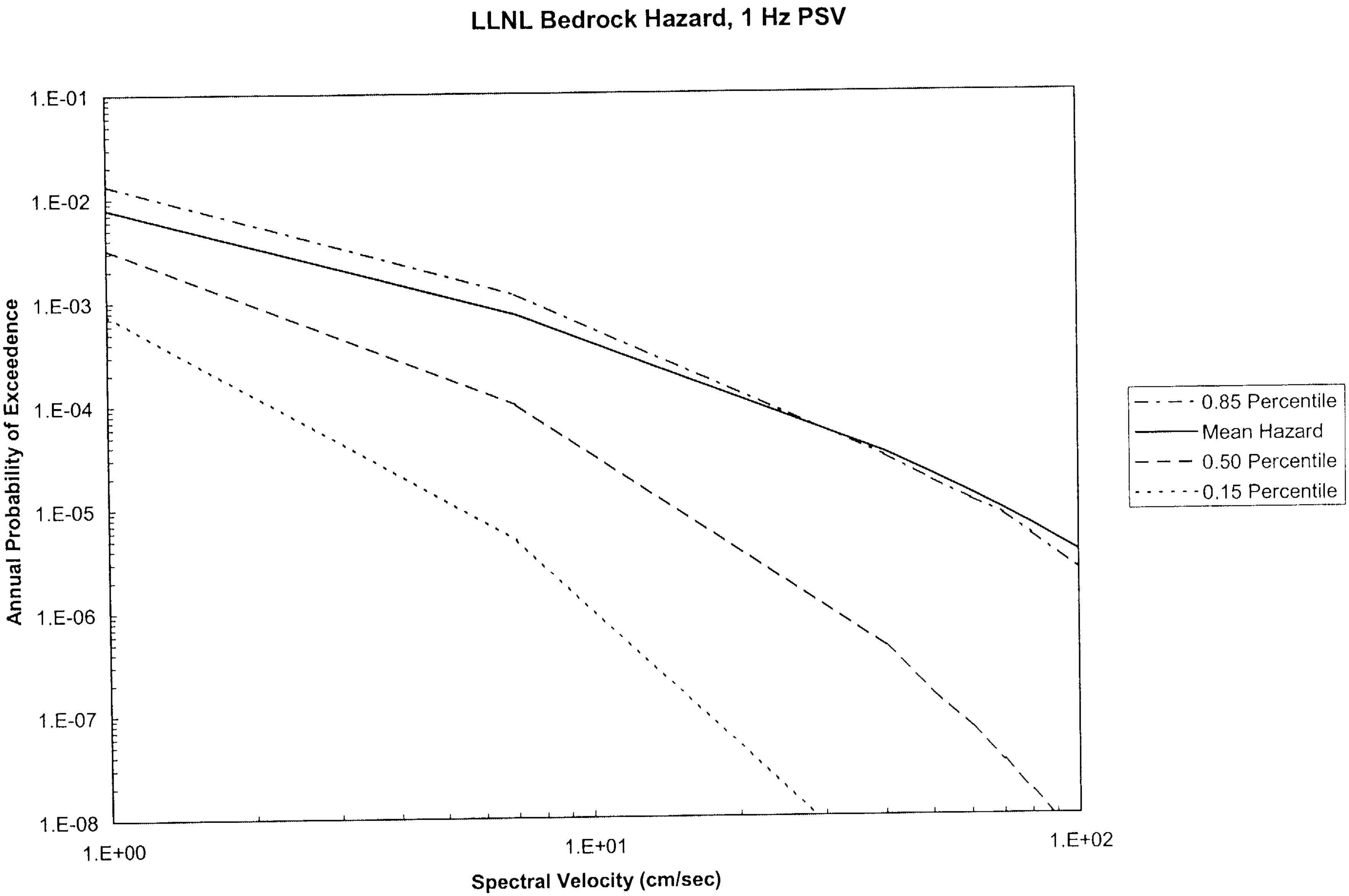


Figure RAI-012-2a. LLNL mean and fractile hazard for SRS hard-rock site conditions for a 1-Hz oscillator frequency.

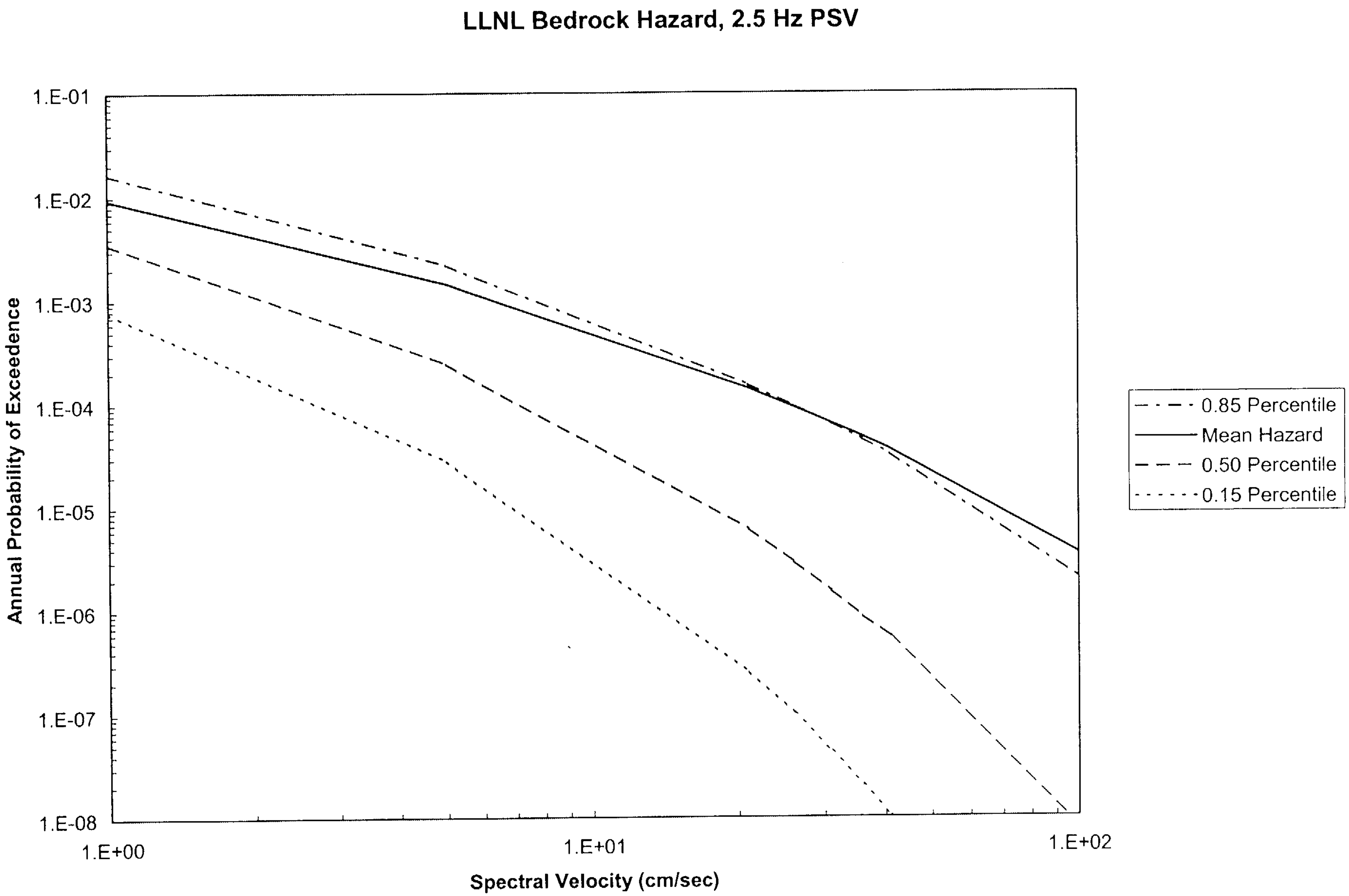


Figure RAI-012-2b. LLNL mean and fractile hazard for SRS hard-rock site conditions for a 2.5-Hz oscillator frequency.

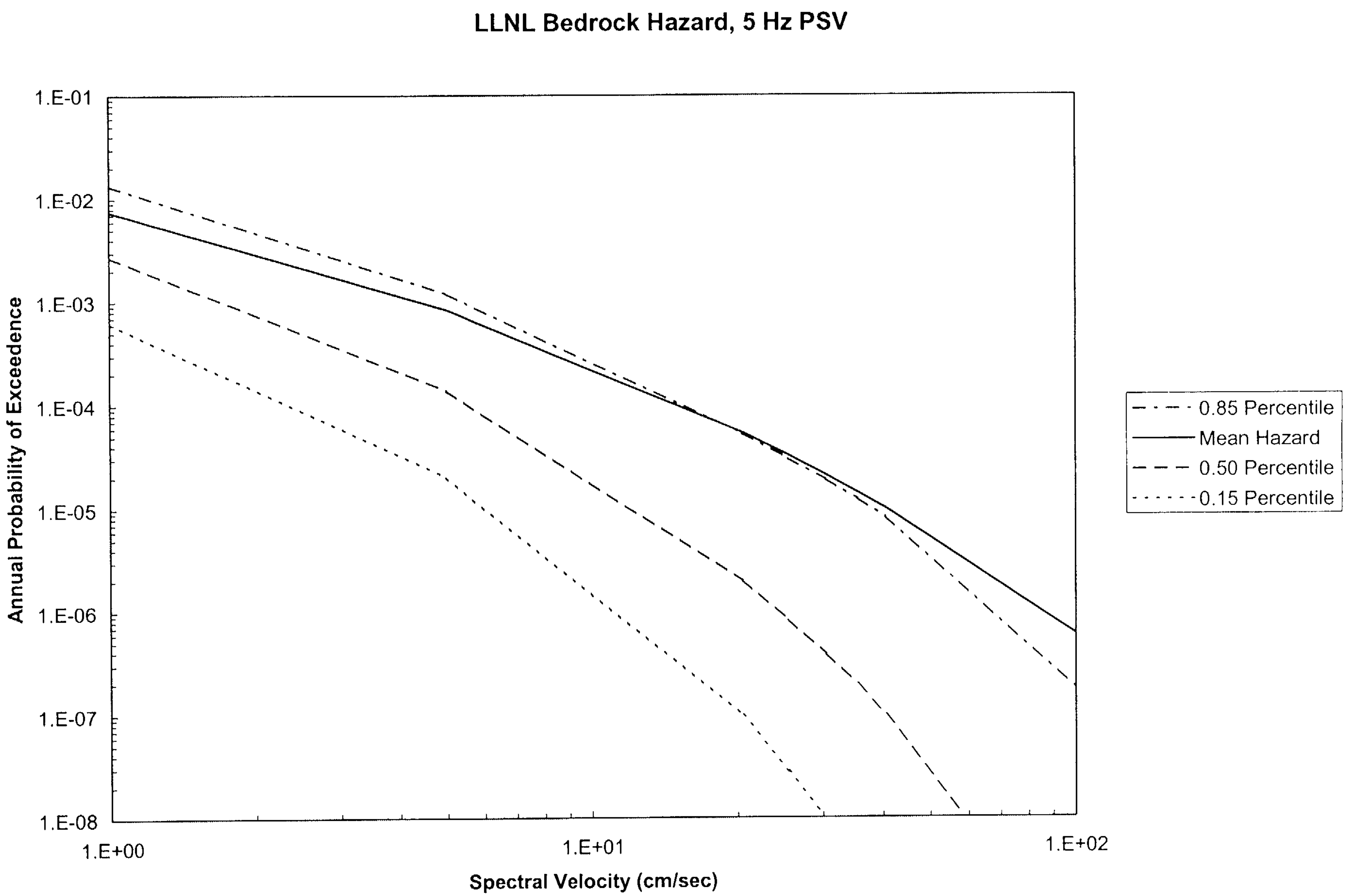


Figure RAI-012-2c. LLNL mean and fractile hazard for SRS hard-rock site conditions for a 5-Hz oscillator frequency.

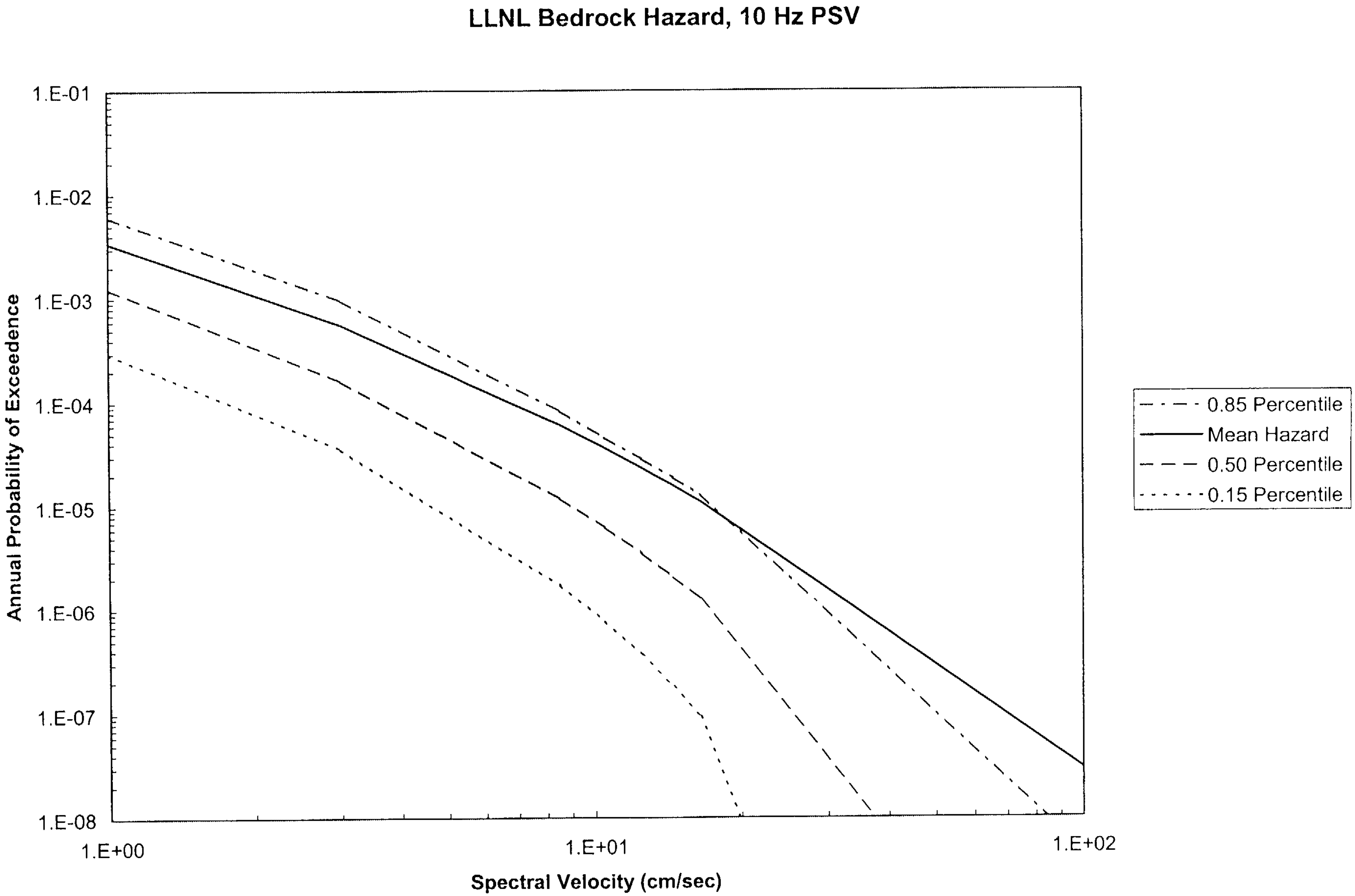


Figure RAI-012-2d. LLNL mean and fractile hazard for SRS hard-rock site conditions for a 10-Hz oscillator frequency.

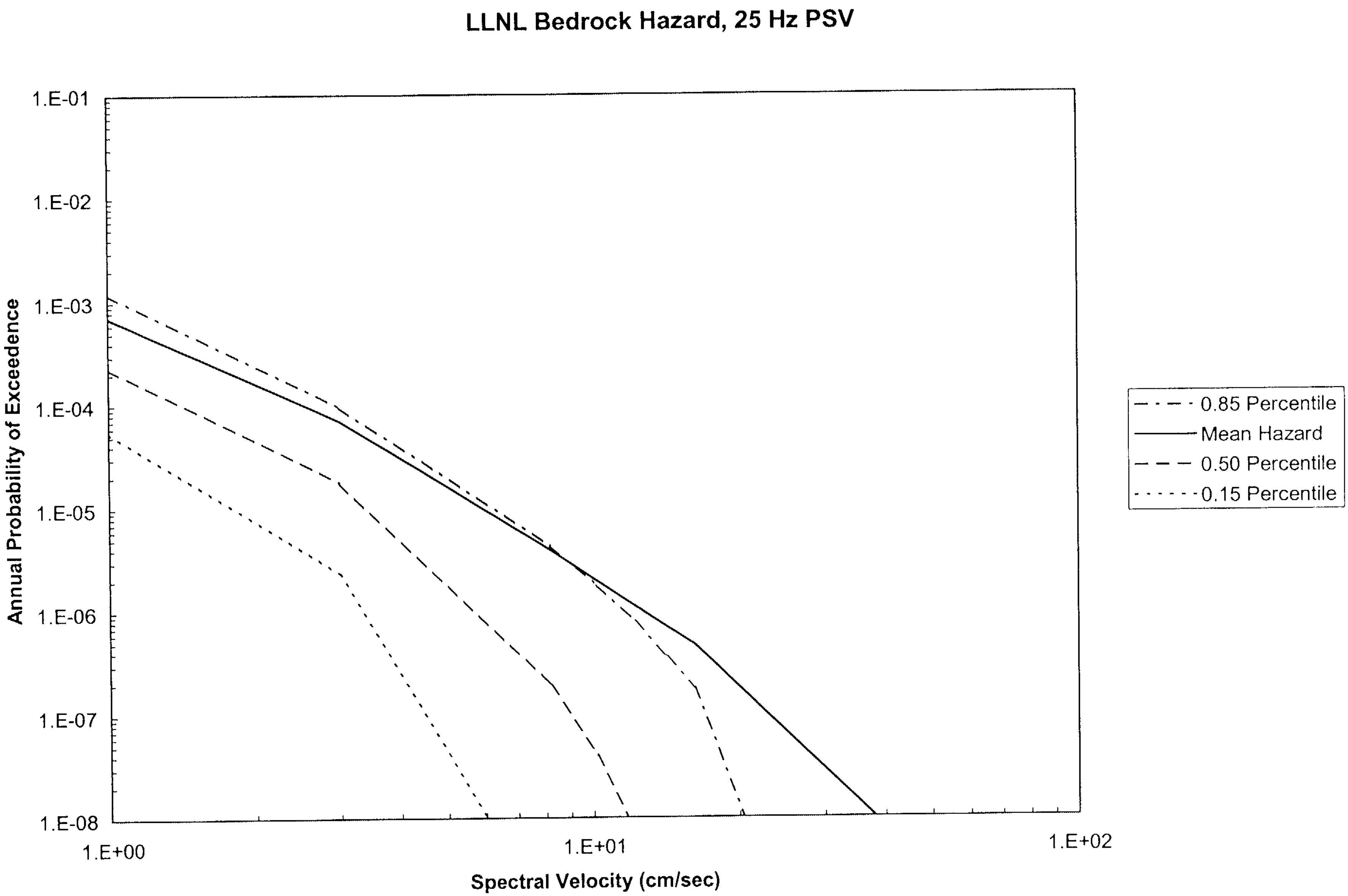


Figure RAI-012-2e. LLNL mean and fractile hazard for SRS hard-rock site conditions for a 25-Hz oscillator frequency.

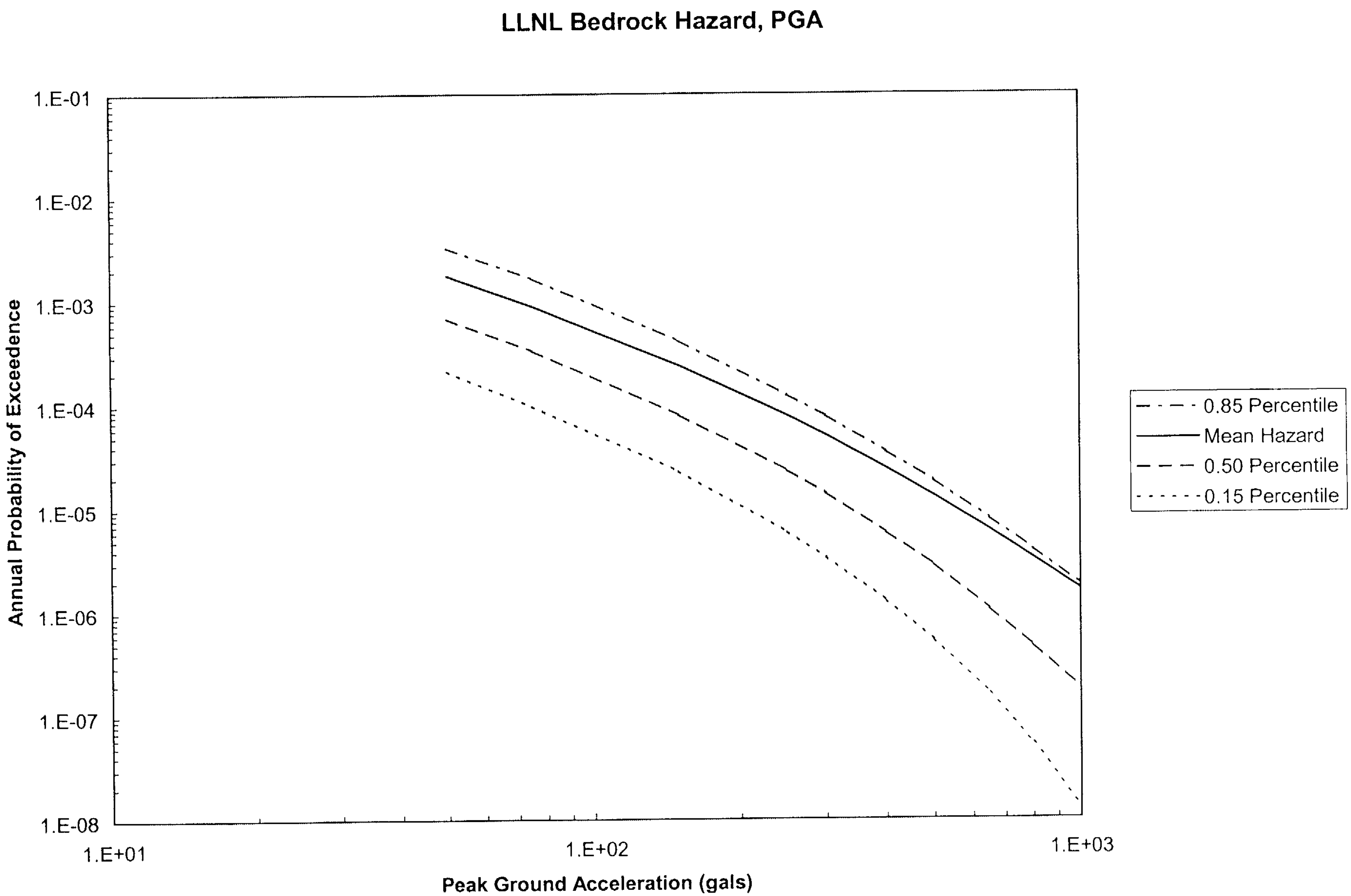


Figure RAI-012-2f. LLNL mean and fractile hazard for SRS hard-rock site conditions for PGA.

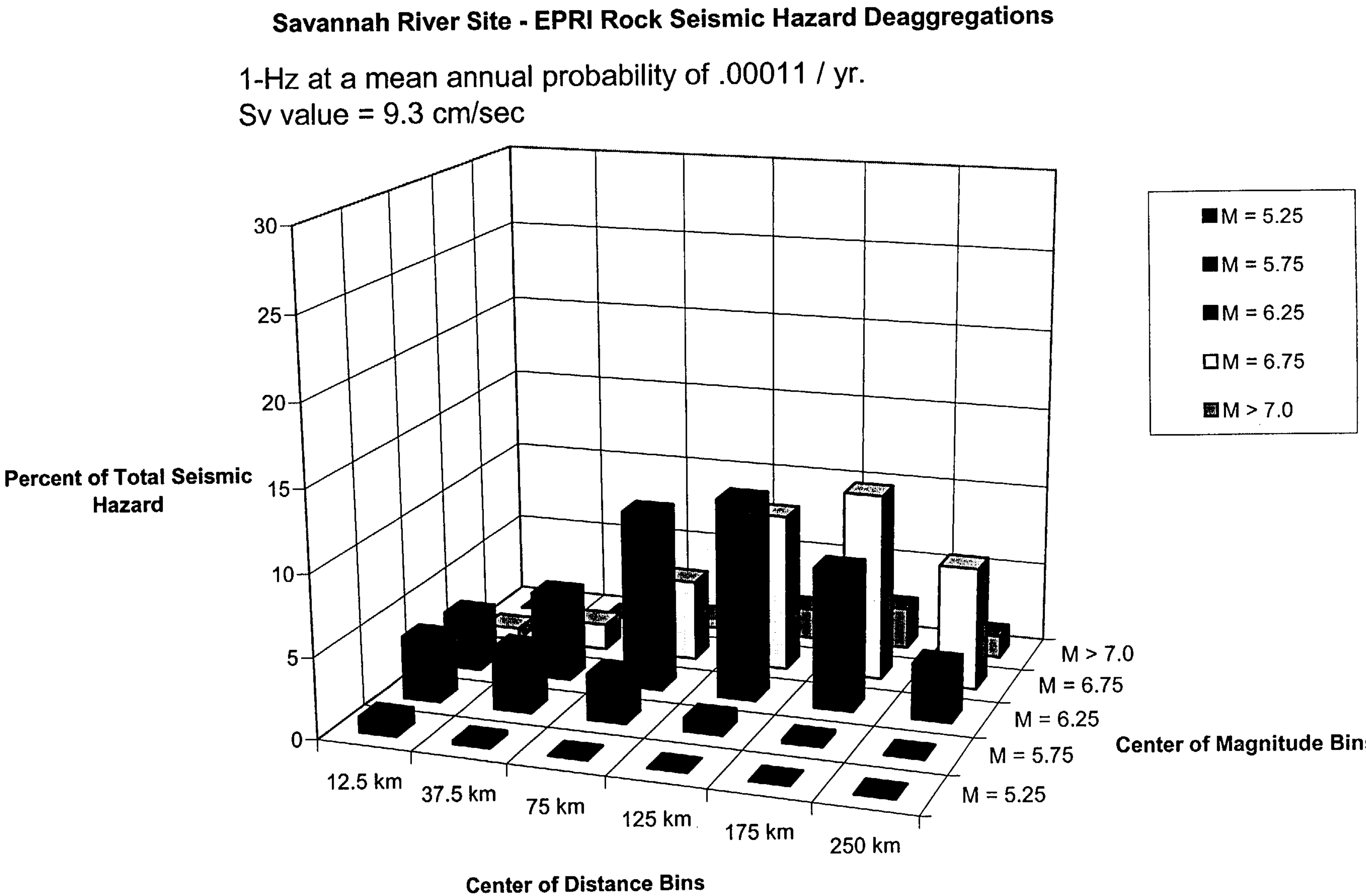


Figure RAI-012-3a. EPRI mean hazard deaggregation for SRS hard-rock site conditions for a 1-Hz oscillator frequency and annual probability of exceedance of approximately 10^{-4}.

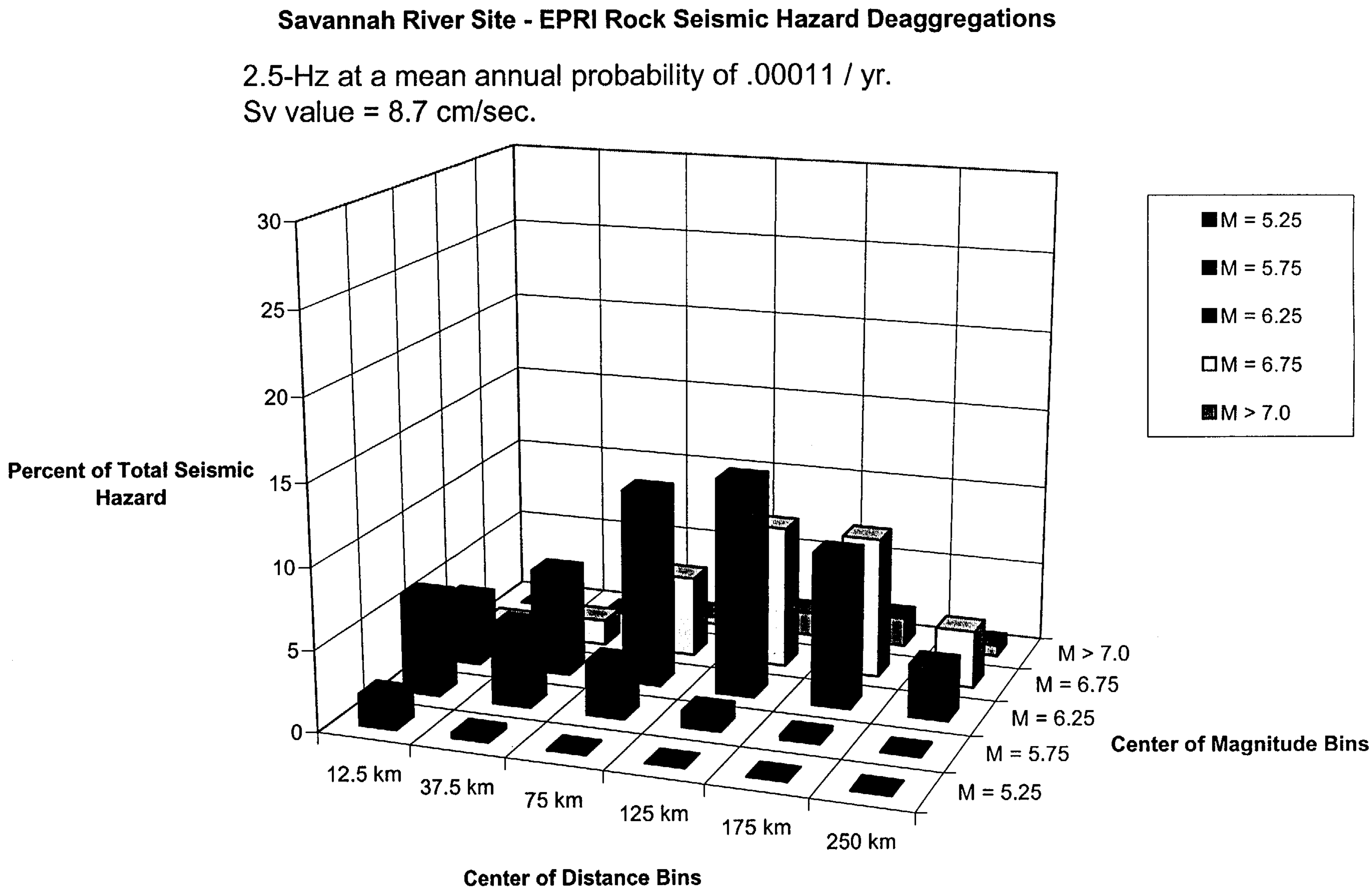


Figure RAI-012-3b. EPRI mean hazard deaggregation for SRS hard-rock site conditions for a 2.5-Hz oscillator frequency and annual probability of exceedance of approximately 10^{-4}.

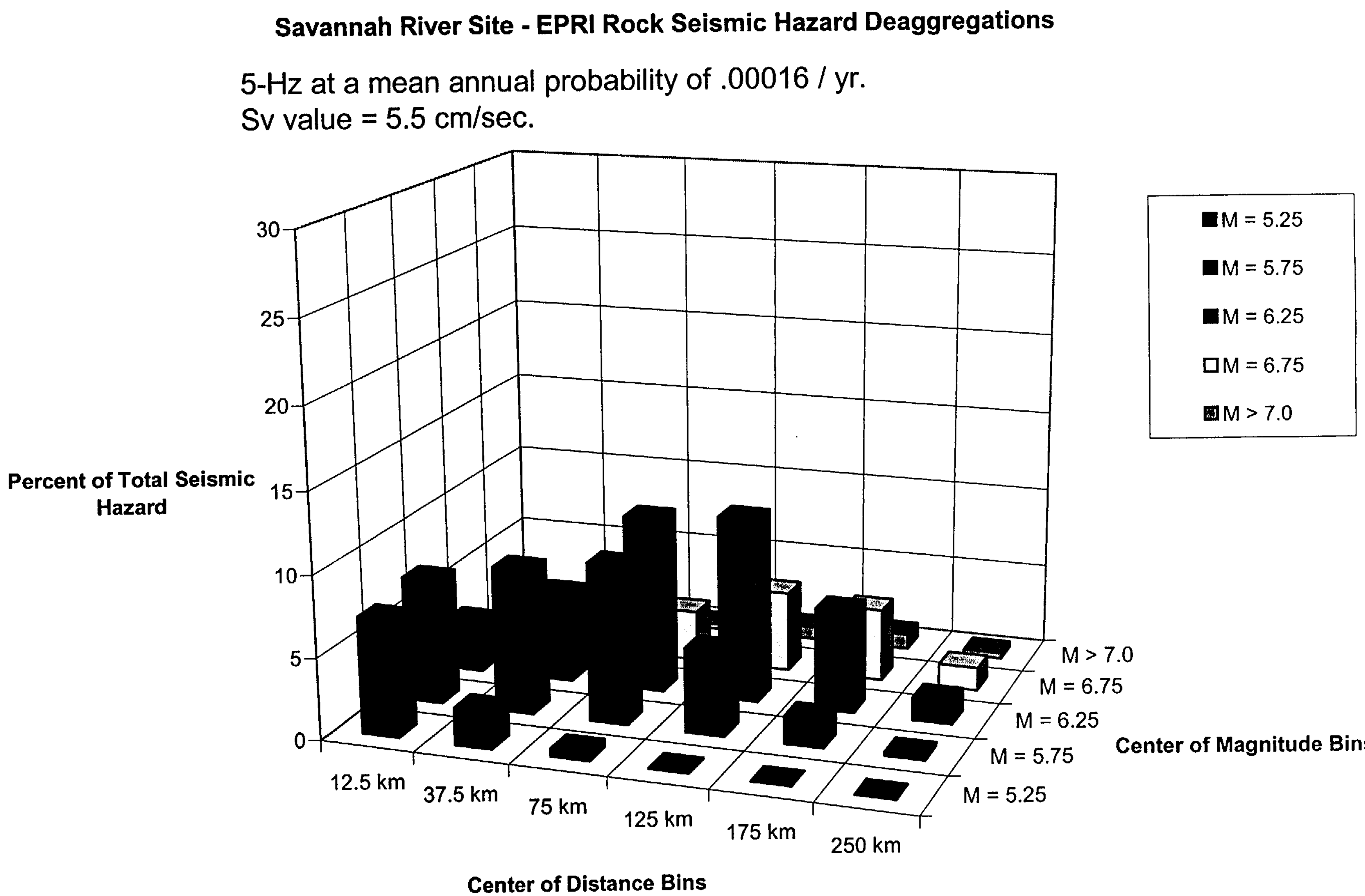


Figure RAI-012-3c. EPRI mean hazard deaggregation for SRS hard-rock site conditions for a 5-Hz oscillator frequency and annual probability of exceedance of approximately 10^{-4}.

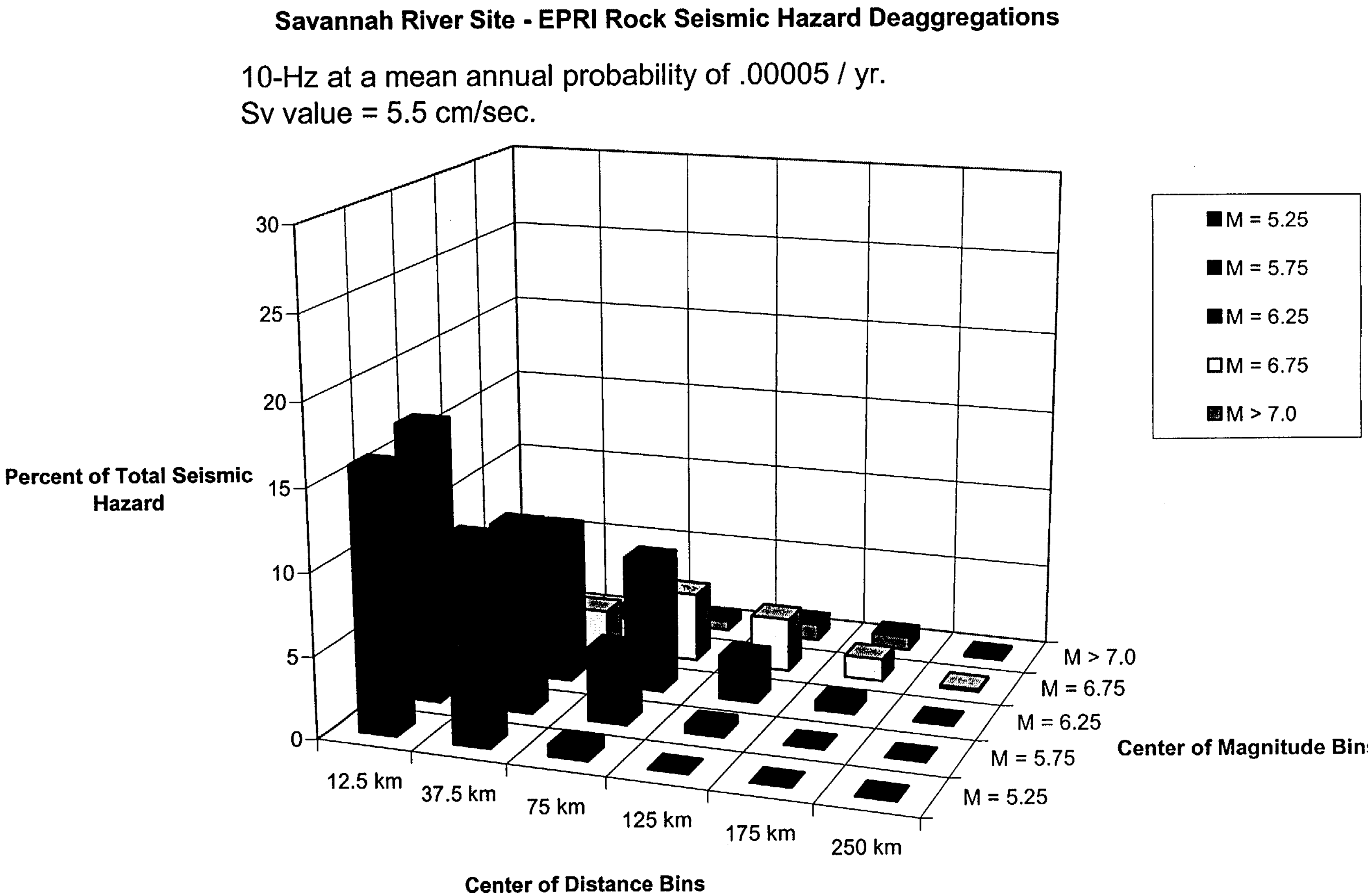


Figure RAI-012-3d. EPRI mean hazard deaggregation for SRS hard-rock site conditions for a 10-Hz oscillator frequency and annual probability of exceedance of approximately 10^{-4}.

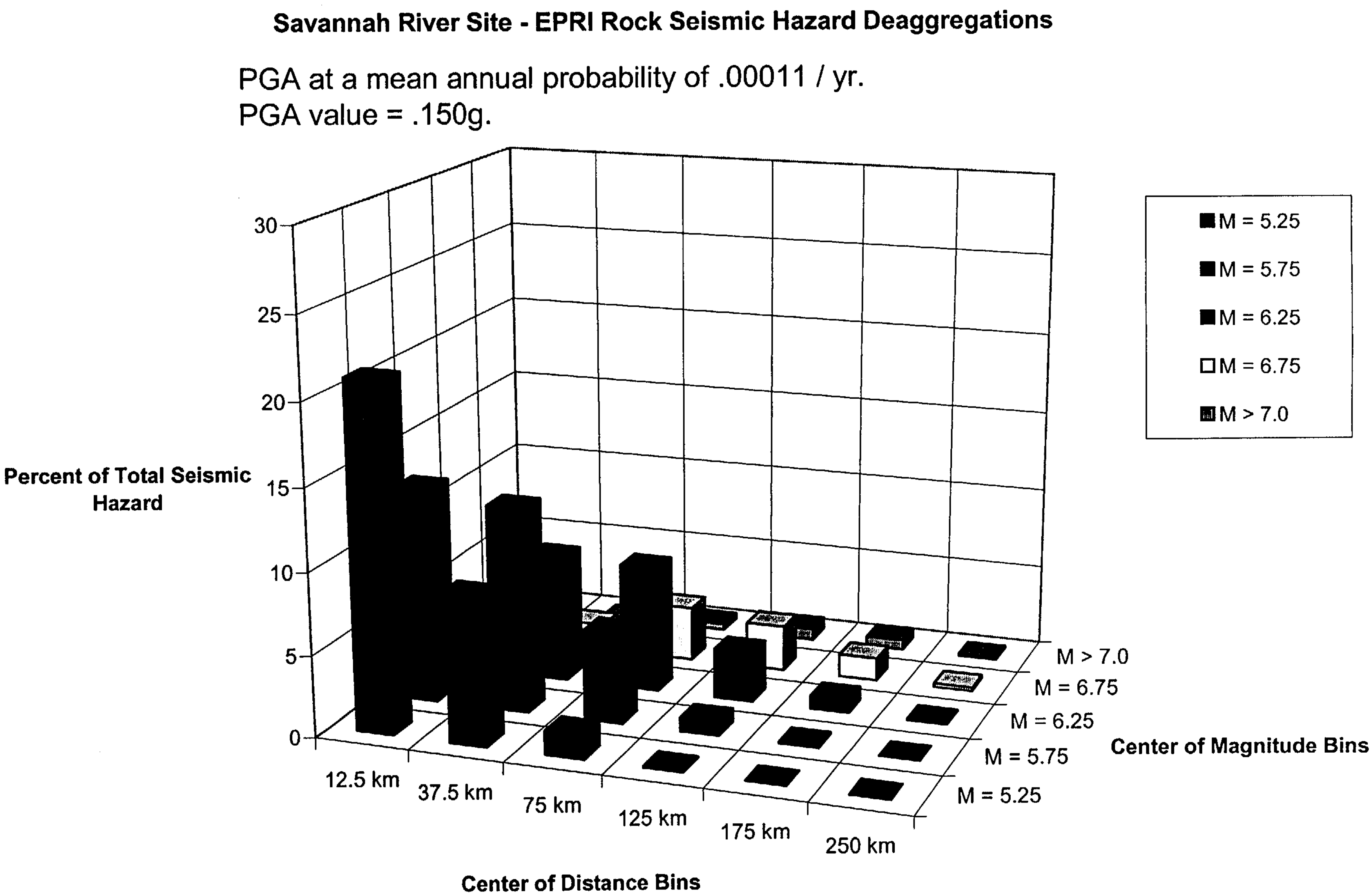


Figure RAI-012-3e. EPRI mean hazard deaggregation for SRS hard-rock site conditions for PGA and annual probability of exceedance of approximately 10^{-4}.

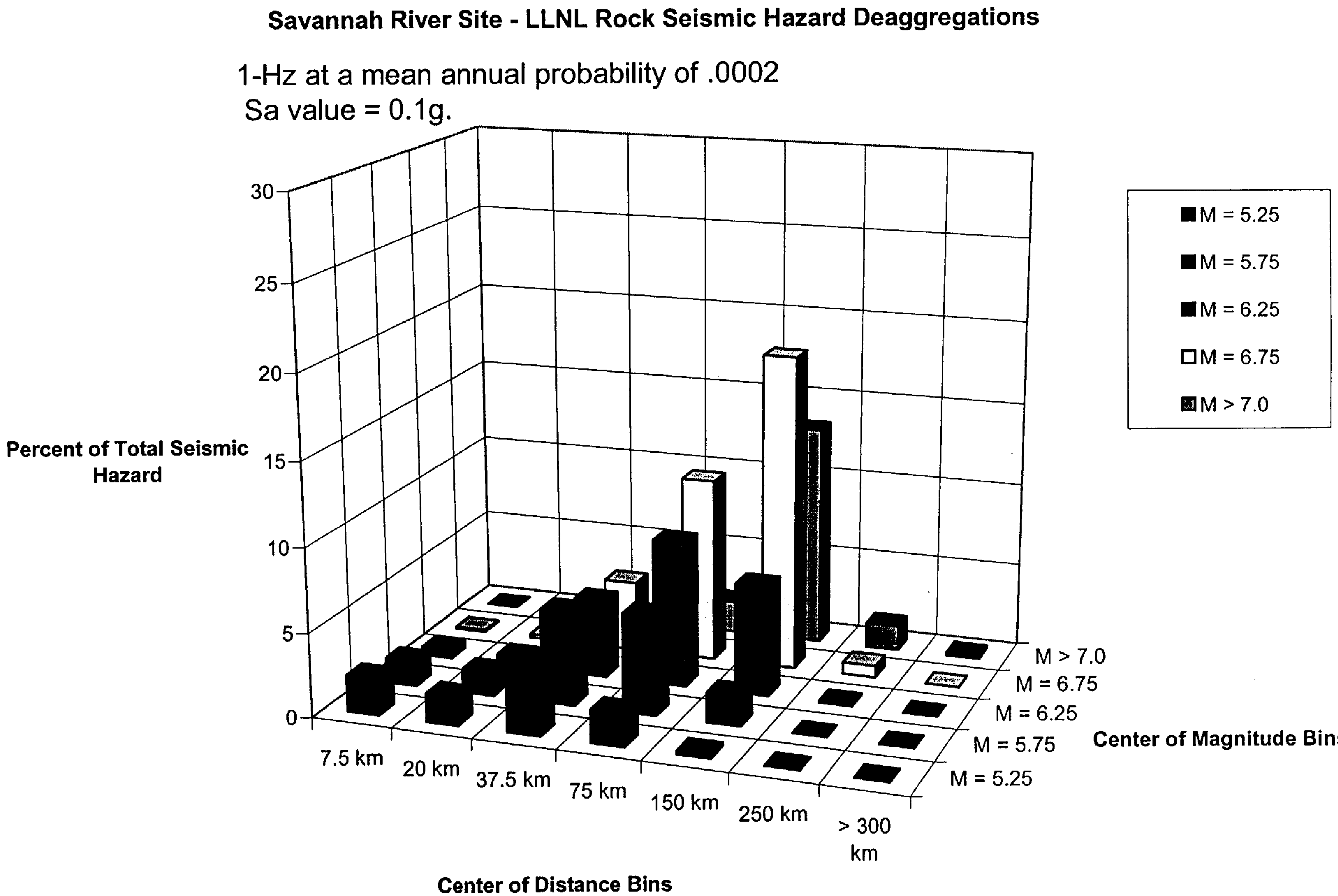


Figure RAI-012-4a. LLNL mean hazard deaggregation for SRS hard-rock site conditions for a 1-Hz oscillator frequency and annual probability of exceedance of approximately $2x10^{-4}$.

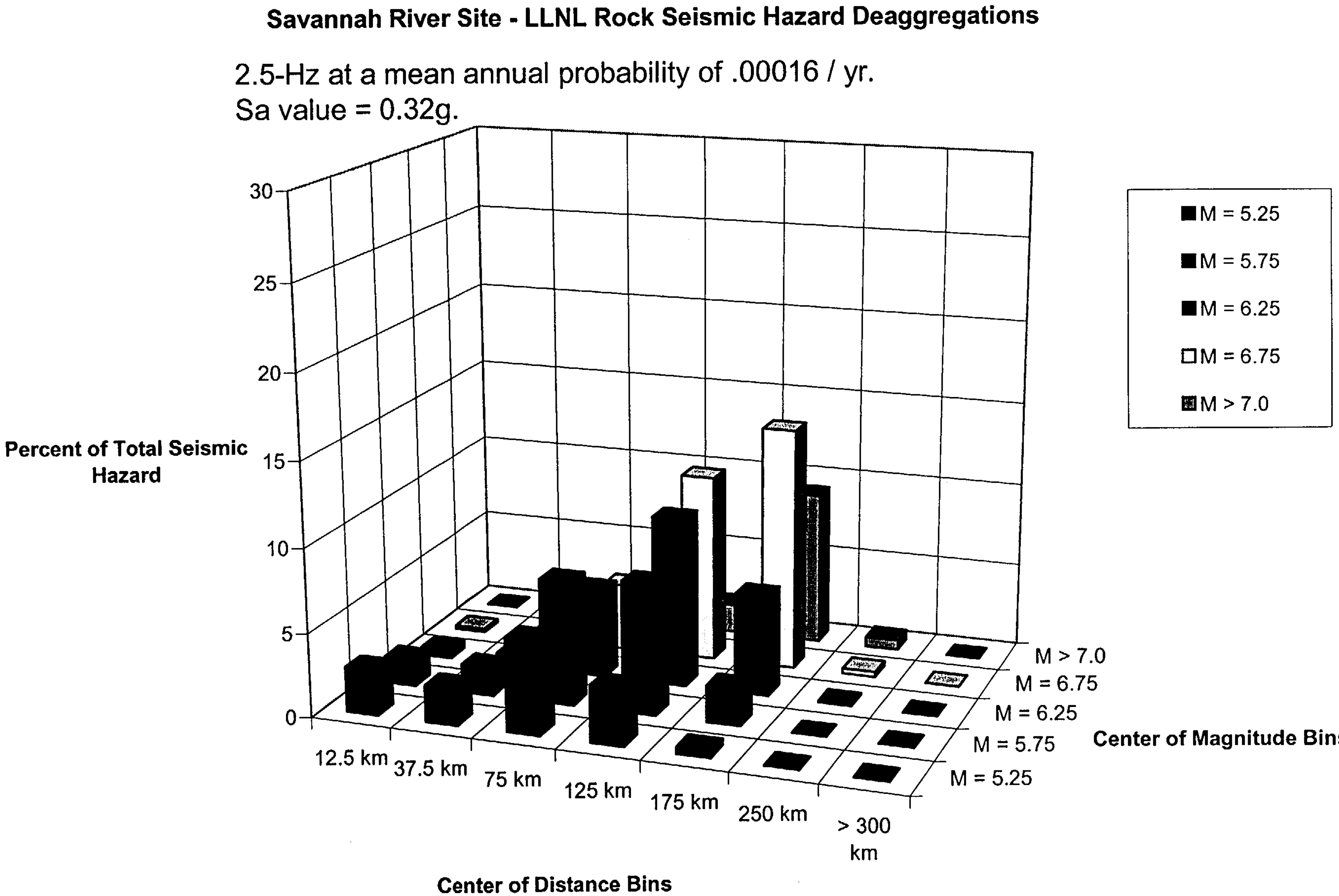


Figure RAI-012-4b. LLNL mean hazard deaggregation for SRS hard-rock site conditions for a 2.5-Hz oscillator frequency and annual probability of exceedance of approximately 2×10^{-4}.

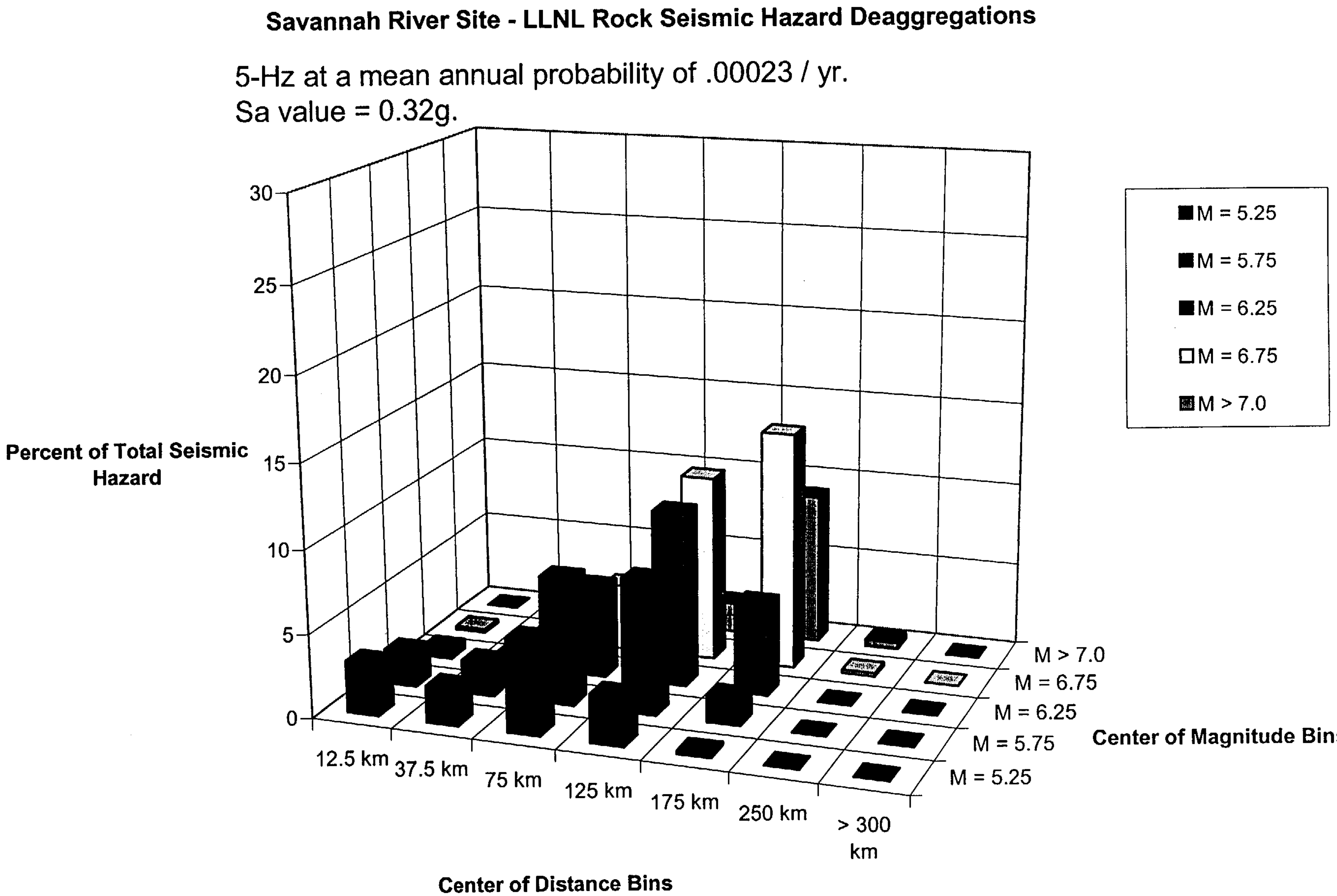


Figure RAI-012-4c. LLNL mean hazard deaggregation for SRS hard-rock site conditions for a 5-Hz oscillator frequency and annual probability of exceedance of approximately $2x10^{-4}$.

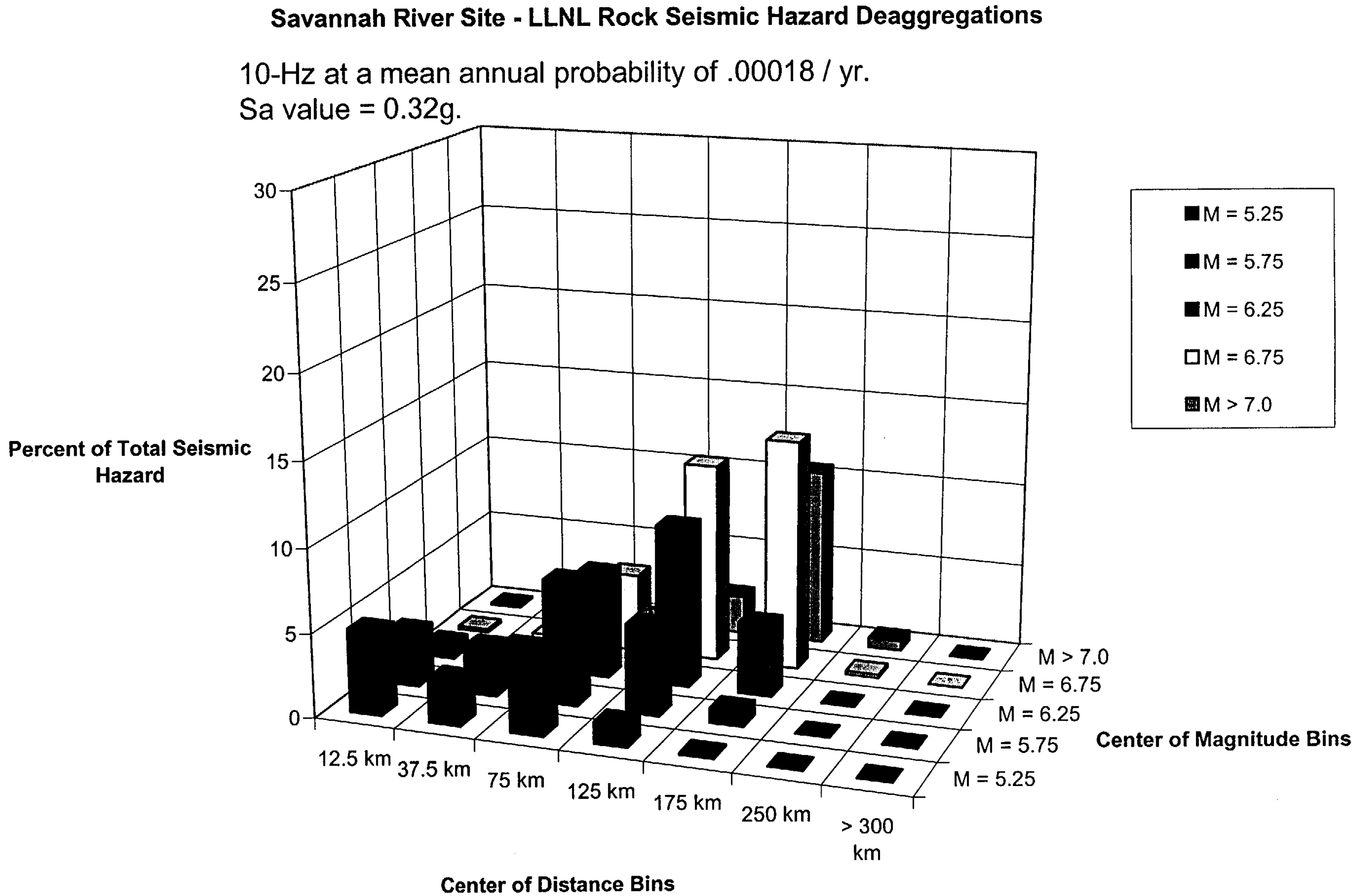


Figure RAI-012-4d. LLNL mean hazard deaggregation for SRS hard-rock site conditions for a 10-Hz oscillator frequency and annual probability of exceedance of approximately $2x10^{-4}$.

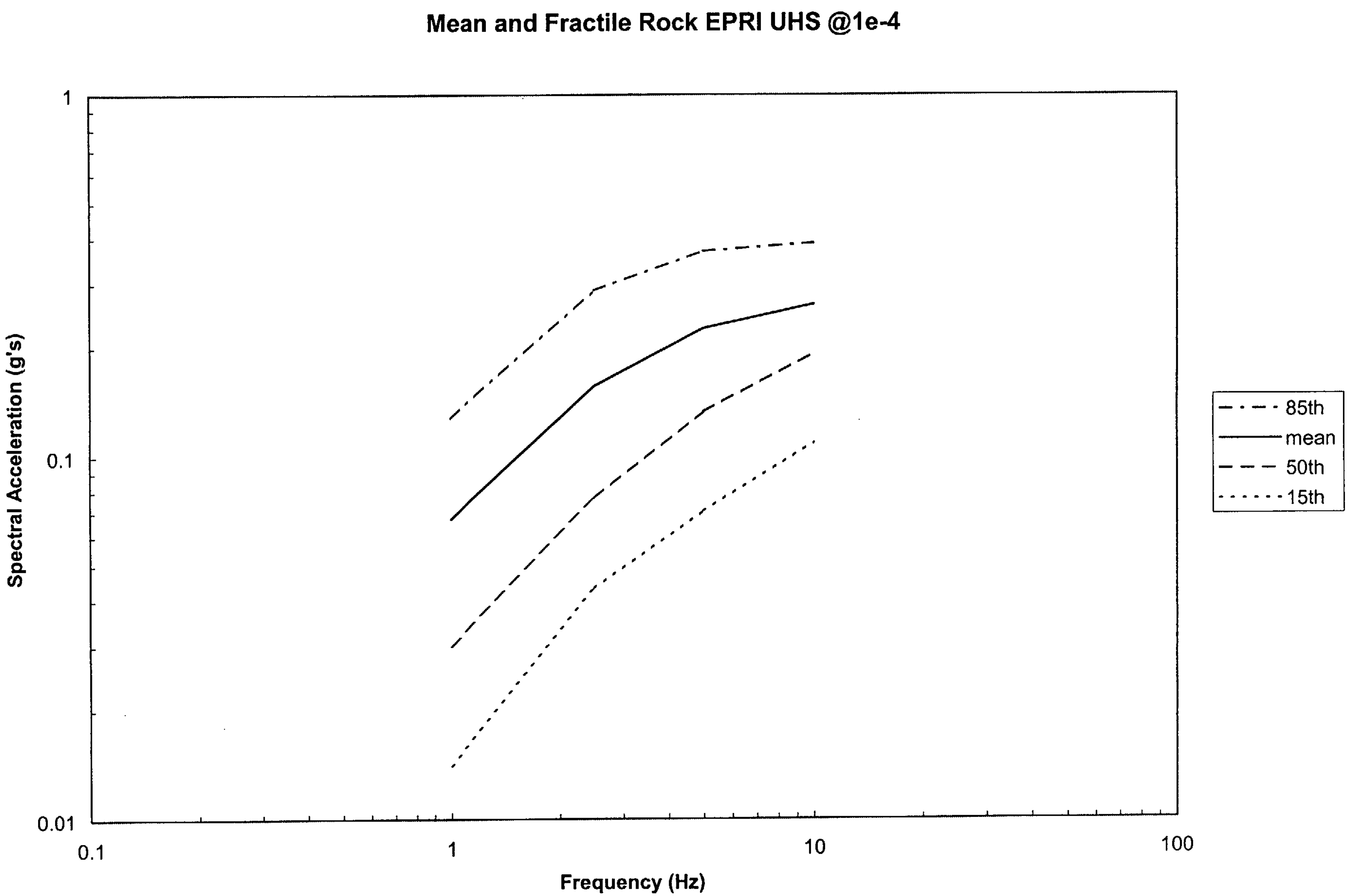


Figure RAI-012-5. EPRI mean and fractile uniform hazard spectra for SRS hard-rock site conditions for an approximate annual probability of exceedance of 10^{-4}.

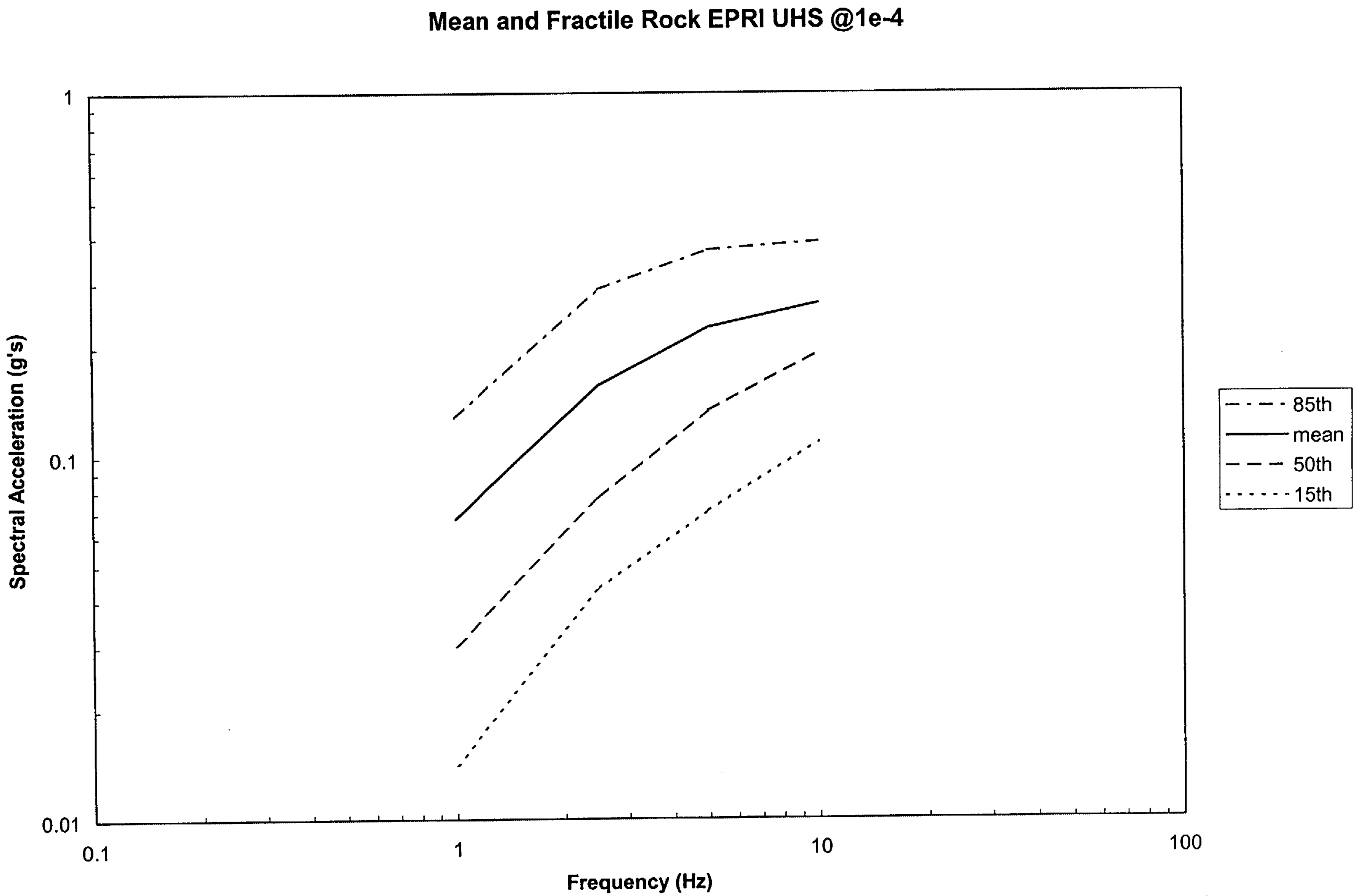


Figure RAI-012-5. EPRI mean and fractile uniform hazard spectra for SRS hard-rock site conditions for an approximate annual probability of exceedance of 10^{-4}.

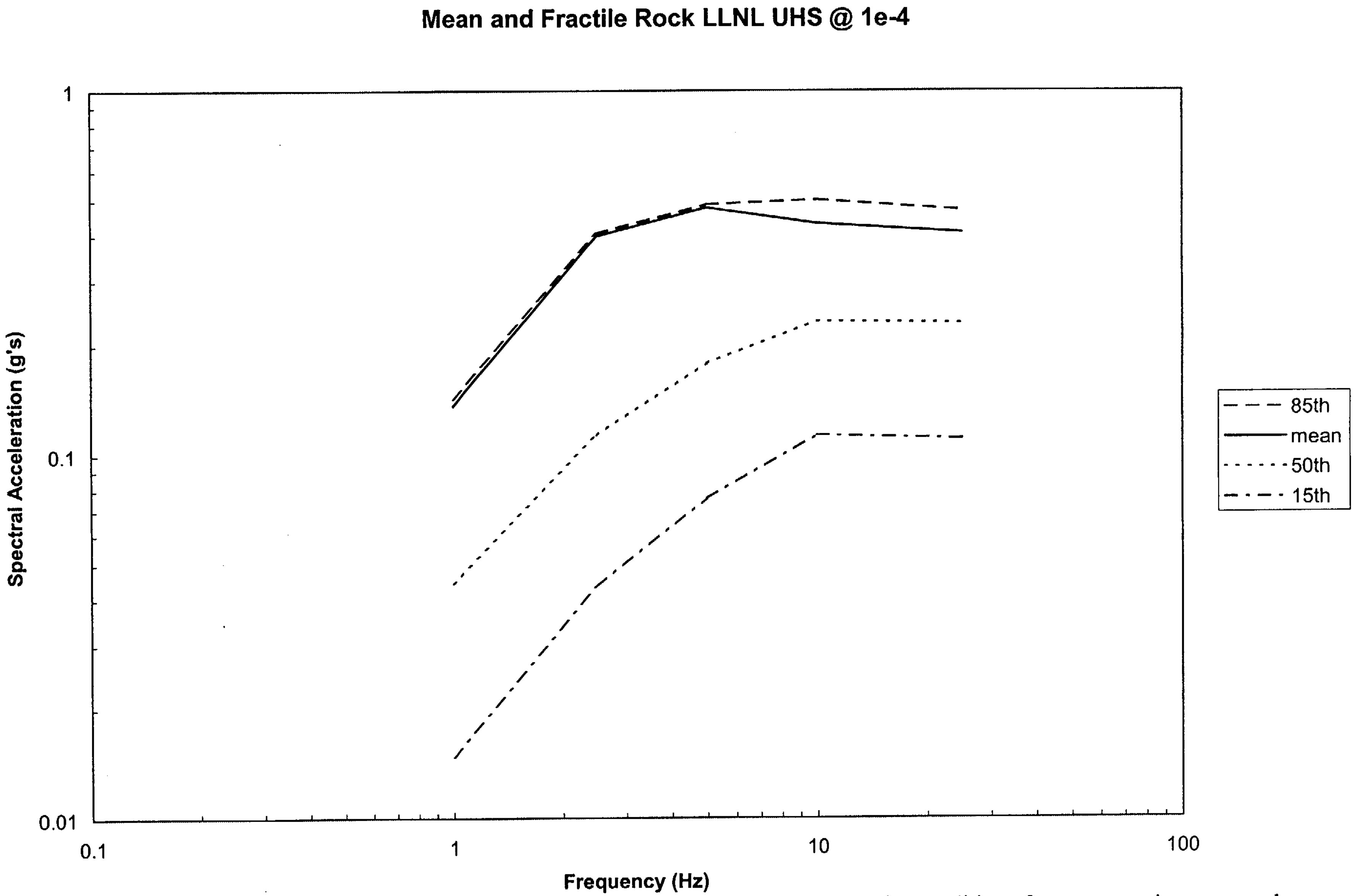


Figure RAI-012-6. LLNL mean and fractile uniform hazard spectra for SRS hard-rock site conditions for an approximate annual probability of exceedance of 10^{-4}.

13. <u>Section 1.3.6, pp. 1.3.6-1 thru 1.3.6-86</u>

Clarify whether updated PSHA has been or will be conducted for the MFFF site that accounts for soil properties derived from geological, geophysical, geotechnical, and seismic investigations that are specific to the MFFF site.

It appears that the applicant did not conduct an updated PSHA for the MFFF site that accounts for soil property data that is specific to the MFFF site. Instead, the existing Lawrence Livermore National Laboratory (LLNL) and Electric Power Research Institute (EPRI) bedrock hazard results were used to evaluate surface uniform hazard spectra and design basis spectra. However, on page 1.3.6-18, the application states, "The PSHA for MFFF <u>will account for</u> soil properties derived from site geological, geophysical, geotechnical, and seismic investigations." (underline added for emphasis). This statement is misleading.

Response:

The development of the PSHA for SRS and its applicability to the MFFF site is discussed in the responses to Questions 12, 14, 15, and 17. On 10 August 2001, DCS submitted DCS01-WRS-DS-NTE-G-00005-C, *MOX Fuel Fabrication Facility Site Geotechnical Report* for NRC review. This report contains a detailed presentation of the site investigations that have been conducted and the results of site-specific analyses. Section 3.4 of this document also demonstrates the applicability of the SRS site generic PSHA to the MFFF site. Section 5.0 presents subsurface conditions at the MFFF site and demonstrates that they are consistent to subsurface conditions that exist at the adjacent F-Area at SRS. Section 6.2 presents dynamic properties for the subsurface soils and the one-dimensional free-field response analyses for the MFFF site. Consequently, PSHA specific to the MFFF site is not required.

Action:

The second sentence of the first paragraph in Section 1.3.6.3.6.1 will be revised to make this clarification in the next update of the CAR.

14. Section 1.3.6, pp. 1.3.6-1 thru 1.3.6-86

Demonstrate that the current site-wide hazard results and Performance Category PC-3 and PC-4 spectra are appropriate for the MFFF site.

The applicant did not justify that the current Savannah River Site site-wide PC-3 and PC-4 spectra are appropriate for the MFFF site. As indicated in Lee, et al. (1997), the site-wide PC-3 and PC-4 spectra were issued as "committed" because of the potential for stratigraphic variation in excess of what was considered and these design spectra need to be "confirmed" using soil parameters at the specific site or facility where the design spectra are being used. Position 4 of Regulatory Guide 1.165 (U.S. Regulatory Commission, 1997b) and other guidance documents (such as U.S. Regulatory Commission, 1997a,c) recommend site-specific soil amplification analysis for seismic design of non-rock sites.

Response:

In the development of the PC-3 and PC-4 design spectra, the available SRS soils and bedrock data reported on or before 1996 were used (WSRC, 1997). Based on the WSRC (1997) analysis of the SRS data, including the range of soil column thicknesses used in the analysis, the bedrock categories and the consistency of the area-to-area shallow soil velocity distribution, it was expected that future new sites (i.e., for possible use for other new facilities) could be evaluated for their use of the SRS design spectra. To validate the applicability of PC-3 and PC-4 design spectra for SRS sites, a confirmation study is performed to verify that the new site-specific data is consistent with the input used to determine the design spectra.

There are four general areas (i.e., stratigraphic conditions) used to validate the suitability of the site-wide response spectra (SWRS) for the MFFF site: (1) the stratigraphy of soils to validate that there are no topographic or subsurface features that could significantly alter ground motion over the modeled cases; (2) validate that the soil column thickness and bedrock type matches one of the ranges used in developing the SWRS; (3) validate that the velocity profiles measured at the site are within the variances used in developing the SWRS; and (4) validate that the geologic formations at the site are reasonably close to the base case formations used for the SWRS so that there is a consistent relationship between the dynamic properties applied in the SWRS and the new site.

For the MFFF site review, the seismic shear-wave velocity data were obtained from DCS (2001). The 15 P- and S-wave seismic cone penetrometer test (SCPT) velocity interpretations from DCS (2001) were compared to the SRS-wide velocity profile distribution. Other available geotechnical data and subsurface geotechnical interpretations for the MFFF site were also obtained from DCS (2001). The seismic velocity data, laboratory measured strain-dependent shear modulus and damping, geologic formation descriptions, and geotechnically inferred engineering cross-sections were reviewed and found to be reasonably consistent with the site properties used in the development of PC-3 and PC-4 SRS design spectra. Further information concerning the review of the MFFF site data for application of the PC-3 and PC-4 design spectra is given in WSRC (2001).

Similarly, on 10 August 2001, DCS submitted DCS01-WRS-DS-NTE-G-00005-C, *MOX Fuel Fabrication Facility Site Geotechnical Report* for NRC review. This report contains a detailed presentation of the site investigations that have been conducted and site-specific analyses. Section 3.4 of this document also clearly demonstrates the applicability of the SRS site generic PSHA to the MFFF site.

The conclusion of these two reports is that examination of general stratigraphic conditions validate the suitability of the site-wide response spectra for the MFFF site. There are no topographic or subsurface features that could significantly alter ground motion over the modeled cases, with the exception of the APSF spoils pile, which will be removed before construction. The soil column thicknesses and bedrock type match ranges used in developing the site-wide criteria. The velocity profiles measured at the MFFF site are within the variances used in developing the site-wide criteria. The formations at the MFFF site, while somewhat different from the assumed formation model used for the SWRS, would not lead to any significant bias in a predicted ground motion model for the MFFF. The variability used in developing the SWRS encompasses the dynamic properties expected at the MFFF. These two analyses have confirmed that the site-wide "committed" criteria are "confirmed" to be applicable to the MFFF site.

Additional information regarding the seismic hazard for the SRS is provided in the responses to Questions 12, 13, and 19.

<u>References:</u>

DCS (2001). *MOX Fuel Fabrication Facility Site Geotechnical Report*, QL-1A (IROFS), DCS01-WRS-DS-NTE-G-00005-C

WSRC (1997). *SRS Seismic Response Analysis and Design Basis Guidelines*, WSRC-TR-97-0085, Rev. 0.

WSRC (2001). *Applicability of SRS Site-wide Spectra to the MFFF Site*, WSRC Calculation No. K-CLC-F-00049, Rev. 0

Action:

None

15. Section 1.3.6, pp. 1.3.6-1 thru 1.3.6-86

Provide soil property data specific to the MFFF site and compare the MFFF soil parameters with those used to derive site-wide design spectra for Performance Categories PC-3 and PC-4 documented in Lee, et al. (1997) and for PC-1 and PC-2 documented in Lee (1998).

The applicant did not provide soil property data specific to the MFFF site. These data are necessary to demonstrate that the Savannah River Site site-wide design spectra could be "confirmed" (see Question 14) and used as bases for MFFF seismic design. The application gives conflicting information with regard to the validation of Savannah River Site site-wide soil parameters for the MFFF site. For example, on page 1.3.6-18, the application states, "The PSHA for MFFF will account for soil properties derived from site geological, geophysical, geotechnical, and seismic investigations." (underline added for emphasis). This sentence implies that the site geological, geophysical, geotechnical, and seismic investigations were not yet completed nor incorporated into seismic design considerations. However, on page 1.3.6-17, the application states, "The soil parameters for the MFFF site have been checked for consistency with the data parameterized in the study, and the spectra have been confirmed to be applicable to the MFFF." Position 4 of Regulatory Guide 1.165 (U.S. Regulatory Commission, 1997b) and other guidance documents (such as U.S. Regulatory Commission, 1997a,c) recommend site-specific soil amplification analysis for seismic design of non-rock sites and site-specific soil amplification analysis requires site-specific soil property data.

Response:

As indicated in the responses to Questions 13 and 14, on 10 August 2001, DCS submitted DCS01-WRS-DS-NTE-G-00005-C, *MOX Fuel Fabrication Facility Site Geotechnical Report* for NRC review. This report contains a detailed presentation of the site investigations that have been conducted and site-specific analyses. Section 3.4 of this report demonstrates the applicability of the SRS site generic PSHA to the MFFF site. Similarly, independent analyses by WSRC have confirmed that the site-wide "committed" criteria are "confirmed" to be applicable to the MFFF site. These reviews are discussed in the responses to Questions 13 and 14. Sections 5 and 6 of this report present the MFFF site subsurface conditions and engineering properties for the MFFF site, respectively. The analysis of the site-specific subsurface conditions at the MFFF site "confirms" that they are consistent with development of SRS site-wide design spectra and that these can be used as design bases for MFF seismic design.

Action:

DCS will update the CAR text to reconcile differences.

16. Section 1.3.6, pp. 1.3.6-1 thru 1.3.6-86

Provide references that document the modification of the Savannah River Site site-wide Performance Category PC-3 spectrum developed in Lee, et al. (1997) to that used in Westinghouse Savannah River Company (WSRC) Engineering Standard 01060 (WSRC 1999a). Provide the memorandum entitled, "Revised Envelope of the Site Specific PC-3 Surface Ground Motion," from Brent Gutierrez to Lawrence Salonmone [*sic* - Salomone] and Fred Loceff, dated September 9, 1999.

The applicant did not document or give references on how and why the PC-3 spectrum developed in Lee, et al. (1997) was modified. The application only stated (page 1.3.6-22) that "Following the development of PC-3 and PC-4 design basis spectra and the PC-1 and PC-2 design basis spectra, additional conservatism was applied to the PC-3 spectral shape at high and intermediate frequencies."

Response:

The requested memo is provided together with other requested documents in the response to Question 29.

Action:

None

17. <u>Section 1.3.6, pp. 1.3.6-1 thru 1.3.6-86</u>

Demonstrate that the selected MFFF design spectrum envelopes the uniform hazard spectra at the soil surface that is specific to the MFFF site.

The applicant did not demonstrate that the selected design spectrum envelopes the surface uniform hazard spectra specific to the MFFF site at the selected return periods. Instead, the applicant showed that the selected design spectrum envelops the Savannah River Site site-wide PC-3 spectrum. Position 4 of Regulatory Guide 1.165 (U.S. Regulatory Commission, 1997b) and Section 2.5.2.6 (Safe Shutdown Earthquake Ground Motion) of NUREG-0800 (U.S. Regulatory Commission, 1997c) recommend that the development of design spectra be site-specific.

Response:

The response to Questions 14 and 15 demonstrate that the MFFF-specific geotechnical data are consistent with the SRS-specific data used to develop the PC-3 and PC-4 design spectra. The response to Question 14 presents the conclusion that the application of the PC-3 and PC-4 design spectrum is appropriate and "confirmed" for the MFFF site in accordance with WSRC (1997). Therefore, based on the site-specific MFFF geotechnical data (DCS, 2001), the SRS PC-3 and PC-4 design spectra are also MFFF site-specific. The PC-3 and PC-4 design spectra are conservative spectra with probabilities of exceedance of $5x10^{-4}$/yr and 10^{-4}/yr respectively, based on evaluation of SRS-specific soil surface hazard curves (WSRC, 1997, 1998). Because the PC-3 design spectrum is also MFFF site-specific, it has a consistent probability of exceedance ($5x10^{-4}$/yr) at each oscillator frequency and envelopes the $5x10^{-4}$/yr uniform hazard spectrum.

The selected MFFF design earthquake spectrum envelopes the PC-3 Spectrum, and therefore has even lower probabilities of exceedance than the PC-3 Spectrum. The attached Figure RAI-017-1 compares the MFFF soil surface design earthquake spectrum (0.2g Regulatory Guide 1.60) to the soil surface uniform hazard spectrum at four frequencies (1, 2.5, 5 and 10 Hz). It can be seen that at a frequency of 1 Hz the spectral acceleration for the MFFF design spectrum is less than the 10,000-year UHS. For frequencies of 2.5, 5 and 10 Hz, the spectral accelerations for the 0.2g Regulatory Guide 1.60 soil surface design earthquake spectrum are greater than the 10,000-year UHS. Thus, at 2.5, 5 and 10 Hz, frequencies of practical structural interest, the MFFF design earthquake spectrum represents 10,000-year accelerations.

<u>References:</u>

Duke Cogema Stone & Webster, 2001. MOX Fuel Fabrication Facility Site Geotechnical Report, QL-1A (IROFS), DCS01-WRS-DS-NTE-G-00005-C

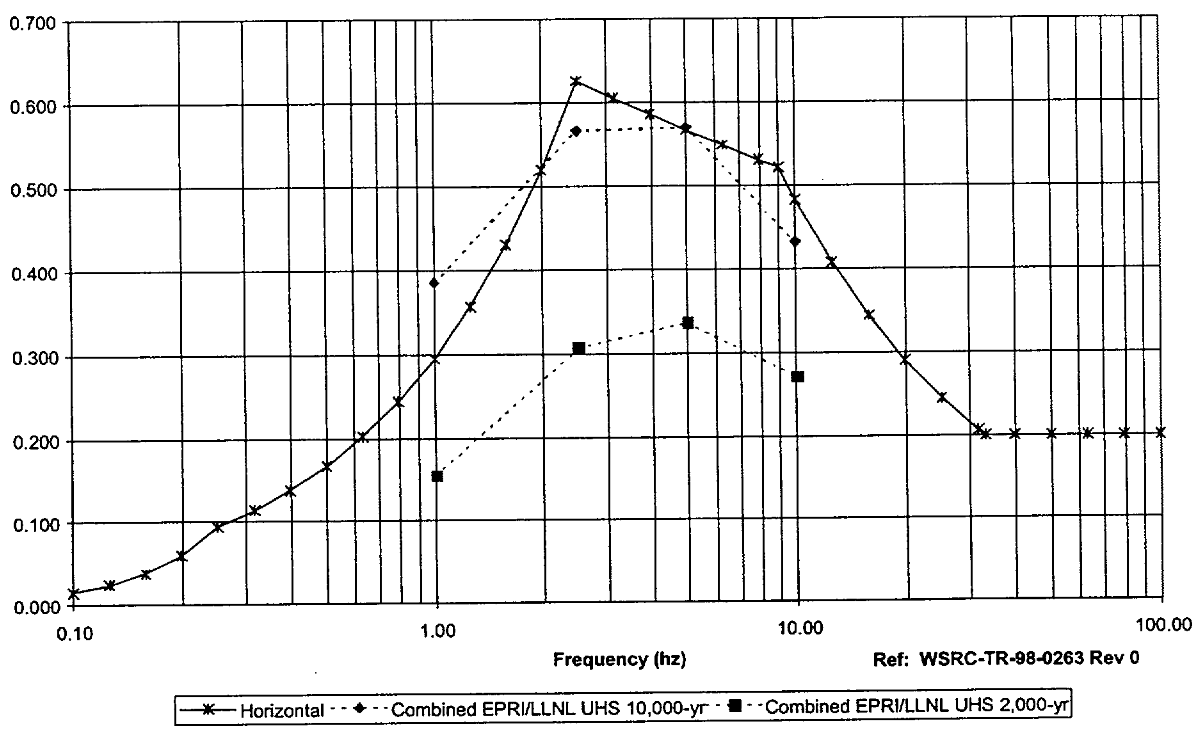
WSRC, 1997. SRS Seismic Response Analysis and Design Basis Guidelines, WSRC-TR-97-0085, Rev. 0.

WSRC, 1998. Soil Surface Seismic Hazard and Design Basis Guidelines for Performance Category 1 & 2 SRS Facilities, WSRC-TR-98-00263, Rev. 0

Action:

This information will be incorporated into the next update of the CAR.

Figure RAI-017-1 Comparison of MFFF Design Earthquake Response Spectra (Horizontal 5% Damping) to Soil Surface UHS at Four Spectral Frequencies



18. Section 1.3.6, pp. 1.3.6-1 thru 1.3.6-86

Provide the bases for selecting vertical spectrum as two thirds of the horizontal spectrum at corresponding frequencies.

The applicant did not provide justifications for the selection of vertical spectrum. Section 2.5.2.6 of NUREG-0800 (U.S. Regulatory Commission, 1997c) states, "Both horizontal and vertical component site-specific response spectra should be developed statistically from response spectra of recorded strong motion records...."

Response:

Using statistical studies of ground motions at SRS, a 0.2g Regulatory Guide 1.60 horizontal spectrum was chosen as the soil surface input for design of MFFF buildings and structures. A vertical spectrum equal to 2/3 the horizontal was initially selected. This selection of the magnitude of the vertical component was consistent with the guidance presented in ASCE 4-98, since available evaluations of SRS earthquake hazards did not indicate that near-field (closer than 15 km) earthquakes would be dominant. This selection formed the basis for the information presented in the MFFF Construction Authorization Request (CAR).

At the time of the selection of these criteria, DCS recognized that it would be necessary to further justify the V = 2/3 H relationship during NRC reviews of the CAR. In the NRC exchange meeting in January 2001, DCS pointed out that V = 2/3 H would be the design basis presented in the CAR, but that an evaluation was ongoing, and that if the value needed to be changed, the criteria would be updated. A review of hazard evaluations for SRS, such as WSRC-TR-99-00271, *Computation of USGS Soil UHS and Comparison to NEHRP and PC1 Response Spectra for the SRS* (WSRC, 1999), indicated that although the near-field earthquakes are not dominant, their contribution is potentially significant. DCS requested WSRC perform a study of the magnitude of the vertical component, and the results of this study were documented in WSRC-TR-2001-00342, *Development of MFFF-Specific Vertical-to-Horizontal Seismic Spectral Ratios* (WSRC, 2001a). The study indicated that the vertical component would be greater than 2/3. After review, DCS has decided that it is appropriate to increase the value of the vertical spectrum.

ASCE 4-98 recommends that if near-field earthquakes are dominant, the ratio of vertical to horizontal spectral ordinates be taken as, at least, unity for frequencies above 5 Hz, 2/3 for frequencies below 3 Hz, and a transition between 3 Hz and 5 Hz. This is closely and conservatively approximated by the Regulatory Guide 1.60 vertical spectrum scaled to the same 0.2g peak ground acceleration. Therefore, for the MFFF, the vertical component of earthquake motion at the soil surface will be selected as the Regulatory Guide 1.60 vertical spectrum scaled to 0.2g peak ground acceleration. This results in vertical and horizontal spectra that are consistent with the guidance in ASCE 4-98 and Regulatory Guide 1.60, and appropriately consider the effects of near-field earthquakes. The attached Figure RAI-18-1 illustrates these spectra.

References:

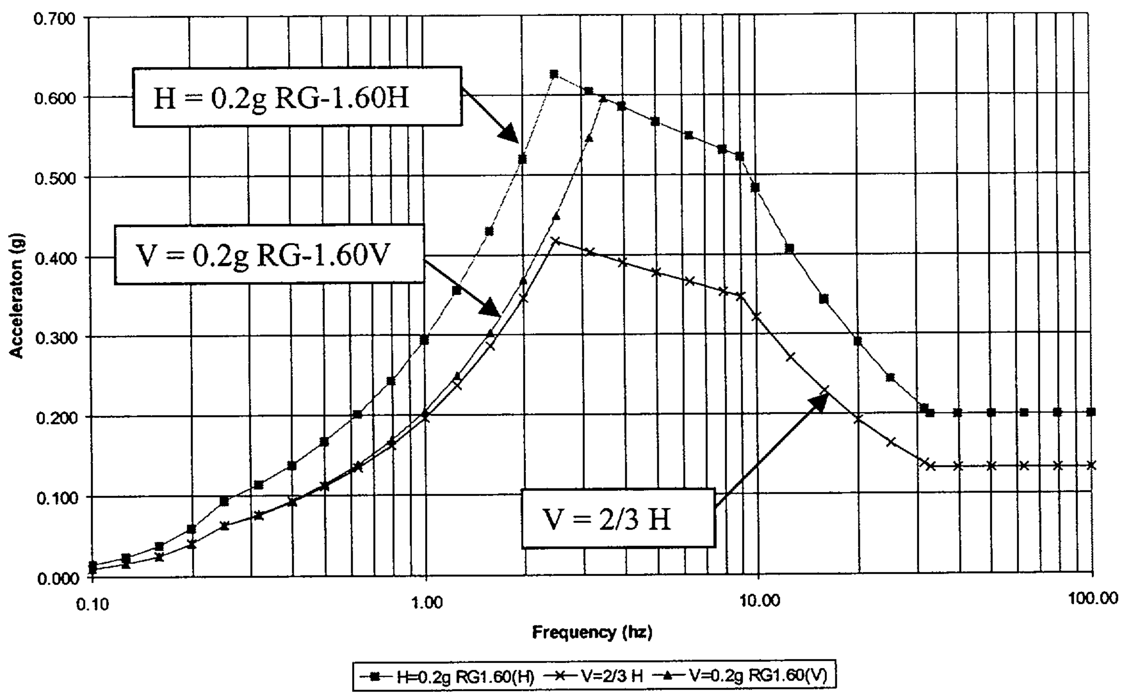
WSRC, 1999. *Computation of USGS Soil UHS and Comparison to NEHRP and PC1 Seismic Response Spectra for the SRS*, by R.C. Lee, WSRC-TR-99-00271, Rev. 0.

WSRC, 2001a. *Development of MFFF-Specific Vertical-to-Horizontal Seismic Spectral Ratios.* WSRC-TR-2001-00342, Rev 0

Action:

The above information, including Figure RAI-18-1, will be incorporated into the next update of the CAR.

Figure RAI-18-1 - Vertical Response Spectra (5% Damping)



19. Section 1.3.6, pp. 1.3.6-1 thru 1.3.6-86

Provide the following items with regard to ground motion modeling:

a. Details of the input and output in ground motion prediction.

b. Description of approaches used to account for uncertainties in ground motion prediction.

c. Results that show sensitivities of predicted ground motion to important input parameters such as stress drop.

d. Documentation on how the ground motion prediction models were used in calculating site-specific seismic hazard and in selecting seismic design spectra specific to the MFFF.

The applicant did not provide these items. Section 1.3.6.4 of MFFF application indicates that Random Vibration Theory (RVT) was used in predicting ground motion attenuation at the Savannah River Site. It further presented earthquake source parameters, bedrock and crustal path properties, and soil properties at the Savannah River Site. It is, however, not clear how the RVT model and path and site properties data were used in calculating the MFFF seismic hazard and in selecting the MFFF seismic design spectra. If Savannah River Site spectra were estimated based on LLNL and EPRI bedrock hazard results, as stated in the application as well as Lee, et al. (1997) and Lee (1998), ground motion models should include those used in the original LLNL and EPRI studies. These RAI items are consistent with regulatory requirements described in Position 4 and Appendix E of Regulatory Guide 1.165 (U.S. Regulatory Commission, 1997b), Section 2.5.2.5 (Seismic Wave Transmission Characteristics of the Site) of NUREG-0800 (U.S. Regulatory Commission, 1997c), and Section 5 (Methodology for Estimating Ground Motion on Rock) of NUREG/CR-6372 (U.S. Nuclear Regulatory Commission, 1997a).

Response:

Response 19a.

The Electric Power Research Institute (EPRI) and the Lawrence Livermore National Laboratory (LLNL) seismic hazard studies conducted for the eastern U.S. from the mid 1980s to early 1990s were the PSHAs used to develop design spectra at the SRS (WSRC, 1997). The basis for the use of the arithmetic mean of the two PSHAs is contained in DOE-STD-1023. This response will point to particular sections as necessary in the EPRI and LLNL documentation rather than repeating or duplicating material. Where requested information is not in the original reports and is available, this information is provided herein. Where requested information is not available, it is so noted. The development of design spectra at the SRS is contained in the response to Question 19d.

The discussions of the ground motion inputs are treated separately for EPRI and then LLNL. Note that in the development of the SRS design spectra, the DOE contracted separately to Jack Benjamin and Associates and the LLNL to provide hard rock results for the SRS. SRS central site coordinates were used for the computation of EPRI and LLNL bedrock PSHAs. While the

SRS results are not specifically documented in the EPRI and LLNL reports, the PSHA for the deep soil site plant Vogtle site (located approximately 19 km west of the center of the SRS) are contained in both the EPRI and LLNL documentation. Because of the close proximity of the Vogtle site to SRS, references to data summaries and results for site Vogtle will be used.

EPRI (1988)

Three equally weighted ground motion attenuation models were used. Section 6.2.2 of Vol 4 (EPRI, 1988) describes the three attenuation models as:

1. "Semi-theoretical methods utilizing a theoretical scaling model and low-magnitude recordings from the central United States.
2. An empirical method, that uses ground motion data from the central and eastern United States (instrumental, including seismograph data, and Modified Mercalli Intensity (MMI) and data from California (in the form of a regression of MMI, magnitude, and distance, on instrumental ground motion).
3. A random-vibration method, which utilizes a source scaling model, a random-process representation of acceleration, and a simplified representation of propagation effects."

LLNL (1989, 1993)

The ground motion update and elicitation is contained in Appendix C (Savy et al., 1993). The ground motion panel members were: K. Aki, J. Anderson, G. Bollinger, M. Chapman, D. Boore, K. Campbell, J. Fletcher, R. Herrmann, and M. Trifunac. Ground motion distributions are contained in Appendix C, Tables 3.2 (for PGA) and 3.3 (for Spectral Velocity). Descriptions of the ground motion attenuation models are contained in Appendix C, Tables 3.4 (PGA) and 3.5 (Sv). Of the 19 models for spectral velocity, all were random vibration theory (RVT) type models except for four models based on intensity attenuation. Appendix C, Table 4.1 contains comparisons of PGA values for the nominal case of an earthquake of magnitude m_b= 6.25 at a distance of 10 km. Appendix C, Table 5.1 contains comparisons of 5 Hz spectral velocity for the nominal case of an earthquake of magnitude m_b= 6.25 at distances of 10, 50, 100, and 250 km. Appendix C, Table 5.2 contains fractile comparisons of 5 Hz spectral velocity for a nominal case of an earthquake of magnitude m_b= 6.25 at a distance of 10 km for σ_{ln} values of 0.3, 0.5, and 0.8.

Response 19b.

The incorporation of uncertainty in ground motion was accomplished in several ways in the LLNL and EPRI studies. Both studies used alternate estimates of ground motion attenuation from the literature and from ground motion experts. For a given attenuation model the LLNL evaluation used alternate values of ground motion uncertainty, whereas the EPRI study used only one value (σ =0.5). Recent recommendations for PSHAs (Budnitz et al., 1997) suggest specific inclusion of two types of uncertainty, aleatory (randomness) and epistemic (lack of knowledge). Both EPRI and LLNL ground motion models incorporate both aleatory and epistemic uncertainties through the incorporation of both the distributions of ground motion and the use of alternate ground motion attenuation models.

Response 19c.

Neither the EPRI or LLNL reports had significant documentation of sensitivity of ground motion parameters to the final hazard results. Sensitivity of the EPRI PGA hazard results to the three attenuation models for the Vogtle site is illustrated in Vol 4, Figure 6-6d of EPRI 1988. Sensitivities to source stress drop or other source parameters were not given.

Response 19d.

This section of the response documents how the SRS PC-3 and PC-4 design spectra are developed.

The site spectra are development in accordance with DOE (1996). DOE-STD-1023 provides guidelines for developing site-specific probabilistic seismic hazard assessments and criteria for determining ground motion parameters for the design earthquakes. It also provides criteria for determination of design response spectra.

The fundamental elements of the ground motion criteria for higher hazard nuclear facilities (PC3 and PC4) are:

1. A probabilistic seismic hazard assessment (PSHA) must be conducted for the site (or use an existing PSHA that is less than 10 years old).
2. A target design basis earthquake (DBE) response spectrum is defined by the mean UHS.
3. Mean UHS shapes are checked by median site-specific spectral shapes that are derived from de-aggregated PSHA earthquake source parameters. The median site-specific spectral shapes are scaled to the UHS at two specific frequencies (average of 1-2.5, and 5-10 Hz).
4. Estimated site specific ground motions from historical earthquakes (significant felt or instrumental with moment magnitude (Mw) > 6 are developed using best estimate magnitude and distance.
5. Spectral shapes are adjusted until DBE response spectra have a smooth site-specific shape.

The detailed approach to meet the above acceptance criteria is given below including a summary of references to reports containing detailed descriptions of the approach:

- SRS PC-3 and PC-4 Design Criteria follow DOE-STD-1023 (DOE, 1996).
- The SRS mean EPRI and LLNL soil surface PSHAs are rejected on the basis that the site response is generic and not representative of site conditions (WSRC, 1997).
- A mean bedrock UHS spectra is computed for two probabilities of exceedance, 5×10^{-4} (PC3) and 1×10^{-4} (PC4) using UHS assessments for SRS from EPRI and LLNL (WSRC, 1997).
- The mean bedrock UHS is broadened using site-specific spectral shapes generated using EPRI M-bar and D-bar values at each POE (WSRC, 1997).

- SRS strain-dependent dynamic properties were measured with calibrated laboratory testing systems and site-wide strain-dependent properties were recommended for the site (WSRC, 1996).
- A soil profile database was prepared and parameterized to simulate SRS velocity profiles (Toro, 1996).
- An RVT ground motion model is used to compute suites of hypothetical bedrock control motions for several magnitudes to evaluate soil response for suites of bedrock and soil configurations (WSRC, 1997).
- Six categories of soil column thickness and bedrock type were selected to explore the range of soil response across the SRS (WSRC, 1997).
- Mean soil amplification functions (SAFs) (ratio of soil response to bedrock input) are computed based on statistical analysis of each of the soil/bedrock categories accounting for earthquake magnitude and bedrock control motion (WSRC, 1997).
- The broadened mean bedrock UHS is multiplied by the frequency and magnitude dependent mean SAFs to obtain a mean surface response spectra (WSRC, 1997).
- The soil UHS for each performance category is conservatively taken as the envelope of the soil spectra obtained by applying magnitude and frequency dependent SAFs (WSRC, 1997).
- Deterministic median and 84th percentile Charleston earthquake design basis spectra are generated for each of the soil and bedrock categories from a suite of conservative Charleston earthquake sources (WSRC, 1997).
- The PC3 SRS design basis spectrum is the envelope of the median Charleston earthquake spectra and the PC3 soil UHS envelope (WSRC, 1997).
- The PC4 SRS design basis spectrum is the envelope of the 84th percentile Charleston earthquake spectra and the PC4 soil UHS envelope (WSRC, 1997).
- Validation that the soil UHS are mean spectra and meet the requirements of DOE-STD-1023 are done by developing a methodology to compute soil hazard directly for a selected number of frequencies from the bedrock hazard spectra (WSRC, 1997)
- Evaluation of the exact mean surface EPRI and LLNL hazard and UHS was done to validate the mean PC3 and PC4 design spectra (WSRC, 1998).
- For application to existing and new PC3 and PC4 facilities, review or collection of site specific data will be required for development of "confirmed" design spectra. Otherwise the design basis spectrum is considered "committed" in accordance with the WSRC E7 manual (WSRC, 1997).
- For the MFFF site the results of site-specific exploration program were reviewed in accordance with the recommendations given in WSRC (1997). The results of the review confirm the acceptability of the SRS PC-3 and PC-4 response spectra for the MFFF site (see response to Question 14 for additional details).

On 10 August 2001, DCS submitted DCS01-WRS-DS-NTE-G-00005-C, *MOX Fuel Fabrication Facility Site Geotechnical Report* for NRC review. This report contains a detailed presentation of the site-specific geotechnical analyses. This document clearly demonstrates the applicability of the SRS site generic PSHA to the MFFF site. (Also refer to the responses to Questions 13 and 14.) Further, Section 6.2 of the report compares the scaled uniform hazard bedrock motion convolved through the soil column to the MFFF design spectrum (0.2 Regulatory Guide 1.60

spectrum). The scaled uniform hazard bedrock motion used for the MFFF site is the SRS PC-3 control motion increased by a factor of 1.25, and is designated as the PC-3+ control motion. The 1886 Charleston earthquake (50th percentile) control motion attenuated to rock at SRS was also analyzed. The results of these analyses, as presented in Section 6.2 of this report, indicates that the PC-3+ control motion generates a surface PGA of 0.2g. Refer to the response for Question 17 for further clarification. The strain-compatible dynamic soil properties generated by the PC-3+ control motion are considered appropriate for use in the dynamic soil-structure interaction (SSI) analyses of the MFFF critical structures.

References:

Budnitz, R.J., G. Apostolakis, D.M. Boore, L.S. Cluff, K.J. Coppersmith, C.A. Cornell and P.A. Morris, 1997. Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts, NUREG/CR-6372, UCRL-ID-122160.

Electric Power Research Institute, 1988. Seismic Hazard Methodology for the Central and Eastern United States, Vol. 1-10, NP-4726.

Savy, J.B., A.C. Boissonnade, R.W. Mensing and C.M. Short, 1993. Eastern U.S. Seismic Hazard Characterization Update, Lawrence Livermore National Laboratory, July 20, 1993.

Toro, G.R., 1996. Probabilistic of Site Velocity Profiles at the Savannah River Site, Aiken, South Carolina, Final Report to WSRC, April 4, 1997. Attached as an Appendix to WSRC (1997).

U.S. Department of Energy, 1996. DOE Standard: Natural Phenomena Hazards Assessment Criteria, DOE-STD-1023-95, Change Notice #1, Washington, D.C., January 1996.

WSRC, 1996. Investigations of Nonlinear Dynamic Soil Properties at the Savannah River Site, WSRC-TR-96-0062.

WSRC, 1997. SRS Seismic Response Analysis and Design Basis Guidelines, WSRC-TR-97-0085, Rev. 0.

WSRC, 1998. Soil Surface Seismic Hazard and Design Basis Guidelines for Performance Category 1 & 2 SRS Facilities, WSRC-TR-98-00263, Rev. 0, 9/30/98.

Action:

DCS will provide the references cited above.

20. Section 1.3.6, pp. 1.3.6-1 thru 1.3.6-86

Clarify how and why the return periods were evaluated on page 1.3.6-13 of the application.

It appears from the application that the return periods were chosen from 5 Hz spectrum accelerations (Table 1.3.6-7 and Figure 1.3.6-16). The applicant did not give the bases and purposes for evaluation of return periods from 5 Hz spectrum accelerations.

Response:

Section 1.3.6.6 on page 1.3.6-23 describes the selection of the MFFF Design Earthquake, using the SRS site-wide criteria as an input. The narrative refers to Figure 1.3.6-16 as representing the soil surface hazard relationships for SRS. Figure 1.3.6-16 expresses the hazard in terms of spectral velocity versus annual probability of exceedance. The source document that developed this hazard definition (WSRC, 1998, shown in the CAR as "Lee, R.C., 1998,") also contains representations of the hazard in terms of spectral acceleration. The attached Figure RAI-20-1 shows the appropriate soil surface acceleration hazard curve.

Using the acceleration hazard relationships shown in Figure RAI-20-1 for each of the four oscillator frequencies (1 Hz, 2.5 Hz, 5 Hz, and 10 Hz) represented in the hazard chart, the spectral acceleration was read off each of the 5% damped response spectra discussed (CAR - Figure 1.3.6-20 for PC-3, Figure 1.3.6-14 for PC-4, and Figure 1.3.6-21 for Regulatory Guide 1.60). These spectral accelerations were used to enter Figure RAI-20-1 and to read the associated annual mean probability of exceedance. Inverting the annual mean probability of exceedance results in the return period reported in CAR Table 1.3.6-7.

The return periods presented in the CAR are used to demonstrate that the selected soil surface Design Earthquake spectrum represents essentially 10,000-year spectral accelerations to be used in the design of the MFFF. Therefore, by selecting accelerations consistent with a (10,000-year) 10^{-4}/yr mean annual probability of exceedance, the design earthquake meets the intent of Regulatory Guide 1.165, which suggests a 10^{-5}/yr median annual probability of exceedance.

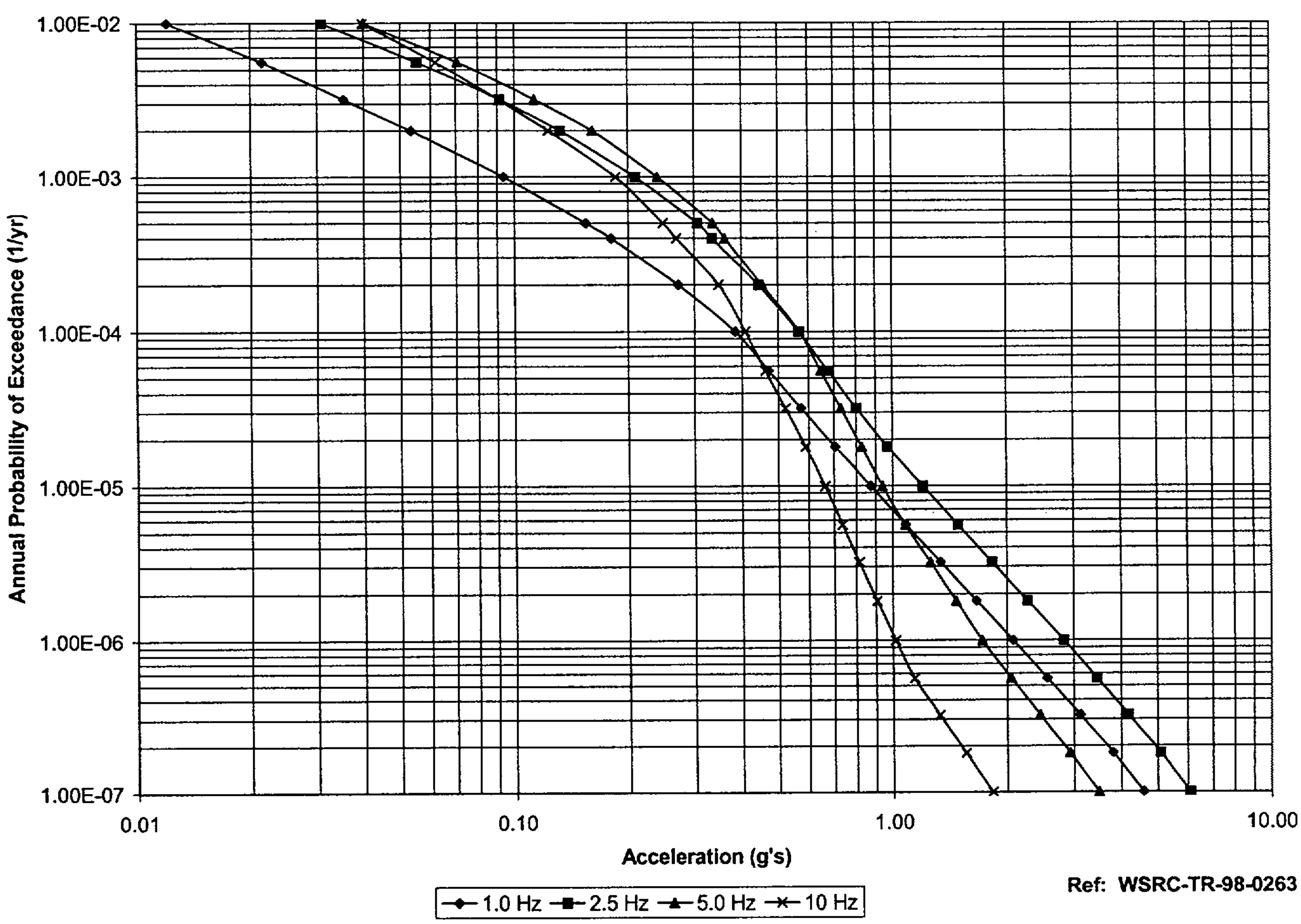
References:

WSRC, 1998. Soil Surface Seismic Hazard and Design Basis Guidelines for Performance Category 1 & 2 SRS Facilities, WSRC-TR-98-00263, Rev. 0, 9/30/98.

Action:

To provide additional clarity, Figure RAI-20-1 will be included in the next update of the CAR, and will be cited in CAR Section 1.3.6.6.

Figure RAI-20-1 SRS Soil Surface Seismic Hazard Curves



Ref: WSRC-TR-98-0263 Rev 0

21. Section 1.3.6, pp. 1.3.6-1 thru 1.3.6-86

Document how the spectral ground accelerations were converted to velocities as shown in Figure 1.3.6-16.

The ground motion in PC-3, PC-4, and 0.2g Regulatory Guide 1.60 Spectra are given as acceleration, whereas ground motion is given as velocity in Figure 1.3.6-16. The applicant did not show how the spectral ground accelerations were converted to velocities to obtain the return period from Figure 1.3.6-16.

Response:

Spectral velocities were determined from spectral accelerations (in g's) using the following relationship derived from the solution of a single degree of freedom harmonic oscillator:

$$Sv = Sa*981/(2*\pi*f)$$

Where Sv is spectral velocity in units of cm/sec, Sa is spectral acceleration in units of Earth's gravity (g's), the factor 981 is the Earth's gravitational acceleration in units of $cm/sec^2/g$ and f is the oscillator frequency in units of sec^{-1}.

SRS hazard curves for both spectral acceleration and velocity are tabulated in WSRC (1998), Tables 6.6, 6.7, 6.8, and 6.9 for 1, 2, 5, and 10 Hz, respectively.

Reference:

WSRC (1998). Soil Surface Seismic Hazard and Design Basis Guidelines for Performance Category 1 & 2 SRS Facilities, WSRC-TR-98-00263, Rev. 0, 9/30/98.

Action:

As stated in the response to Question 20, to provide additional clarity, Figure RAI-20-1 will be included in the next update of the CAR, and will be cited in CAR Section 1.3.6.6, illustrating the appropriate acceleration hazard curve.

22. Section 1.3.6, pp. 1.3.6-1 thru 1.3.6-86

Clarify the description given on page 1.3.6-19 (first paragraph) for Figure 1.3.6-16.

It appears that the description given on page 1.3.6-19 is inconsistent with the figure itself.

Response:

Figure 1.3.6-16 illustrates the combined EPRI and LLNL SRS soil surface hazard envelope for oscillator frequencies of 1, 2.5, 5, and 10 Hz. The second sentence of text on page 1.3.6-19 refers to segments of the bedrock hazard curves that were extrapolated to probabilities of exceedance beyond those available from the LLNL and EPRI hazard evaluations. These bedrock hazard curves were not included in Figure 1.3.6-16. However, the bedrock hazard curves and their extrapolations are illustrated in WSRC (1998) in Figures 6.1, 6.2, 6.3, and 6.4 for the EPRI 1, 2.5, 5, and 10 Hz frequencies, respectively and in Figures 6.5, 6.6, 6.7, and 6.8 for the LLNL 1, 2.5, 5, and 10 Hz frequencies, respectively.

References:

WSRC (1998). Soil Surface Seismic Hazard and Design Basis Guidelines for Performance Category 1 & 2 SRS Facilities, WSRC-TR-98-00263, Rev. 0, 9/30/98.

Action:

The second and third sentences in the first paragraph on page 1.3.6-19 will be deleted in the next update of the CAR.

23. Section 1.3.7, pp. 1.3.7-1 thru 1.3.7-3

Provide technical basis for excluding slope instability hazard, with a detailed topographic contour map (1-ft interval) of the site including the locations of the principal structures, systems, and components (SSCs).

The application does not discuss adequately slope instability at the site. Slope instability is one of the natural phenomena hazards that could affect the performance of the principal structures and systems. Regulatory requirement 10 CFR 70.22(f) requires an application to contain a safety assessment of the design bases of the principal SSCs for the proposed facility, including provisions for protection against natural phenomena. Regulatory requirement 10 CFR 70.62(c)(iv) requires the applicant to conduct an integrated safety analysis to identify potential accident sequences. The analysis should include natural phenomena hazards. Also, 10 CFR 70.64(a)(2) requires applications to address the baseline design criteria related to the natural phenomena hazards. In Section 5.5 of the application, avalanche and landsliding events are determined to be not credible because the site is relatively flat. Figure 1.3.1-2 of the application seems to support the statement that the site is relatively flat. However, Figure 1.3.1-2 is a regional topographic map with a relatively large scale. Consequently, this map does not contain sufficient information for staff to determine with reasonable assurance that the slope instability hazard can be excluded as a baseline design criterion.

Response:

The attached site contour map, Figure RAI-023-1, defines the original topography, proposed finish grades, the location of the major cut and fill slopes, and the location of the principal SSCs. The nearest cut slopes will be over 400 feet both north and west of the MOX Fuel Fabrication Building (BMF) and are only approximately 15 feet high. The BMF and the Emergency Diesel Generator Building (BEG) will be located with finished floor elevations below the existing ground elevation, and are both over 400 feet from the top of the nearest fill slope or steeper topographic slope. Figure RAI-023-1 shows the fill and steep slopes to the northwest, northeast and southeast of the BMF and BEG. Therefore, slope stability from existing topography and planned fills and cuts at the MFFF site will not have any adverse impact to principal SSCs.

Action:

Include information from the above referenced figure in the next update of the CAR.

Figure removed under 10 CFR 2.390.

24. Section 1.3.7, pp. 1.3.7-1 thru 1.3.7-3

Provide the following items which are related to the effect of the Actinide Packaging and Storage Facility spoil at the site on the principal SSCs.

a. Discuss the location of the spoil pile relative to the site, particularly to the principal structures and systems.

b. Discuss the characteristics of the spoil pile including material properties.

c. Assess the static and dynamic stability of the spoil pile, if necessary.

d. Discuss the potential effects of the spoil pile, if determined to be unstable, on the principal structures and systems, if necessary.

e. Discuss remedial measures if the spoil pile is determined to be potentially unstable and the performance of a structure or system will be affected.

Although the spoil pile is a man-made feature, its instability may still affect the performance of the principal SSCs at the site. It is, therefore, necessary to be addressed as a baseline design criterion unless the analysis performed pursuant to 10 CFR 70.62(c) demonstrates that the spoil pile instability is not a credible event or will not affect the performance of the principal SSCs. These items are consistent with Section 2.5.5 (Stability of Slopes) of NUREG-800 (U.S. Nuclear Regulatory Commission, 1997c) and Regulatory Guide 3.11 (U.S. Nuclear Regulatory Commission, 1977).

Response:

The site contour map provided in response to Question 23 shows the location of the spoil pile created from the excavated materials removed from the Actinide Packaging and Storage Facility (APSF). During MFFF site grading, this APSF spoil pile will be removed and will not be used in connection with foundations for the principal SSCs. Therefore, the pile cannot adversely impact principal SSCs.

Action:

This information will be included in the next update of the CAR.

25. <u>Section 1.3.7, pp. 1.3.7-1 thru 1.3.7-3</u>

Provide analysis on the liquefaction susceptibility of loose subsurface materials at the site. The analysis results should include the following items:

a. Analysis/interpretation of the data collected from the field programs and laboratory testing to characterize site conditions and to define behavior characteristics of the native soils at the site.

b. Results of the liquefaction potential analysis at the site.

c. Effects of liquefaction on the principal SSCs.

The application does not contain a technical discussion on liquefaction susceptibility, its potential effect, and remedial measures although it identifies liquefaction as a potentially credible event (Section 5.5 of the application). Liquefaction is a natural phenomena hazard required to be included in the integrated safety analysis and baseline design criteria pursuant to 10 CFR 70.62(c) and 10 CFR 70.64(a)(2). Also, 10 CFR 70.22(f) requires the application to contain a safety assessment of the design bases of the principal SSCs for the proposed facility, including provisions for protection against natural phenomena. These items are consistent with Section 2.5.4 (Stability of Subsurface Materials and Foundations) of NUREG-800 (U.S. Nuclear Regulatory Commission, 1997c) and Regulatory Guide 3.11 (U.S. Nuclear Regulatory Commission, 1977).

Response:

On 10 August 2001, DCS submitted DCS01-WRS-DS-NTE-G-00005-C, *MOX Fuel Fabrication Facility Site Geotechnical Report* for NRC review. This report contains a detailed presentation of the site investigations that have been conducted and site-specific analyses, including liquefaction analyses. Section 8 of this document demonstrates the acceptability of the MFFF site with respect to liquefaction and post-earthquake dynamic settlement. This section of the report also demonstrates that the 1886 Charleston earthquake control motion (also refer to the response for Question 19) is the controlling earthquake motion for liquefaction and post-earthquake dynamic settlement for the MFFF site. The results of the analyses indicate that any areas of potential liquefaction are at depth and exist as isolated pockets that are confined by strong non-liquefiable soils. The analyses also indicate that post-earthquake dynamic settlements are not excessive and considered acceptable for the design of principal SSCs.

Action:

None

26. <u>Section 1.3.7, pp. 1.3.7-1 thru 1.3.7-3</u>

Define the lateral extent/boundary of each soft zone identified at the site.

Section 1.3.7.2 of the application states that, "The exploration programs indicated that soft zones at the MFFF site are isolated and found as soft soil pockets at depth." However, the lateral extent of these soft zones were not provided in the application or its supporting documents.

Response:

On 10 August 2001, DCS submitted DCS01-WRS-DS-NTE-G-00005-C, *MOX Fuel Fabrication Facility Site Geotechnical Report* for NRC review. This report contains a detailed presentation of the site investigations that have been conducted and site-specific analyses, including evaluation of the potential effects of soft zones. Section 3.2 of this report presents the development and nature of soft zones at SRS and the MFFF site. Section 5.2 of this report presents the criteria used to define the location and extent of soft zones at the MFFF site and engineering properties used to evaluate soft zones are presented in Section 6.1. The effects of static and dynamic settlement from soft zones is presented in Sections 7 and 8 of this report. This report confirms that any soft zones identified beneath or adjacent to principal SSCs will not have any adverse effect on these structures.

Action:

None

27. Section 1.3.7, pp. 1.3.7-1 thru 1.3.7-3

Clarify the following two statements:

a. "Once the location and extent of the soft zones on the MFFF site were identified, the MFFF principal SSCs, such as the MOX Fuel Fabrication Building and the Emergency Diesel Generator Building were relocated to areas of the site found to be free of soft zones." (Section 1.3.5.2, MFFF Site Geology)

b. "MFFF principal SSCs, such as the MOX Fuel Fabrication Building and the Emergency Diesel Generator Building, were located on the MFFF site to avoid placement directly over significant soft zones." (Section 1.3.7.2, Evaluation of Soft Zones)

These two statements appear to be contradictory with each other. Furthermore, definition on "significant" soft zones needs to be provided. Compare Figure 11.1-1 of the MFFF Construction Authorization Request with Figure 2 of the MOX Fuel Fabrication Facility Site Geotechnical Report (DCS, 2000), the MOX Fuel Fabrication Building seems to be located above the soft zones, particularly the northern portion of the building.

Response:

The location of principal SSCs on the MFFF site was changed to account for the soft zones discovered during site exploration. The SSCs were relocated to the configuration that is evaluated in DCS01-WRS-DS-NTE-G-00005-C, *MOX Fuel Fabrication Facility Site Geotechnical Report*. This report contains a detailed presentation of the site-specific geotechnical analyses. This report also demonstrates the acceptability of the soft zones identified at the MFFF site. The response to Question 26 presents sections of this report that address soft zones. Tables and Figures in these sections of the report reflect recent evaluation and analysis of soft zones at the MFFF site and at locations beneath and adjacent to the BMF and BEG.

Action:

Clarification of this issue will be provided in the next update of the CAR.

28. Section 1.3.7, pp. 1.3.7-1 thru 1.3.7-3

Provide an analysis on soft zones regarding:

a. Mechanical and strength properties of the soft zone soils and the representativeness of these properties.

b. Potential load increase in soft zone soils due to static and seismic design loads.

c. Deformations that may result in the soft zones from static and dynamic foundation loading.

d. Structure settlement that may result from the deformations of the soft zones to critical structures.

The application does not provide analysis on the stability of the soft zones identified when subjected to foundation and dynamic loads. Consequently, the effect of the soft zone deformation on the design of the critical structures cannot be evaluated by the staff. Regulatory requirement 10 CFR 70.64(a)(2) requires that the design to provide adequate protection against natural phenomena hazards.

Response:

As indicated in the response to Question 26, on 10 August 2001, DCS submitted DCS01-WRS-DS-NTE-G-00005-C, *MOX Fuel Fabrication Facility Site Geotechnical Report* for NRC review. This report contains a detailed presentation of the site investigations that have been conducted and site-specific analyses, including evaluation of the potential effects of soft zones. This document demonstrates the acceptability of the MFFF site with respect to identified soft zones beneath and adjacent to principal SSCs. Specific sections of this report that are applicable to the response for this question are discussed in Questions 26 and 27.

Action:

None

29. Section 1.3.7, pp. 1.3.7-1 thru 1.3.7-3

Provide the following references to facilitate review of the application:

a. Westinghouse Savannah River Company, 1999. *Significance of soft zone sediments at the Savannah River site-Historical review of significant investigations and current understanding of soft zone origin, extent and stability*, Report No. WRSC TR 99 4083, Rev 0, September.

b. Duke Cogema Stone & Webster. 2000a. *Specification for Geotechnical Test Borings and Sampling*. DCS01-WRS-DS-SPE-G-00002-A, April 11.

c. Duke Cogema Stone & Webster. 2000b. *Specification for Laboratory Testing of Soils Quality*. DCS01-WRS-DS-SPE-G-00003-A, May.

d. Duke Cogema Stone & Webster. 2000c. *Specification for Cone Penetration Testing of Soil*. DCS01-WRS-DS-SPE-G-00001-A, April 11.

e. "Revised Envelope of the Site Specific PC-3 Surface Ground Motion", Memorandum from Brent Gutierrez to Lawrence Salonmone and Fred Loceff, September 9, 1999.

f. Stokoe, K.H., et al., 1995. "Correlation Study of Nonlinear Dynamic Soil Properties: Savannah River Site, Aiken, South Carolina," Rev. 0, File No. Savannah River Site-RF-CDP-95, University of Texas at Austin, Department Civil Engineering, September 13.

Response:

The items referenced above will be provided as requested. In addition, the following reference will be provided to summarize and to extend the conclusions of Stokoe 1995, requested above:

g. Westinghouse Savannah River Company, 1996. *Investigations of Nonlinear Dynamic Soil Properties at the Savannah River Site*, Report No. WRSC-TR-96 0062, Rev 0, March.

Action:

Include references as part of the response to the RAI. In addition, the following references cited in the responses to questions on CAR Sections 1.3.3, 1.3.6, and 1.3.7 are also provided:

References for Questions 9, 10, and 11 on Section 1.3.3:

- McDonald, James R., 1982, "Assessment of Tornado and Straight Wind Risks at the Savannah River Site", October 1982.
- Coats, D. W. and R. C. Murray, 1985, "Natural Phenomena Hazards Modeling Project: Extreme Wind/Tornado Hazards Models for Department of Energy Sites", UCRL-53526, Rev. 1, Lawrence Livermore National Laboratory, University of California, August 1985.

- *Savy, J.B., A.C. Boissonnade, R.W. Mensing and C.M. Short, 1993. Eastern U.S. Seismic Hazard Characterization Update, Lawrence Livermore National Laboratory, July 20, 1993.
- Savy, J.B., 1996. Fission Energy and Systems Safety Program, May 28, 1996, SANT96-147JBS, letter from J. B. Savy, Deputy Associate Program Leader Natural Phenomena Hazards to Jeff Kimball, DOE.
- Toro, G.R., 1996. Probabilistic of Site Velocity Profiles at the Savannah River Site, Aiken, South Carolina, Final Report to WSRC, April 4, 1997. [Attached as Appendix A to WSRC, 1997 (WSRC-TR-97-0085).]
- WSRC, 1996. Investigations of Nonlinear Dynamic Soil Properties at the Savannah River Site, WSRC-TR-96-0062. (provided as part of response to Question 29) (shown in CAR as Lee, R.C., 1996)
- WSRC, 1997. SRS Seismic Response Analysis and Design Basis Guidelines, WSRC-TR-97-0085, Rev. 0. (shown in CAR as Lee, R.C., et al., 1997)
- WSRC, 1998. Soil Surface Seismic Hazard and Design Basis Guidelines for Performance Category 1 & 2 SRS Facilities, by R.C. Lee, WSRC-TR-98-00263, Rev. 0. (shown in CAR as Lee, R.C., 1998)
- WSRC, 1999. Computation of USGS Soil UHS and Comparison to NEHRP and PC1 Seismic Response Spectra for the SRS, by R.C. Lee, WSRC-TR-99-00271, Rev. 0.
- WSRC, 2001. Applicability of SRS Site-Wide Spectra to the MFFF Site. Calculation Number K-CLC-F-00049, Rev. 0.
- WSRC, 2001a. Development of MFFF-Specific Vertical-to-Horizontal Seismic Spectral Ratios. WSRC-TR-2001-00342, Rev 0

* These documents are considered to be publicly available and are not included as part of the response.

** This document has been separately submitted to NRC on 10 August 2001 (Letter DCS-NRC-000057).

CHAPTER 2, FINANCIAL QUALIFICATIONS

30. Section 2.1, p. 2-1

Provide information on project costs.

Under Section 2.3.A of SRP, the applicant needs to provide information of project costs. In the application, DCS stated that project cost information has been prepared and submitted to DOE for approval. Once approved this information will be submitted under separate cover. This information has not yet been submitted to the NRC for review.

Response:

DCS is designing, constructing, and operating the MFFF under contract to the U.S. Department of Energy (DOE). Project cost information is provided in a separate proprietary transmittal. This information represents the most recent cost estimate provided to DOE. Note that the design costs are currently under review. Note also that these costs are not identical to those in the information provided with the ER RAI response (submitted by DCS to the NRC 12 July 2001; Letter Number DCS-NRC-000053). The difference is primarily attributable to the fact that the program life cycle cost values include DOE and other contractor costs.

Action:

Project cost information is submitted under separate cover.

31. Section 2.4, p. 2-3

Provide financial statements and Securities and Exchange Commission Report 10-K applicable to fiscal year 2000. If no Report 10-K is required, provide such a statement.

Under Section 2.3.D of the SRP, the applicant needs to provide the most recent financial statements and Securities and Exchange Commission Report 10-K. In the application, the applicant provided financial statements for the period March 22, 1999, to December 31, 1999. No Report 10-K was provided. The most recent financial statements and Report 10-K for the fiscal year ending December 31, 2000, is needed to assess the applicant's current financial status.

Response:

DCS is not a publicly held entity, and therefore does not prepare a Securities and Exchange Commission Report 10-K. DCS will submit, under separate cover, a proprietary financial statement providing information for the year ending December 31, 2000, the most current year available.

Action:

DCS is submitting, under separate cover, its financial statement for the year ending December 31, 2000.

CHAPTER 4, ORGANIZATION AND ADMINISTRATION

32. Chapter 4, General

Explain experience requirements.

Section 4.4.3.A.iii.a of the SRP states, for key management positions, "The personnel to design and construct the facility have the appropriate breadth and level of experience for their respective authorities and responsibilities..." [A similar requirement exists for operations. Section 4.4.3.B.iii.a of the SRP states, for key management positions, "The personnel to operate the facility have the appropriate breadth and level of experience for their respective authorities and responsibilities..."]

Section 4 provides the experience requirements for the management at the proposed facility. However, no requirement is identified for experience in fuel fabrication for any of the managers. In addition, there is no requirement for plutonium handling experience and training. More information is needed to demonstrate that the positions proposed for the facility will have the necessary experience and qualifications for safe operations at the MOX.

Response:

As discussed in CAR Section 4.1, the minimum qualifications of each of the key management positions include, in addition to education, a minimum amount of related experience, including (for design managers) experience in the design of nuclear facilities.

DCS could not find specific fuel-cycle-facility guidance on minimum qualifications and experience related to design and construction for key management personnel. Therefore, DCS reviewed ANSI/ANS 3.1-1993 and Regulatory Guide 1.8, Revision 3, which provide guidance for requirements for personnel at an operating nuclear reactor site. In addition, DOE Order 5480.20A uses these documents as guidance to develop minimum requirements both for DOE's operating reactor and non-reactor nuclear facilities. However, no substantial precedent was found for fuel-cycle facilities licensed under 10 CFR 70, as non-plutonium facilities are not required to secure authorization for construction (thus, minimum qualifications are provided exclusively for operating facilities, not for facilities during design and construction). Nonetheless, minimum qualifications in the license applications for several Part 70 facilities were reviewed, as well as the minimum qualifications for personnel at selected non-Part 70 facilities.

Each of the cited documents was examined for guidance. Values described in CAR Section 4.1 for years of experience were derived from reasonably analogous positions during operations. The minimum qualifications of the analogous positions are adjusted to account for the fact that the "special requirements" in the guidance are oriented towards an operating facility, whereas the positions described for the MFFF are for engineering and construction.

While deriving the minimum qualifications for MFFF management personnel, DCS reviewed and considered the SRP's statement that "...personnel to design and construct the facility have the appropriate breadth and level of experience for their respective authorities and responsibilities...." DCS managers meet the minimum qualifications stated in the CAR and have a breadth of experience commensurate with their responsibilities, including experience in facilities with activities similar to those anticipated in the MFFF. Significant nuclear facility experience – including MOX processing experience – is represented with the MOX project staff, including the engineering organization. These personnel supplement the key managers' skills consistent with the proviso in the "special requirements" paragraphs of the cited guidance documents that allow the various managers to supplement their own skills with those of their assistants. Consequently, DCS does not believe that the specifics of "experience in fuel fabrication," "plutonium handling experience," operating facility experience, or other types of experience relevant to their management functions, are necessary additions to the minimum qualifications for the engineering and construction managers.

Action:

In accordance with the SRP, DCS will address the minimum qualifications and experience for key management personnel responsible during MFFF operations in its application for a license to possess and use special nuclear material.

33. Chapter 4, pp. 4-1 thru 4-6

Provide a full description of the applicant's organization for construction.

SRP Section 4.3 states that the applicant's identification and functional description of the specific organizational groups responsible for designing and constructing the facility should include contractors, consultants, and other outside service organizations in addition to the applicant. This should also include, but not be limited to, the process designers, architect engineering firm, and the construction contractor.

Chapter 4 of the application has brief descriptions relating to "construction management," but does not address the actual construction responsibilities and interfaces. Clarify if there will be major delegation of work such as to an architect/engineer, a MOX plant constructor, an integration contractor, system suppliers or other on- or off-site organizations. The authorities and responsibilities among the organizational groups and the means of communication should be addressed, including the DCS design and engineering functions and the various contractors during construction. Organization charts should reflect the lines of responsibility and authority. Clear and unambiguous controls and communications, and responsibility and authority between the construction, equipment and system suppliers and DCS design, engineering, project management, procurement, construction management and quality assurance (QA), should be identified. All key management positions for construction activities should be adequately addressed. Specific activities such as inspection and testing of construction activities, equipment and SSCs should be adequately addressed as to what organization performs them, and what, how and by whom, QA controls and management measures are applied.

(Note: The same issue applies to Application Section 15.1.1, DCS Organization.)

Response:

Question 1 of the NRC's "Request for Additional Information on the Duke Cogema Stone & Webster (DCS) Mixed Oxide Project Quality Assurance Plan, Revision 2" (NRC letter dated 6/19/01) requested the same information. See the DCS response to that question in DCS letter dated 7/18/01, DCS-NRC-000054, "Response to NRC Request for Additional Information on MOX Project Quality Assurance Plan (MPQAP) Revision 2".

Action:

The DCS response to the MPQAP RAI question indicates that various sections of the MPQAP will be revised to describe changes discussed in the response. In addition, CAR Chapter 4 will be revised to describe changes for support of construction to the project management organizational chart and specifically adding the construction management organization details for the Construction Manager, Resident Engineer, and CM Area Superintendents including roles, responsibilities, and interfaces described in the response. Section 4.1.4.1, QA Manager, will also be revised to describe the oversight of construction subcontractors. Functions for each of the positions will also be added to Figure 4-1.

34. Chapter 4, pp. 4-1 thru 4-6

Specifically explain the organizational responsibilities, authorities, interfaces, and means of communications for configuration management, in particular change control, during construction, both within DCS and for subcontractors, including those for fabrication, assembly and construction.

SRP Section 4.3.A.ii. states that the applicant's organization should commit to formal management measures and that the organization and plans for transition from design and construction to operation should be adequate to maintain the design bases at all times.

Response:

Configuration management is discussed in CAR Section 15.2, and change control during construction is discussed in CAR Section 15.2.4.2. Because DCS is responsible for design, construction, and operation of the facility, a single technical baseline can be maintained throughout the three phases.

Configuration management of this baseline during the design and construction phase is governed by QA procedures, and is the responsibility of the Deputy Project Manager - MFFF Engineering and Construction. Interface and methodology for change control is delineated in QA procedures, and will be required for fabricators and subcontractors through specifications and procurement documents. Control of subcontractors is also the subject of the responses to MPQAP RAI Questions 3 and 10 (submitted by DCS to the NRC 18 July 2001; Letter Number DCS-NRC-000054).

Project procedures will be adapted to implement the configuration control requirements of 10 CFR 70.72 as the project progresses through the construction an operation phases. Additional information is contained in the response to Question 239.

Action:

Update CAR Chapter 4 to clarify DCS commitment to configuration management and refer to Chapter 15 for details of the configuration management process.

35. Chapter 4, pp. 4-1 thru 4-6

Please explain the organizational responsibilities, authorities, interfaces, and means of communications with DOE and Savannah River Site contractor organizations that may affect principal SSCs.

SRP Section 4.3 states that the applicant's identification and functional description of the specific organizational groups responsible for designing and constructing the facility should include.....other outside service organizations in addition to the applicant. SRP Section 4.3.A.ii. states that the applicant's organization should commit to formal management measures and that the organization and plans for transition from design and construction to operation should be adequate to maintain the design bases at all times.

The organizational responsibilities, authorities, interfaces, and means of communications with DOE and Savannah River Site contractor organizations that supply, maintain, perform or provide services, systems or utilities that may affect principal SSCs and the responsibilities to assure that management measures are adequate, including configuration management, in particular change control, and procedures should be addressed.

Response:

Organizational Responsibilities, Authorities, Interfaces

DCS is the licensee responsible for producing the detailed design of the MFFF and for operation of the facility. The management organization of DCS is addressed in the response to Question 37. DCS establishes the interface points and requirements with Savannah River Site (SRS) and ensures that the MFFF design is integrated into the SRS infrastructure. Configuration management of the integrated DCS design is addressed in the response to Question 239.

The National Nuclear Security Administration (NNSA) facilitates the interface with DOE's Savannah River Operations Office (DOE-SR) and other SRS organizations such as the U.S. Forest Service, the security contractor, and other service providers to DOE-SR such as SCE&G. NNSA and DOE-SR also provide interfaces with respect to integration with SRS' emergency planning and response programs, worker training (as discussed in the responses to Questions 1 and 2), and control of the SRS site (i.e., facilitating DCS' ability to remove or exclude personnel and property from the MFFF controlled area as necessary). In conjunction with NNSA and DOE-SR, the SRS M&O contractor (Westinghouse Savannah River Company, or WSRC) is responsible for providing the utilities required to operate the MFFF and for receiving and treating the waste generated by the MFFF, and provides various support to DCS in areas such as site infrastructure, utilities, waste management, emergency services, site transportation, security, and training. Interfaces between DCS and NNSA (along with DOE-SR) are controlled via DCS' contract, administered through DOE's Chicago Operations Office (DOE-CH), with technical direction provided by the NNSA office in DOE Headquarters. The interfaces between DCS and WSRC are described through a Work Task Agreement (WTA) administered by the NNSA (along with DOE-SR). During the design phase of the MFFF project, a master WTA is agreed upon and signed by each of the organizations having responsibilities under the agreement. The WTA is updated based on the need for information or interfaces, and will continue through construction and operations.

The WTA addresses utilities (i.e., telephone, electricity, domestic and fire water, etc.), emergency medical and health physics services, security (i.e., outside the Protected Area), emergency response, site infrastructure, environmental services (e.g., monitoring, lab analysis, etc), and transportation and disposal of low level, hazardous, non-hazardous, mixed and TRU waste.

Action:

Clarify Chapter 4 to describe site interfaces.

36. <u>Chapter 4, pp. 4-1 thru 4-11</u>

Indicate responsibility for fire protection.

SRP Section 7.4.3.1 recommends that responsibilities for fire protection management, implementation and assessment be identified. The application does not show how fire protection is integrated into the Environmental Safety and Health oversight or construction design reviews.

Response:

Evaluation of fire risk and design of engineered controls for fire protection is provided by the Facilities Design organization under the direction of the Facilities Design Manager. Fire protection evaluations and engineered controls are developed through a cross-discipline review process involving the Facilities Design organization, Process Design organization, safety assessment (i.e., ISA) team, and ES&H staff as appropriate. The fire analysis results are reflected in the design and integrated with other elements of safety through the ISA process.

As stated in CAR Section 4.1.2.3, Environment, Safety and Health (ES&H) management is responsible for establishment of top-level project ES&H requirements and oversight of integration of ES&H requirements for nuclear safety, radiation protection, environmental protection, and industrial safety, which includes fire protection. Further details of the organization for operations, which includes fire protection management, implementation, and assessment during operations, will be provided as part of the license application.

Action:

None

37. Figures 4-1 and 4-2, pp. 4-9 thru 4-11

Provide greater specificity in the management organization plan.

Section 4.4.3.A.iii.a of the SRP states, for key management positions, "The personnel to design and construct the facility have the appropriate breadth and level of experience for their respective authorities and responsibilities, as indicated in the organizational structure ..."

[A similar requirement exists for operations. Section 4.4.3.B.iii.a of the SRP states, for key management positions, "The personnel to operate the facility have the appropriate breadth and level of experience for their respective authorities and responsibilities, as indicated in the organizational structure ..."]

Figures 4-1 and 4-2 provide organizational charts. The NRC would anticipate that Figure 4-1 would be more explicit vis-a-vis positions, the location of the Safety Assessment manager and safety activities, and current design and regulatory activities. Current names and qualifications of individuals should be supplied. In addition, on Figure 4-2, the NRC would anticipate, there should be a link between regulatory manager and corporate management, including corporate safety management.

Response:

The Question 33 response refers to detailed organization charts depicting the present project organization and the proposed construction management organization. The project organization is the intended organization for initial construction activities. As the project progresses, individual organizations will change in numbers and reporting relationships to best show the emphasis for that particular phase of the project. The operations elements of the project will become the predominant organization as the project approaches the start-up phase. Safety assessment activities are presently under a manager reporting to the Facilities Design Manager since they are closely related to the development of the design. The regulatory activities are presently managed by the Licensing and ES&H organizations under the Deputy Project Manager for Technical and Project Integration. These organizations will also develop new reporting relationships to best meet the needs of each phase of the project.

The current incumbents for the positions shown on the project organization are as follows. Their resumes are also included in this response.

Project Manager – Bob Ihde
Deputy Project Manager – Technical & Project Integration – Tommy Touchstone
Deputy Project Manager – Engineering & Construction – Ed Brabazon
Quality Assurance Manager – Jim Brackett
Project Services & Administration Manager – Naresh Jain
Licensing Manager – Peter Hastings
Procurement Manager – Mitch Laney
ES&H Manager – Mary Birch
MFFF Plant Operations Manager – John Matheson

Process Design Manager – Jean-Marc Belmont
Facilities Design Manager – Dick Berry
Site Engineering Manager – Jack Clemmens
Equipment Design Manager – Alden Segrest
Construction Manager – Joe King
Manufacturing Design Manager – Remi Bera
Software Design Manager – Gary Bell

Additional management personnel not associated with MFFF IROFS functions include:

Project Controls Manager – Frank Kania
Fuel Qualification Manager – George Meyer
Irradiation Services Manager – Richard Clark
Packing & Transportation Manager – Richard Clark (acting)
Communications Outreach – Todd Kaish

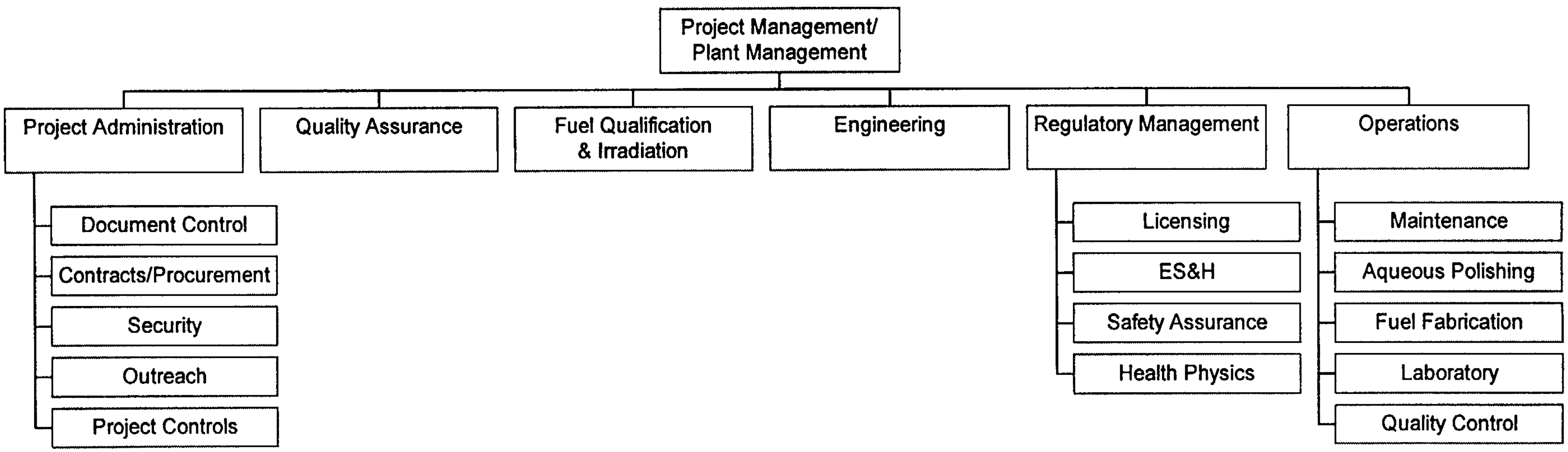
Figure 4-2 has been updated to show the functional relationships of the organization for the operations phase of the project. This organizational chart (see attached sketch) details the proposed DCS organization for the operations phase of the project. The organization for the initial operation phase will include the engineering, construction, and start-up elements until final turnover to operations is complete.

DCS was formed for the purposes of the MOX project, and the MOX Fuel Fabrication Facility is a primary interest; thus, the plant organization essentially constitutes DCS' corporate organization structure during this phase. Therefore, the functions such as the Regulatory and QA functions that require direct access to "corporate" management (i.e., "corporate safety management," as stated in the question) have that access through the plant/project management function as shown in Figure 4-2. The project is also accountable to a joint Board of Governors and has utilized and will continue to utilize the member companies for additional reviews and lessons learned. Areas utilized include quality assurance, design control, safety assurance, regulatory management, and project management.

Action:

In the next update of the CAR, replace Figures 4-1 and 4-2 with the new project, construction, and operations charts; add position descriptions for the additional positions indicated above; and clarify reporting relationships as discussed above.

DCS Operations Phase



Robert H. (Bob) Ihde

Duke Cogema Stone & Webster (DCS)
Project Manager
M.B.A. Kent State University
B.S. Nuclear Engineering
M.S. Coursework in Nuclear Engineering

DUKE COGEMA
STONE & WEBSTER

28 Years Experience

Summary

Bob Ihde offers 28 years of leadership in nuclear fuel management. His diverse experience extends across domestic and international business and high technology projects of national significance ranging from the DOE Naval Nuclear Program and nuclear fuel fabrication to commercial nuclear power plants. He has led workforces of 400 personnel and managed annual budgets of $60 million.

Project Manager, Duke Cogema Stone & Webster
Primary Responsibilities

- Provide overall strategic direction, leadership, management, and integration of all employees, subcontractors, and project activities.
- Establish and implement project oversight policies and procedures for quality assurance; environment, safety, and health requirements; technology integration; and site liaison.
- Plan, procure, integrate, expedite, and oversee project activities to safely and cost-effectively complete the work on time and within budget.
- Establish and maintain effective communication with DOE and identify organizational interfaces to ensure clear lines of communication.
- Manage the project in compliance with applicable NRC regulations and DOE orders, contract terms, and all applicable environment, safety, and health regulations.
- Build productive partnerships with DOE; the host site and local communities; the IAEA; mission reactor liaisons; the public; regulators; subcontractors; and employees; and support DOE's outreach program.

Professional Development

EEO in the 90s
Quality Improvement and Cost Reduction

Clearances

DOE Q (Inactive)

Previous Work Experience

President and Chief Operating Officer, COGEMA Technologies, Inc., Bethesda, MD, 1995-1998

- Directed all aspects of company operations, including contracts, marketing, and quality assurance

President, B&W Fuel Co. (now a part of Framatome ANP), Lynchburg, VA, 1987-1994

- Directed all aspects of company operations, including contracts, marketing, and quality assurance

General Manager, Nuclear Fuel Services, Babcock & Wilcox, Lynchburg, VA, 1980-1987

- Led the Nuclear Fuel Services Business Unit, comprised of 400 personnel and three manufacturing facilities.

Manager, Consumers Power Project, Babcock & Wilcox, Lynchburg, VA, 1979-1980

- Led NSSS project management team to support the two-unit site in Midland, MI.

Manager, Contracts and Marketing Dept., Babcock & Wilcox, Lynchburg, VA, 1974-1979

- Managed contracts and marketing projects for the Nuclear Materials Division with DOE and the U.S. utilities as the ultimate customers.

International Marketing Manager, Babcock & Wilcox, Lynchburg, VA, 1972-1974

- Provided marketing and agreement support for penetration of the European market.

Thomas E. Touchstone, PE

Duke Cogema Stone & Webster (DCS)
Deputy Project Manager – Technical & Project Integration/Executive Vice President/ Chief Operating Officer
B.S., Civil Engineering

DUKE COGEMA STONE & WEBSTER

34 Years Experience

Summary

Mr. Touchstone's engineering and construction expertise includes project management for nuclear and conventional electric generation facilities, industrial projects, and a high-level waste repository exploratory facility. He has performed design, licensing, construction, procurement, start-up, planning and scheduling, budgeting, maintenance, quality assurance, quality control, and regulatory compliance for commercial and Department of Energy (DOE) projects. Mr. Touchstone has also managed maintenance and modification activities on operating facilities. He has developed and implemented Integrated Safety Management on a DOE program. He has performed strategic planning and risk management activities to assure project success.

Deputy Project Manager-Technical & Project Integration/ Executive Vice President/Chief Operating Officer Duke Cogema Stone & Webster 03/01 to Present

Primary responsibilities:

- Support the DCS Project Manager in managing the day-to-day technical and project integration activities of the DCS Project.
- Achieve excellence in ES&H compliance.
- Identify and oversee the cross-coordination functions of the project.
- Coordinate DCS licensing efforts with the NRC for the MOX Fuel Fabrication Facility, fuel qualification project and mission reactors to ensure consistency.
- Serve as Chairman of the Duke Cogema Stone & Webster Project Change Control Board.
- Maintain project technical, cost, and schedule baselines.
- Monitor project elements to ensure consistency with the approved budget and efficient relationships among project elements.
- Serve as Executive Vice President and Chief Operating Officer of Duke Cogema Stone & Webster.

Professional Development

Registered Professional Engineer: N.C., S.C.
Various Duke Power Company Management Development Training

Clearances

DOE Q Clearance (Inactive)
DOE L Clearance (Active)

Previous Work Experience

Project Manager DE&S 2000-2001
- Project Manager of ORNL Highly Enriched Uranium Materials Facility.

Project Manager DE&S 1999-2000
- Special Assignment to develop the Integrated Management System for the Yucca Mountain Project (YMP).

Deputy Manager - Site Construction and Operations DE&S 1995-1999
- Managed YMP construction, drill and sample facility management, test coordination, operations, and project controls for the DOE's M&O contractor. Proj. Mgr for all DE&S work on project.

Co-Owner Genesis Environmental, Inc 1992-1995
- Co-owner of environmental and occupational safety training company. Responsible for marketing, business and program development and training.

Project Manager Morris Construction Company, Greenville, S.C. 1991-1992
- Provided proj. mgmt. for industrial/ commercial construction projects.

Construction Manager, Tyger Construction 1989-1990
- Managed overall business development and marketing activities for commercial, industrial and power projects throughout the Southeast.

Construction Manager, Central Division Duke Power 1986-1989
- Managed nuclear, fossil and hydro power plant modification and maintenance activities for the Central Division.

Management & Technical Services Manager Duke Power 1986
- Managed construction services to other utilities and industrial clients through the Management and Technical Services (MATS) Division, the forerunner of Duke Engineering & Services (DE&S).

Construction & Engineering Manager Duke Power1967-1984
- Managed craft, engineering and planning work for the completion, start-up, maintenance and modification of the Oconee, McGuire and Catawba nuclear stations.

Edward J. Brabazon

Duke Cogema Stone & Webster (DCS)
Deputy Project Manager—Engineering and Construction
B.S., Mechanical Engineering

DUKE COGEMA
STONE & WEBSTER

36 Years Experience

Summary

Prior to joining DCS on the MOX Project, Mr. Brabazon was Manager of Projects overseeing Stone & Webster's design and construction projects for DOE, including: the design of the Actinide Packaging and Storage Facility at the Savannah River Site; Rocky Flats Engineers and Constructors; and plutonium stabilization and packaging designs for the Rocky Flats and Savannah River sites. Prior to that he was the Chief Engineer and directed engineering on the Modular High Temperature Gas-Cooled New Production Reactor for DOE, a large, multi-company project for the production of tritium, and participated in engineering analysis for the Plutonium Consumption Reactor, a MOX fuel project for dispositioning surplus weapons-grade plutonium for DOE.

Deputy Project Manager—Engineering & Construction, Duke Cogema Stone & Webster
Primary Responsibilities

- Direct the engineering and design activities for the MFFF, including conceptual, preliminary, and final design, including establishment of the overall plant requirements and system functional requirements, and configuration management.
- Implement engineering and quality assurance procedures for the execution of the design work.
- Develop and implement a process-design technology-transfer process from COGEMA to the U.S.
- Prepare technical input for license and permit applications.
- Ensure that technical, cost, and schedule baselines are controlled for the facility.
- Prepare the cost estimate for construction, start-up, and deactivation of the MFFF.

Professional Development

Registered Professional Engineer, 1970: MA #24028, PA #015394 (inactive), NY #045663-1 (inactive)

Clearances

DOE L (active)

Previous Work Experience

Manager of Projects, DOE Facilities, Stone & Webster, Denver, CO, 1996-1998

- Management oversight responsibility for all design and construction projects for DOE, involving management and technical reviews of the project engineering, including plutonium facility designs.

Deputy Director Nuclear Operations, Stone & Webster, Boston, MA, 1995-1996

- Oversight responsibility for the design and construction management of projects in the commercial nuclear power sector.

Chief Engineer MHTGR New Production Reactor, Stone & Webster, San Diego, CA, 1991-1995

- Directed 750 engineers of a consortium of four companies designing the Modular High Temperature Gas-Cooled Reactor (MHTGR) project.

Deputy Director, Comanche Peak Response Team, Stone & Webster, Glen Rose, TX, 1986-1988

- Directed a group of 200 engineers and quality assurance personnel performing a third party review to determine the quality of construction for the Comanche Peak Nuclear Power Plant.

Project Manager, Stone & Webster, Shoreham Nuclear Power Plant, 1982-1985

- Project Manager on the Shoreham Nuclear Power Plant responsible for engineering, construction assistance, cost and schedule and quality assurance.

Ralph James Brackett

Duke Cogema Stone & Webster (DCS)
Quality Assurance Manager
B.S., Metallurgical Engineering
M.S., Materials Engineering

DUKE COGEMA
STONE & WEBSTER

28 Years Experience

Summary

Mr. Brackett has 28 years of nuclear experience, including 16 years of commercial nuclear utility experience at Duke's Oconee Nuclear Station, in developing and implementing nuclear program processes, plans and procedures. He also has 2.5 years experience in managing the DE&S Richland, WA engineering office while also serving as the DE&S Project Manager on the Tank Waste Remediation System Privatization (TWRS-P) Project at Hanford. He has extensive experience in developing and implementing 10CFR830.120, 10CFR50, Appendix B, NQA-1 based nuclear QA programs, including: managing Duke Engineering & Services (DE&S) QA activities worldwide; DOE's Office of Civilian Radioactive Waste Management's M&O Contractor (TRW) QA Program; QA for the Utility Engineering Group in support of DOE's New Production Reactor Project; QA Program for the design, licensing, procurement, construction, testing, operation, and maintenance of a dry fuel storage facility in the Ukraine; and QA for the Louisiana Energy Services (LES) Clairborne Enrichment Center. Mr. Brackett is presently serving as the QA Manager for the Duke Cogema Stone & Webster (DCS) team on the Mixed Oxide (MOX) Fuel Project.

Quality Assurance Manager, Duke Cogema Stone & Webster 03/99 to Present

- Coordinate and direct all project QA activities for the MOX Fuel Project.
- Develop, implement and maintain the MOX Project Quality Assurance Plan (MPQAP).
- Coordinate development, review, approval and implementation of DCS QA project procedures.
- Serve as management liaison with NRC, DOE, state agencies, and other project stakeholders on QA requirements and concerns.
- Serve on the DCS Project Change Control Board.
- Monitor and evaluate QA compliance and effectiveness.

Professional Development

Duke Power Company Advanced Management Development Training
QA Lead Auditor Training
INPO Observation Training
DE&S Advanced Project Management "Boot Camp"

Clearances

DOE Q and L (inactive)

Previous Work Experience

Richland DE&S Engineering Office Manager, Richland, WA /DE&S Project Manager for TWRS-P, Hanford 1996-1999

- Established DE&S engineering office providing engineering support to nuclear projects in the Pacific Northwest and managing DE&S activities on the Hanford TWRS-Privatization Project.

Quality Assurance Manager, DE&S, Charlotte, NC 1994-1996

- Managed Duke Engineering & Services QA activities worldwide.

QA Manager DOE OCRWM M&O Contractor, Vienna VA 1991-1993

- Managed development and implementation of the M&O QA Program for the Monitored Retrievable Storage (MRS), Multi-Purpose Canister (MPC), Mined Geological Disposal System (MGDS) and the spent fuel transportation system.

QA Manager, DE&S Utility Engineering Group for the DOE New Production Reactors Project, Charlotte, NC/Washington DC 1989-1991

- Supported DOE in developing and implementing the NPR QA Program.

QA Director, Operations/Station QA Manager/Sr. QA Engineer Oconee Nuclear Station, Seneca, SC 1973-1989

- Managed all operational quality assurance activities under the Duke Power Company QA Program at Duke's Oconee Nuclear Station.

Naresh C. Jain

Duke Cogema Stone & Webster (DCS)
Project Services and Administration Manager
Ph.D., Management
M.B.A., McColl School of Business
B.S., Electrical Engineering
B.S., Communications Technology

DUKE COGEMA
STONE & WEBSTER

28 Years Experience

Summary
Dr. Naresh Jain has over 28 years of progressive experience in Design Engineering Management, Federal Contracting, Government Procurements, Accounting & Finance, Safeguards & Security, Human Resource Management, Training and Government Cost Accounting Systems for the Departments of Energy and Defense (DOE and DOD), and the National Aeronautics Space Administration (NASA). He is responsible for negotiating team subcontracts and providing advice to the Executive Vice President on a variety of strategic business issues.

Project Services & Administration Manager, Duke Cogema Stone & Webster

Primary Responsibilities

- Management of DOE's Prime Contract
- Integration with prime of the 1st and 2nd tier subcontracts and purchasing orders, teaming Agreements, Contractor Integration, Government Property Controls, Small, Disadvantaged and Woman Owned Business Subcontracting Plans.
- Management of the LLC employment process, Human Resources activities, Workforce training and other corporate services such as Insurance, Legal Counsel, and Audits.
- Project Direction on Technology and Security matters, plans and procedures designed to mitigate certain foreign ownership and control interests (FOCI) related to U.S. Export Control laws.
- Direct efforts associated with the establishment and maintenance of a MOX Safeguards and Security Program compliant with the Department of Energy Protective Needs and National Security interests.
- Programmatic Compliance pursuant to Executive Order 12829, National Industrial Program (NISP) as applicable to the MOX project.
- Serve as an officer of the LLC with titles: Corporate Secretary,

Professional Development

Registered Professional Engineer, NC #15961, TN #16651
Certified NQA Auditor, DE&S Nuclear QA Program

Clearances

Active DOE Level L #FC15337
Facility Security Officer—DE&S Classified Facility

Previous Work Experience

Director, Federal Contracts, Duke Engineering & Services, Inc., Charlotte, NC, 1989-1998
- Contract direction, project management, project planning, cost control, and problem solving for OCRWM, INEEL and other M&O projects under various Government contracts to DE&S Federal Group.

Project Engineer, Duke Engineering & Services, Inc., Charlotte, NC, 1985-1989
- DE&S client relations, contract administration, invoice operation, project control, and cost benefit analysis for assigned projects.

Senior Engineer, Duke Power Co., Knoxville, TN, 1982-1985
- AFBC project, EPRI liaison, electrical design lead, cost control and project reporting for 160MW Fluid Bed Research Project.

Design Engineer, Duke Power Co., Charlotte, NC, 1977-1982
- Engineering lead for hardware system design of electromechanical systems for McGuire and Catawba Nuclear Stations.

Assistant Design Engineer, Duke Power Co., Charlotte, NC, 1973-1977
- Equipment specification writing and P&IDs for control systems using digital design techniques for Oconee Nuclear Station located in Seneca, South Carolina.

Peter S. Hastings

Duke Cogema Stone & Webster (DCS)
Licensing Manager
B.S. Nuclear Engineering

18 Years Experience

DUKE COGEMA
STONE & WEBSTER

Summary

Mr. Hastings has over 18 years of commercial and DOE engineering and management experience in the nuclear industry. As Licensing Manager for the Mixed Oxide Fuel Fabrication Facility, he has participated in the rulemaking and development of regulatory guidance for the recent change to 10 CFR Part 70 and oversees their implementation for the construction authorization and possession-and-use license for the facility. Prior to his current assignment, he established and oversaw systems and processes required for licensing of the nation's first high-level radioactive waste repository, and managed nuclear safety analyses and long-term performance assessment (PA) for the repository. He established the NRC licensing and programmatic basis for the repository's pre-closure nuclear safety and accident analysis program, and established the DOE's program for assessing long-term performance impacts during site characterization to meet NRC requirements. He has coordinated aspects of the quality assurance of the repository PA program, process improvements to repository engineering and licensing systems, developed PA input to the repository license application, and handled resolution of NRC Key Technical Issues. In addition, prior to his repository work, he had several years of nuclear station operations, startup testing, and surveillance experience, and design engineering experience.

Licensing Manager, Duke Cogema Stone & Webster
Primary Responsibilities

- Coordinate and develop the license application for the MOX Fuel Fabrication Facility (MFFF).
- Manage regulatory processes associated with the MFFF and serve as interface with regulatory agencies regarding MFFF activities.
- Ensure that NRC submittals for the MFFF license are of high quality and are made in a timely manner.
- Establish a close working relationship with the NRC staff to facilitate timely technical review and issue of the license.

Professional Development

Registered Professional Engineer, NC #18204, SC #13891
DOE-STD-3009—Nonreactor Nuclear Facility Safety Analysis
Computational Methods in Reactor Analysis, Univ. of Tennessee
B&W Integrated Control System

Clearances

DOE L

Previous Work Experience

Deputy Operations Manager, Duke Engineering & Services, Inc., Las Vegas, 1997-1998

- Served as Assistant Manager for Performance Assessment as part of M&O contract for DOE's Office of Civilian Radioactive Waste Management (OCRWM).
- Managed the post-closure licensing basis for the high-level radioactive waste repository.

MGDS Safety Assurance Manager, Duke Engineering & Services, Inc., Las Vegas, NV, 1992-1997

- Coordinated definition and analysis of design-basis events, including accident analyses for the repository License Application.
- Responsible for compliance with NRC requirements for limiting impacts to the high-level radioactive waste repository during site characterization.

Design Engineer, Duke Engineering & Services, Inc., Charlotte, NC, 1991-1992

- Served as a key member of the monitored retrievable storage (MRS) design team for the OCRWM.

Reactor Engineer, Duke Power Co., Seneca, SC, 1984-1991

- Senior staff engineer and work leader in the Reactor Engineering Unit at Oconee Nuclear Station (900 Mwe B&W PWR). Primary duties included reactor and primary system surveillance and testing and fuel performance monitoring.

Mitchell L. Laney

Duke Cogema Stone & Webster (DCS)
DCS Procurement Manager
B.S., Management
A.S., Contracts Management

25 Years Experience

DUKE COGEMA STONE & WEBSTER

Summary

Mr. Laney has 25 years Federal experience in Procurement, with 16 of those years in the DOD arena and the remainder with DOE National Laboratories. During his military career, he worked at all levels of Procurement beginning as a Buyer and Contracts Administrator and then as a section head over a Contract Repair Branch. He later worked as a Staff Officer at the Headquarters level in the Contract Review Division. After a successful tour on staff, he was given his own Procurement Squadron to command prior to military retirement. Once leaving the Air Force, he began working as a Senior Subcontracting Officer at Jefferson Laboratory responsible for buying services, supplies, and construction. He then took an assignment to Oak Ridge National Laboratory working on the Spallation Neutron Source Project as a Supervisor for the Architect Engineer/Construction Manager. Mr. Laney is presently serving as the Duke Cogema Stone & Webster (DCS) Procurement Manager for the Long Lead Equipment purchases on the Mixed Oxide Fuel Project.

DCS Procurement Manager, Duke Cogema Stone & Webster
02/01 to Present

- Develop and manage a Procurement Organization.
- Prepare and award all solicitations for Long Lead Procurements for the MOX Fuel Fabrication Facility.
- Coordinate and work closely with DCS Engineering Staff on specifications and drawings.
- Work Closely with the DCS Contracts Manger.
- Prepare and implement all terms and conditions for DCS.
- Establish pre-qualification procedures for Vendors.
- Assist the DCS Director of Procurement in efforts to comply with Small Business Goals.

Professional Development

UT-Battelle Leadership Enhancement and Development Course (LEaD) Program
Writing Statements of Work Training

Clearances

None

Previous Work Experience

SNS AE/CM Procurement Supervisor, Oak Ridge National Laboratory – Ut-Battelle, Oak Ridge, TN, 1999 - 2001
- Supervised Construction Manager for Spallation Neutron Source (SNS) Project valued in excess of $1.4 billion dollars.

Subcontracting Officer, Jefferson Laboratory, Newport News, VA, 1992 - 1999
- Served as senior subcontracting officer for $28 million in solicitations and subcontracts for construction, services and supplies.

Commander, Moody AFB, GA, 1988 - 1992
- Served as Commander for a Procurement Squadron supervising 33 personnel and ensured over $91 million was obligated during leadership.

Acquisition Staff Officer, Langley AFB, VA, 1984 - 1988
- Responsible for reviews on Base contracts and providing advisory services to Procurement Squadron Commanders.

Buyer & Administrator, Various Air Force Bases, 1975 - 1984
- Responsible for purchasing and administering over $25 million in base solicitations and subcontracts.

Mary L. Birch

Duke Cogema Stone & Webster (DCS)
Environment, Safety, & Health Manager
M.S., Radiation Sciences
B.S., Chemistry

DUKE COGEMA
STONE & WEBSTER

34 Years Experience

Summary

Mary Birch has 34 years of environment, safety, and health experience. She has managed radiation protection, environmental protection, industrial safety, and hygiene programs as well as nuclear assessments and audits for commercial nuclear facilities. She also has experience in managing multi-tiered technical organizations and annual budgets up to $7 million. Her responsibilities have ranged from safety and health program restructuring to site-wide safety assurance management. Ms. Birch has successfully interfaced with regulatory agencies, held responsible committee positions in industry groups, and served as an expert witness.

ES&H Manager, Duke Cogema Stone & Webster

Primary Responsibilities

- Serve as management liaison with regulatory agencies, including DOE, NRC, South Carolina Department of Health and Environmental Control, and OSHA on ES&H requirements.
- Establish MFFF ES&H policy. Implement policy in an environmental protection program, industrial safety and hygiene program, radiation and criticality safety program, and emergency response program.
- Work closely with line management to ensure that ES&H requirements are consistently interpreted and integrated into activities across the project.
- Provide consultation and direction to project staff in planning and conducting work activities safely.
- Serve as a member of accident and near-miss investigation processes.
- Ensure compliance with ES&H requirements and resolution of concerns.
- Ensure effective selection and management of ES&H professional resources.

Professional Development

Registered Profession Engineer NC #10451
Certified by the American Board of Health Physics

Clearances

Duke Nuclear Station
DOE BAO Security Clearance (Inactive)
DOE Savannah River Site Access

Previous Work Experience

Manager, Safety Assurance Catawba Nuclear Station, Duke Power Co., York, SC, 1997-1998

- Managed the Safety Review, Regulatory Compliance, Environmental Management, Safety Services, and Emergency Planning groups (37 personnel).
- Interfaced with NRC, EPA, OSHA, DOT, state and county government agencies, INPO, and nuclear insurers.

Consultant, ATL International, Inc. (On a senior advisory panel to DOE-EM 30), Germantown, MD, 1996-1998

- Provided advice to EM-30 on DOE O 435.1 development.

Project Manager/Engr. Supervisor/ Acting Manager Catawba Nuclear Station, Duke Power Co., York, SC, 1994-1997

- Supervised the activities of site/contractor engineering personnel working on instrumentation and control equipment.
- Resolved complex equipment problems and component failures.

Regulatory & Licensing Manager, DOE OCRWM M&O Contractor, Duke Engineering & Services, Inc., Vienna, VA, 1991-1994

- Supervised Regulatory and Licensing personnel in support of repository and MRS facilities.

Technical Corporate System Manager, Radiation Protection/RadWaste Engineering, Duke Power Co., Charlotte, NC, 1982-1991

- Supervised personnel maintaining existing health physics and radioactive waste management programs

John E. Matheson

Duke Cogema Stone & Webster (DCS)
MFFF Plant Operations Manager
M.S., Mechanical Engineering
M.S., Aerospace Engineering
B.S., Physics

DUKE COGEMA
STONE & WEBSTER

33 Years Experience

Summary

John Matheson has 33 years of experience mainly in nuclear fuel assembly and related component design and manufacturing. He has provided contract management for fuel delivered to Duke Power's seven nuclear reactors. He managed the fuel consolidation project for FCF. He has been responsible for the design and manufacture of nuclear fuel assemblies, components, and in-core detectors. Mr. Matheson was also responsible for materials management, facilities management, and scheduling and cost management for FCF's Lynchburg Manufacturing Facility. In his most recent assignment he is responsible for the operation of both the Richland and Lynchburg Fuel Operations as well as Engineering, Quality and R&D for the Fuel Business Group.

MFF Plant Operations Manager, Duke Cogema Stone & Webster
Primary Responsibilities

- Direct and coordinate operability reviews during design.
- Provide operational input to design and licensing.
- Direct fabrication and development of lead assemblies.
- Develop operations procedures, training, and qualification programs for plant operations staff.
- Direct and participate in startup testing.
- Serve as primary interface with NRC during facility operations.
- Incorporate American fuel fabrication operating experience into MOX Fuel Fabrication Facility (MFFF) operations.
- Ensure design provisions that will facilitate compliant operation of the MFFF.

Professional Development

Registered Professional Engineer, VA #04020 12314

Clearances

DOE L (Inactive)
DOD Secret (Inactive)

Previous Work Experience

Vice President and Operating Director, Framatome ANP INC, Fuel Business Group, Lynchburg, VA, April 2001 – present

- Responsible for Operations, Quality, Engineering and R&D for both Richland and Lynchburg Operations of the Fuel Business Group of Framatome ANP Inc.

Vice President of Operations, Framatome Cogema Fuel Co., Lynchburg, VA, 1997-2001

- Responsible for fuel manufacturing, materials management, facilities management, and scheduling and cost management.

Manager of Design and Development, Framatome Cogema Fuel Co., Lynchburg, VA, 1994-1997

- Had lead responsibility for technical content and design basis of all fuel assemblies and components.

Contract Manager, Framatome Cogema Fuel Co., Lynchburg, VA, 1990-1994

- Managed the fuel contracts for fuel delivered to Duke Power's seven reactors. Responsible for defining contract needs and issuing release authorizations for manufacturing and design.

Project Manager, High-Level Waste, Framatome Cogema Fuel Co., Lynchburg, VA, 1987-1990

- Managed the fuel consolidation project for FCF.

Supervisor, Fuel Rod Design and Analysis, Framatome Cogema Fuel Co., Lynchburg, VA, 1985-1987

- Managed the design and analysis of fuel rods. Supervised seven engineers.

Principal Engineer, Fuel Design, Framatome Cogema Fuel Co., Lynchburg, VA, 1978-1985

- Performed various tasks in the design of fuel assembly components.

Jean-Marc Belmont

Duke Cogema Stone & Webster (DCS)
Process Design Manager

DUKE COGEMA
STONE & WEBSTER

30 Years Experience

Summary

In 30 years of experience, of which 23 years in nuclear industry, Jean-Marc Belmont has managed multicultural teams across a spectrum of disciplines: research, design, engineering, construction, and operation. He led the successful construction and startup of nuclear power plants in France and South Africa as well as a plutonium reprocessing plant. Mr. Belmont managed the 70 MThm/year MELOX Extension project, from design conception through operational testing. This was a state-of-the-art, third-generation MOX fuel fabrication facility. Mr. Belmont kept the project within budget and on schedule, even with an aggressive 3.5-year schedule for procurement, construction, and start-up.

Process Design Manager, Duke Cogema Stone & Webster
Primary Responsibilities

- Ensure design functionality of the plutonium polishing, powder, pellet fabrication, cladding and assembly, waste management, and laboratory areas.
- Implement process control design based on European technology, as proven in the La Hague and MELOX facilities operated by COGEMA.
- Prepare the startup plan under the direction of the Fuel Fabrication Services Manager.
- Oversee facility startup and transition to operation, with particular emphasis on European facility lessons learned.
- Ensure compliance with environment, safety, and health requirements.

Professional Development

Uranium/Plutonium Risk & Protection, Plutonium Academy, 1993
Business Management, South African Institute of Management, 1984
Neutronic and Nuclear Physics, Nuclear Science & Techniques, National Institute, France, 1980

Clearances

French Confidential Defense and Secret Defense

Previous Work Experience

Manager, Melox Facility Ext. Project & SGN's South East Eng. Div., COGEMA, Engineering Group, France, 1995-1998

- Managed a 600-person engineering division that covered enrichment plants, cleanup activities, and construction and startup of new facilities. This includes hot laboratories, hot cells, and fuel storage waste treatment.

Project Manager, SGN, La Hague, France, 1994-1995

- Directed the $100 million project to recover plutonium and uranium from incinerator ash and waste from the reprocessing plant and to dechlorinate filter dust before vitrification.

Head of Construction Division/Project Manager, SGN/COGEMA Engineering Group, La Hague, France, 1992-1994

- Managed the construction of a shearing and dissolution facility and a plutonium redissolution facility, and $835 million project with a staff of 300.

Chief Executive Officer, FOUGEROLLE Group, Lyon, France, 1990-1991

- At FOUGEROLLE, managed two subsidiaries specializing in HVAC and fluid transfer.

Manager, Development & Construction Unit, Framatome Group, Chalon sur Saone, France, 1985-1990

- Managed engineering for nuclear power plant maintenance, leading multi-disciplinary teams of 350 engineers.

Richard A. Berry

Duke Cogema Stone & Webster (DCS)
Facility Design Manager
B.S., Nuclear Engineering
M.S., Mechanical Engineering

DUKE COGEMA
STONE & WEBSTER

30 Years Experience

Summary

Mr. Berry has 18 years of A-E design experience in the commercial nuclear power industry. During his career he has assumed increasing levels of responsibility. He has been assigned as Principal Nuclear Engineer, Lead Power Engineer, Assistant Project Engineer and Project Engineer on various nuclear power projects. Since 1989 he has primarily worked in the DOE complex. He has performed individual engineering consulting and project engineering management at Savannah River, Rocky Flats, Hanford and at the Lawrence Livermore National Laboratory.

Prior to joining Stone and Webster in 1971, he served as an officer in the US Navy.

Facility Design Manager, Duke Cogema Stone & Webster
2000 to Present
Primary Responsibilities:

- Establish the facility safety strategy and design criteria necessary to meet 10CFR70
- Develop unique facility design features to satisfy safeguards and security requirements.
- Integrate the facility design with the Savannah River Site and the Pits Disassembly and Conversion Facility.
- Integrate process design requirements into the overall facility design.
- Direct the development of all design documents leading to facility construction specifications and drawings.
- Coordinate facility design input into the construction cost estimate.
- Provide technical input and support to the NRC licensing effort.

Professional Development

Registered Professional Engineer: Massachusetts, South Carolina
Authorized Derivative Classifier
Reviewing Official

Clearances

Active L Clearance
Inactive DOE Q Clearance

Previous Work Experience

Assistant Facility Design Manager, Duke Cogema Stone and Webster, 1999 – 2000

- Directed technical development of structures, systems and components required to support the process design of the MOX Fuel Fabrication Facility

Project Engineering, Stone and Webster, Denver, Co, January 1996 – 1999

- Managed a multi company design team engaged in the production of the conceptual, preliminary and final design of the Actinide Packaging and Storage Facility located at the Savannah River Site. The client for this contract was Westinghouse Savannah River Company.

Project Engineer, Stone and Webster, Denver, Co., 1995 – 1996

- Managed the Stone & Webster staff preparing system design criteria, facility integration specification, criticality, radiation shielding, time and motion, ALARA, fire hazards analysis and control system design for the Plutonium Stabilization and Packaging System. The client for this contract was BNFL.

Project Engineer, Stone &Webster, Lawrence Livermore National Laboratory – Various Projects, 1993 - 1995

- Developed a pre conceptual design, cost estimate and construction schedule for the deployment of a U-AVLIS pilot plant.
- Completed a detailed study and design concept for handling and storing depleted uranium.

John P. Clemmens

Duke Cogema Stone & Webster (DCS)
Site Engineering Manager
Deputy Manager—MFFF Design
B.S., Civil Engineering
M.S., Civil Engineering

DUKE COGEMA
STONE & WEBSTER

27 Years Experience

Summary

Prior to joining DCS on the MOX Project, Mr. Clemmens was the site engineering manager for the Actinide Packaging and Storage Facility at the Savannah River Site (SRS). Previously, he was the project manager for Stone & Webster on two technical support contracts for DOE at SRS; the "K" Reactor restart and Defense Waste Processing Facility. Prior to that he managed the construction aspects of all corporate conceptual design projects. He also managed the successful construction completion of the civil/structural and mechanical aspects of the Nine Mile Nuclear Station- Unit 2 in Oswego, NY.

Site Engineering Manager
Duke Cogema Stone & Webster
Primary Responsibilities

- Provides primary interface between the Duke COGEMA Stone & Webster (DCS) engineering and construction team and the Savannah River Site.
- Develops all Work Task Agreements (WTA) between DOE, DCS and the M&O contractor at SRS. Key activities include site development, utilities, waste management, construction, and operations interfaces.
- Conducts MFFF related research and development tasks with Clemson University
- Assists the Deputy Project Manager- Engineering and Construction for managing the engineering and design activities for the MFFF.
- Responsible for staffing of the engineering effort to ensure that technical, cost, and schedule baselines are met for the facility

Previous Work Experience

Senior Project Manager
Actinide Packaging and Storage Facility
Stone & Webster
Aiken, SC, 1996-1998

- Provided interface between the Stone & Webster engineering team in the Denver office and the WSRC team at SRS. Ensured that SRS requirements were integrated into the Title I and II design. Managed the Title III engineering effort during the initial phases of project construction.

Project Manager
Stone & Webster
Savannah River Site, 1990-1995

- Directed all technical support contract activities for the "K" Reactor Restart effort and startup of the Defense Waste Processing Facility (DWPF). Principal areas included engineering and cost and scheduling reviews, and oversight of startup testing and facility operations

Construction Manager
Stone & Webster
Boston, MA, 1986-1989

- Manager of the construction conceptual design organization. Performed constructability reviews and prepared construction management plans, budgets, and schedules for all conceptual design projects.

Construction Manager
Stone & Webster
Nine Mile Nuclear Station- Unit 2
Oswego, NY, 1981-1985

- Managed the mechanical and civil/structural construction, including a work force of 115 non-manuals, 1,000 manuals and 800 subcontractor personnel.
- Responsible for safety, quality, cost and schedule

Professional Development

Registered Professional Engineer, 1981: PA, SC (active); MA, NJ(inactive)

Clearances

DOE L (active)

Alden M. Segrest

Duke Cogema Stone & Webster (DCS)
Manager, Equipment Design Group and Systems Engineering
B.S.M.E, M.B.A.

DUKE COGEMA
STONE & WEBSTER

31 Years Experience

Summary

Mr. Segrest has provided engineering management and leadership for major projects and large organizations. He has managed engineering activities to design an underground exploratory studies facility, a nuclear waste repository and a spent nuclear fuel disposal package. He managed design engineering work during final construction and startup of McGuire Nuclear Station. He has successfully managed engineering organizations in the highly regulated nuclear environment and in a competitive commercial environment.

Manager of Equipment Design Group and Systems Engineering, Duke Cogema Stone & Webster
11/99 to Present

- Design of MOX process and Aqueous Polishing gloveboxes
- Development of glovebox process and equipment design specifications.
- Manage engineering integration activities.
- Manage configuration management, design control and document control functions for engineering organization.
- Manage engineering integration activities including design reviews, action item lists, and other intergroup activities.
- Manage functions as required to establish and maintain technical baseline requirements, engineering processes and integration and certain administrative functions.
- Maintain Design Requirements Document to comply with requirements of the DOE contract, SAW and applicable regulatory requirements and provide high level resign requirements relevant to the MOX Fuel Fabrication Facility.

Professional Development

Professional Engineer in North Carolina and South Carolina

Clearances

DOE L (Inactive)

Previous Work Experience

Manager, Special Projects, Duke Engineering & Services, Inc. 1/2000-10/2000

- Managed assigned projects related to business improvement processes, development of new business and corporate strategy.

Engineering Manager, DE&S, 1999

- Responsible for developing and implementing analysis process improvements for performance assessment functions for high-level waste repository to achieve compliance with NRC and DOE requirements.

Chief Engineer and Director, DE&S, 1998

- Managed engineering and safety analysis functions for spent nuclear fuel storage facility at Hanford.

Monitored Geologic Repository Engineering Manager, DE&S, 1994-1997

- Managed Yucca Mountain Project engineering activities to design an underground exploratory studies facility, a nuclear waste repository, a spent nuclear fuel disposal package, and a monitored retrievable storage facility for spent nuclear fuel.

Project Manager, DE&S, 1981-1990

- Manager of all DE&S projects, including cogeneration plant design, consulting studies, nuclear plant outage support, and hydroelectric facility modernization projects.

Joseph M. King

Duke Cogema Stone & Webster (DCS)
Construction Manager

B.S., Mechanical Engineering

31Years Experience

DUKE COGEMA
STONE & WEBSTER

Summary

Joe King has 31years of experience and management achievements in the construction industry. He also has 23 years of experience in the government nuclear and commercial nuclear environment. He has been responsible for project management, construction management, and construction activities at government nuclear facilities, nuclear power and fossil power plants, and various industrial facilities. He has managed large construction projects and fixed-price subcontractors, and has supervised direct-hire crafts. His experience includes new facility construction, renovation, and expansion work for industrial and commercial projects involving multidisciplinary activities.

Construction Manager, Duke Cogema Stone & Webster
Primary Responsibilities

- Manages all Construction Management and construction related activities at the MFFF
- Provides construction planning, estimating, scheduling, long lead procurement and other preconstruction activities
- Develop a close working relationship with procurement and engineering for timely release of the construction bid package; participate on the subcontractor selection board; and assure timely support for subcontractor mobilization.
- Assure early involvement with the project to provide constructability reviews and to identify opportunities for using advanced construction methods.
- Provides construction input to the construction acceptance testing and cold startup planning
- Assure adherence to construction quality plans.
- Verify that the constructor and subcontractor companies incorporate project safety and health standards in their processes.

Professional Development

Engineering in Training, Part I, Michigan
Construction Management Techniques and Procedures, Univ. of Wisconsin
Engineer As Manager, Battelle Institute
Contract Administration, San Francisco
Construction Contract Negotiating, Corps of Engineers

Clearances

Active DOE L

Previous Work Experience

Senior Site Representative, Rocky Flats Environmental Technology Site, Stone & Webster, Denver, CO, 1996-1998

- Directed all construction management and construction activities at the Rocky Flats Environmental Technology Site (RFETS).

Construction Manager, Savannah River Site, Stone & Webster, Aiken, SC, 1990-1996

- Provided project management services, including evaluation of radiological waste removal activities, tank farm management, high level waste vitrification management and engineering and construction activities in radiologically contaminated areas

Vice President, Construction Management, The Smoot Corporation, Columbus, OH, 1987-1990

- Managed and coordinated all construction and personnel activities for this minority-owned construction company.

General Superintendent of Construction, Uranium Enrichment Project, Stone & Webster, Piketon, OH, 1980-1986

- Responsible for all construction management activities, managing a large, creative fixed-price subcontracting program, and successfully implementing an innovative small and small disadvantaged business contracting program.

Chief Construction Supervisor, Shoreham Nuclear Power Station, Stone & Webster, Long Island, NY, 1975-1980

- Responsible for reactor vessel, fuel pool and safety related equipment installation.

Rémi Béra

Duke Cogema Stone & Webster (DCS)
Integration Design Group Manager

M.S., Mechanical Engineering
M.S., Nuclear Engineering

16 Years Experience

DUKE COGEMA
STONE & WEBSTER

Summary

Prior to joining DCS on the MOX Project, Mr. Béra has 16 years of experience in the nuclear industry in France and worldwide, including four years of experience in the U.S. as a Design Engineer with Babcock & Wilcox Fuel Company under a U.S. Department of Energy contract for which he managed the transfer of French spent nuclear fuel transportation cask technology to the U.S. He has eight years of experience with COGEMA, including seven years with a major COGEMA supplier (five as Nuclear Business Manager), during which he managed the design, fabrication, installation, and start-up support for integrated functional equipment for the MELOX MOX fuel fabrication plant, the MELOX extension project, the Cadarache revamping project and the La Hague facility. Mr. Béra has managed a number of nuclear equipment R&D, design, fabrication, and testing projects during his career, including the conceptual design of an interim spent fuel storage facility.

Integration Design Group Manager, Duke Cogema Stone & Webster
Primary Responsibilities

- Direct the manufacturing engineering for the MFFF process units, including manufacturing design, procurement engineering and integrated testing.
- Deliver to the site process units that meet MFFF requirements.
- Direct transfer of the process units to Construction and Operations.
- Ensure that technical, cost and schedule baselines are controlled for the IDG scope.
- Implement engineering and quality assurance procedures for the execution of the IDG scope.

Professional Development

M.S. Nuclear Engineering, 1983; Institut des Sciences et Techniques Nucleaires, France
M.S. Engineering, 1985; Ecole Centrale de Lyon, France

Clearances

None

Previous Work Experience

Manager, International Projects, COGEMA, France, 1999-Present

- Manages home office technical support provided to COGEMA, Inc., a U.S. subsidiary, for all U.S. proposal development and project management activities.

Manager, Nuclear Department, Robatel, France, 1993-1999

- Management responsibility and authority for nuclear department of a French corporation with annual sales of $30 million and a workforce of 200.

Project Manager, COGEMA Projects, Robatel, France, 1992-1993

- Technical and management responsibility for several nuclear design and fabrication projects for COGEMA valued at $7 million, including three vitrified high-level waste canister transfer casks and integrated glove boxes for plutonium calcine handling (R2 facility) for the La Hague spent fuel reprocessing plants.

Design Engineer, Babcock & Wilcox Fuel Company, 1988-1992

- Responsible for mechanical design and design configuration for the BR-100, a 100-ton spent nuclear fuel transportation cask, under a contract with the U.S. Department of Energy (DE-AC07-88ID12701).

Gary Bell

Duke Cogema Stone & Webster (DCS)
Software Design Group Manager and Lead Electrical Engineer
B.S., Electrical Engineering

24 Years Experience

DUKE COGEMA
STONE & WEBSTER

Summary

24 years of technical experience in electrical engineering and engineering management in both commercial nuclear power and U.S. Department of Energy projects. Most recent experience as the lead electrical engineer for the DOE Actinide Processing and Storage Facility which was designed stabilize, repackage and store special nuclear materials, primarily plutonium metal and oxide powder.

Software Design Group Manager and Lead Electrical Engineer, Duke Cogema Stone & Webster
03/99 to Present

- Direct MFFF software development activities
- Implement engineering and quality assurance procedures for the execution of Software Design Group scope.
- Design of the electrical distribution system for the MOX Fuel Fabrication Facility.
- Design of the Instrument and Controls for the mechanical and electrical utilities.
- Development of the construction specifications for the MOX process and the mechanical and electrical facilities.
- Design of the physical security systems for the MOX Fuel Fabrication Facility and the preparation of the security program documents
- Development of the Measurement Control and Accounting program for the MOX Fuel Fabrication Facility and design of the required instrument systems.
- Incorporation of the surveillance equipment of the IAEA into the design of the MOX Fuel Fabrication Facility process.
- Preparation of the licensing documents of Electrical systems, I&C systems, physical security systems, MC&A program, human factors design, and protection of classified material.

Professional Development

- Professional Engineer, Colorado
- Professional Engineer, South Carolina

Clearances

Active DOE L clearance

Previous Work Experience

Lead Electrical Engineer, Actinide Packaging and Storage Facility, 2 years

- Responsible for the electrical system design.
- Responsible for the security system design.

Project Manager, Electrical Engineering Support Contract, Hanford Site (Westinghouse Hanford Company), 3 years

- Manager of a number of electrical engineering and software tasks for the Hanford tank farms and Hanford utility systems

Project Manager, Continuing Engineering Services to Omaha Public Power District Fort Calhoun Station, 3 years

- Manager of a various upgrade tasks, electrical, mechanical, and structural.

Lead Electrical Engineer and Project Engineer, Continuing Engineering Services to Omaha Public Power District Fort Calhoun Station, 5 years

- Design Radiation Monitor Upgrade
- Electrical and controls design for post accident sampling system.
- Electrical and controls design for control room habitability upgrades.

Lead Electrical and Controls Engineer, N-Reactor Control Room Habitability Upgrades, 2 years

- Electrical design of new standby power to new control room ventilation system.
- I&C design for control room habitability upgrade

CHAPTER 5, INTEGRATED SAFETY ANALYSIS

38. Section 5.2, p. 5.2-1

Clarify the makeup and functions of the "Safety Assessment Team."

Section 4.4.3.A.iii.a of the SRP states, for key management positions, "The personnel to design and construct the facility have the appropriate breadth and level of experience for their respective authorities and responsibilities, as indicated in the organizational structure..."

[A similar requirement exists for operations. Section 4.4.3.B.iii.a of the SRP states, for key management positions, "The personnel to operate the facility have the appropriate breadth and level of experience for their respective authorities and responsibilities, as indicated in the organizational structure..."]

Section 5.2 of the application is entitled, "Safety Assessment Team Description." However, no specific individuals and their qualifications are identified. In addition, fuel fabrication and/or MOX experience is not mentioned. The NRC would anticipate that personnel of appropriate experience and qualifications would be identified to ensure that safety and safety-related issues are adequately assessed.

Response:

The makeup of the SA team changes over time and with the processes and hazards evaluated. Therefore, no specific individuals and their qualifications are identified in the CAR. However, in accordance with the guidance in the SRP, key roles were identified for the SA Manager and the SA Team Leaders. In addition, DCS committed to the use of subject matter experts relative to the hazard being evaluated. Guidance provided in NUREG-1718 Section 5.4.3.2(B)(iv) requires that the ISA team leader is trained and knowledgeable in the safety assessment process chosen and that at least one member of the team has knowledge and familiarity in the process under evaluation. CAR Section 5.2 does note that, "The SA Team Leader(s) is knowledgeable in the specific SA methodologies chosen for the hazard and accident analyses and has an understanding of process operations and the hazards under evaluation." Instead of naming specific individuals, Section 5.2 of the CAR specifies the disciplines and experience base of the team.

More information on the key roles in the ISA team is provided below.

The current SA Manager has extensive experience in management related to safety analysis at commercial nuclear and DOE facilities. The SA manager's commercial nuclear experience includes management, safety analysis, and licensing of BWRs, PWRs, and spent fuel storage projects. Past DOE experience includes the management of Savannah River Reactor restart efforts.

The current SA Team leader also has extensive experience in Safety Analysis at commercial nuclear and DOE facilities. Directly reporting to the SA Team leader are a Criticality Lead, a Safety Analysis Lead, and a Radiation Safety Lead.

The Criticality Lead is responsible for performing all criticality analyses, as described in Chapter 6 of the CAR

The Safety Analysis Lead is responsible for MFFF safety analysis including the hazard analysis, accident analysis, and the safety assessment of the design basis. This individual and the current team that reports to him have extensive experience in performing all forms of safety analysis including hazards assessments, HAZOP, What If, Fault Tree/ PRA, and other forms of accident analysis. This experience consists of work performed on DOE and NRC licensed facilities, as well as direct safety analysis experience on the La Hague and MELOX facilities.

The Radiation Safety Lead is responsible for the implementation of radiation protection as described in Chapter 9 of the CAR.

Although "fuel fabrication" experience is not specifically mentioned, Section 5.2 does specify that team members "possess operational experience at similar facilities" (the "similar facilities" means, for example, MELOX, a MOX fuel fabrication facility). It also specifies that the team has "MOX-specific safety analysis experience." The DCS SA team employs the operational experience of persons from COGEMA's La Hague and MELOX facilities. The bulleted list in the second paragraph of Section 5.2 provides examples of the expertise used.

Action:

None

39. <u>Section 5.4.3, pp. 5.4-8 and 5.4-9; Section 5.5.2.4.6, pp. 5.5-27 thru 5.5-31</u>

Provide more quantification or other specificity for likelihood definitions and reliability requirements of the process safety I&C system.

Section 8.3 of the SRP states, "Information contained in the application should be of sufficient quality and detail to allow for an independent review, assessment, and verification by the reviewers. Some information may be referenced to other sections of the application, or incorporated by reference, provided that these references are clear, specific, and essentially complete." SRP Section 8.4.3.1 states that an application would be acceptable if it addresses the baseline design criteria for chemical safety and includes information on the chemicals, process, equipment, inventories, ranges, and limits. At the construction permit stage, this would be expected to include design bases and values for these items, with sufficient system description to allow verification of the design bases and values. Sections 8.4.3.5 B, C, D, and F recommend that design bases, process safety features, and IROFS be included in the application.

Section 5.4.3 mentions likelihood definitions and states, "In general, qualitative methods will be used ..." Section 5.5.2.4.6, "Safety Evaluation," relies upon the process safety I&C system as the principal SSC/IROFS for many events. However, it is not clear how the likelihood/reliability of the process safety I&C system - and other principal SSCs/IROFSs for that matter - meets the likelihood requirements of Part 70. This is exacerbated by the lack of quantification or numerical values and ranges for the reliability of the process safety I&C system. An explanation of the layers of protection and their reliability/likelihoods is not included. For example, commitment to a Class 1 safety instrumentation system might have a Probability of Failure on Demand (PFOD) of 0.1, a class 2 system a PFOD of 1E-02, and a Class 3 system a PFOD of 1E-03. In the absence of more quantitative information, it is not possible to make a safety determination.

Response:

The SRP (NUREG-1718) discusses the regulatory acceptance criteria for likelihood under 5.4.3.2.B.iiv:

> The regulations of 10 CFR 70.65 require the applicant's ISA Summary to provide definitions of the terms unlikely, highly unlikely, and credible. The applicant's definitions of these terms is acceptable if, when used with the applicant's method of assessing likelihoods, they provide reasonable assurance that the performance requirements of 10 CFR 70.61 can be met. The applicant's <u>method</u> of likelihood evaluation and the <u>definitions</u> of the likelihood terms are closely related. Qualitative methods require qualitative definitions. Such a qualitative definition would identify the qualities of IROFS controlling an accident sequence that would qualify that sequence as "unlikely" or "highly unlikely."

In the CAR, DCS has committed to ensuring that all event sequences are either of low consequence or are made highly unlikely, through the application of sufficient controls, to meet the performance requirements of 10 CFR 70.61. The following definition of highly unlikely is provided in the CAR:

> **Highly Unlikely** – Events originally classified as Not Unlikely or Unlikely to which sufficient principal SSCs are applied to further reduce their likelihood to an acceptable level (see discussion below).

To clarify the level of risk reduction provided, DCS has developed an approach for implementing the proposed definition of highly unlikely that includes the following elements.

(1) For the facility worker, DCS commits to utilizing the following deterministic design criteria to establish the safety/licensing basis (i.e., to meet the performance requirements of 10 CFR 70.61), consistent with the historical licensing basis for nuclear facilities:

- Application of the single failure criteria (or double contingency for nuclear criticality);
- Application of 10 CFR 50 Appendix B, NQA-1;
- Application of Industry Codes and Standards; and
- Management Measures, including surveillance of IROFS (i.e., failure detection and repair or process shutdown capability).

(2) For the site worker and public, DCS commits to the aforementioned deterministic design criteria to establish the safety/licensing basis.

In conjunction with (but separate from) the safety/licensing basis, and to provide additional confidence in the demonstration of the adequacy of these deterministic design criteria, DCS also commits to a supplemental likelihood assessment for events, excluding NPH, that could result in consequences that exceed the threshold criteria for the site worker or the public. This supplemental likelihood assessment will be based on the guidance provided in the MFFF SRP (NUREG-1718) and will demonstrate a target likelihood index comparable to a "score" of -5 as defined in Appendix A of the SRP.

The demonstration of compliance will be part of the ISA and reflected in the ISA Summary accompanying the license application.

Action:

Revise the CAR to include the above commitments.

40. Section 5.4.3, pp. 5.4-8 and 5.4-9

State whether "unlikely" as defined in Section 5.4.3 of the application for intermediate consequence events is considered equivalent to "unlikely" as defined in the double contingency principle.

"Unlikely" is defined in this section for the purposes of performing the ISA, but no definition of the term in the double contingency principle is provided. This term is part of the definition of double contingency, and the staff must understand what the applicant means by this to determine whether the requirements of 10 CFR 70.64(a)(9) have been met.

Response:

The 10 CFR 70.61(d) requirement to limit the risk of criticality by ensuring subcritical conditions under all normal and credible abnormal conditions, and the 10 CFR 70.64(a)(9) requirement to adhere to the double contingency principle, combine to effectively preclude criticality. DCS had anticipated that meeting 10 CFR 70.61(d) and 10 CFR 70.64(a)(9) obviated the need to demonstrate separately compliance with 10 CFR 70.61(b) and (c) (i.e., neither intermediate nor high consequences will occur from an event that is prevented). Therefore, the CAR was not intended to equate "double contingency" with "highly unlikely." Rather, the discussion in Section 5.4.3 was intended to convey the satisfaction of two *separate* criteria: that application of the double contingency principle would provide for compliance with the performance requirement of 10 CFR 70.61(d), and that the application of deterministic/single-failure criteria (in accordance with traditional engineering methods) would satisfy the performance requirements of 10 CFR 70.61(b) and (c).

For preventing or mitigating postulated events other than criticality, the response to Question 39 provides the rationale and clarifies the commitments regarding making the consequences of such events highly unlikely.

For criticality events, DCS has committed in Section 6.3 of the CAR to demonstrate application of the double contingency principle in order to comply with the requirements of 10 CFR 70.64(a)(9) and 70.61(d), using established and accepted methods and definitions found in the ANS/ANSI 8-series standards. As indicated in NUREG-1718 (Section 5.4.3.2.B.vii.b(4)), "one acceptable definition of *highly unlikely* is a system of IROFS that possesses double-contingency protection, where each of the applicable qualities is present to an appropriate degree." DCS believes that the application of these standards along with the provisions below result in the "appropriate degree" necessary for demonstration of double contingency to equate to *highly unlikely* in the event separate compliance with 10 CFR 70.61(b) and (c) is required:

- Commitment to the use of engineered features over administrative controls wherever practical
- Conservative assumptions and use of design margin

- Demonstration of independence in unlikely process changes that would have to occur for criticality to be possible

- Application of management measures such as IROFS surveillance under a 10 CFR 50 Appendix B QA program.

Action:

DCS requests clarification from the NRC of the separate applicability of 10 CFR 70.61(b) and (c) to postulated criticality events that are prevented in accordance with 10 CFR 70.61(d) and 64(a)(9). Upon receipt of additional guidance from the NRC, the CAR will be revised to reflect this information appropriately.

41. Section 5.4.3, pp. 5.4-8 and 5.4-9

Explain the statement in Application Section 5.4.3 that "application of the double contingency principle and/or single-failure criteria (in accordance with traditional engineering methods) is sufficient to satisfy the performance requirements of 10 CFR §70.61." Demonstrate how application of the double contingency principle ensures that criticality is "highly unlikely".

10 CFR Part 70 contains two different requirements (among others) that must be met for criticality safety: the requirement in §70.61(b) that high-consequence events must be "highly unlikely", and the requirement in §70.64(a)(9) for criticality control including adherence to the double contingency principle. These are not necessarily equivalent.

Response:

Refer to the response to Question 40.

Action:

None

42. Section 5.4.4, pp. 5.4-9 thru 5.4-14

Revise the description in Section 5.4.4 that members of the public and the environment are considered to be outside the controlled area boundary approximately 5 miles (8 km) from the MFFF building stack.

NRC considers Savannah River Site workers who are outside the MFFF restricted area but within the controlled area boundary and who are not closely and frequently connected to the licensed activity as members of the public. Further, the definition of a member of the public does not preclude these individuals from being present in the controlled area.

Section 10.4.3.C.ii.c of the SRP recommends that the applicant use acceptable methods for estimating consequences from accident sequences that result in radiological releases to the environment. With regard to considering the affected environment to be outside a 5 mile radius from the MFFF building stack, NRC regulation 10 CFR 70.61(c) describes several intermediate consequence events, including a "24-hour average release of radioactive material outside the restricted area in concentrations exceeding 5000 times the values in Table 2 of Appendix B to Part 20." The calculation of this 24-hour average release concentration is described in Equation 5.4-3 on page 5.4-11 of the application. A term in this equation is the atmospheric dispersion factor for a member of the public. The derivation of the value for this factor is described in section 5.4.4.1.3.1, which states that the factor was derived using a distance of 5 miles (8 km) from the point of release. However, 10 CFR 70.61(c)(3) describes the intermediate consequence event as a release outside the restricted area. In section 1.1.2.1, DCS describes the restricted area boundary as "coincident with the protected area, an area encompassed by physical barriers and to which access is controlled, as shown in Figure 1.1-2." From Figure 1.1-2, it appears that the protected area boundary is approximately 650 feet from the plant stack, not 5 miles (8 km).

Response:

This response is provided in two parts. The first part addresses the controlled area boundary questions; the second part addresses accidental releases to the environment.

CAR Section 5.4.4 will be revised to delete the reference to the "environment" concerning consideration of radiological consequences outside the controlled area boundary. Radiological consequences to the environment will be assessed outside the MFFF restricted area in accordance with the regulation.

Controlled Area Boundary

As discussed in the response to Question 2, the controlled area boundary has not been changed; it is described in Section 1.1.2.1 of the CAR. The boundary is approximately 5 miles from the MFFF site. The response to Question 1 provides the basis for DCS' position that SRS workers are not "members of the public."

Accidental Releases to the Environment

The calculation of the 24-hour average release concentration (environmental concentration calculation) will be revised in accordance with 10 CFR 70.61(c)(3) for the intermediate consequence event as a release outside the restricted area.

Action:

CAR Section 5.4.4 will be revised to delete the reference to the "environment" concerning consideration of radiological consequences outside the controlled area boundary. Radiological consequences to the environment will be assessed outside the MFFF restricted area in accordance with the regulation.

43. Section 5.4.4.1.1, pp. 5.4-9 thru 5.4-10

Explain how the duration of entrainment events is limited in deriving the source term for entrainment events.

Section 10.4.3.C.ii.c of the SRP recommends that the applicant use acceptable methods for estimating consequences from accident sequences that result in radiological releases to the environment. In Section 5.4.4.1.1, DCS states that "for entrainment events, the airborne release fraction is replaced with the airborne release rate (ARR) multiplied by the entrainment duration (i.e., ARF = ARR x duration)." However, in Section 5.4.4, DCS states that "no evacuation is credited for the assessment of the unmitigated radiological consequences." Since exposure to material at risk is not limited by evacuation for entrainment events, DCS should explain how the exposures are otherwise limited. For example, does the duration of the event coincide with the complete expenditure of the material at risk?

Response:

A few events identified in the PHA include entrainment as part of the consequence analysis. These events involve backflow through a system connected to a glovebox and involve small amounts of material. The unmitigated analysis assumes the entrainment portion of the release occurs for one-hour assuming no evacuation. Although a one-hour time period has been assumed in the calculation of unmitigated consequences for entrainment events, assuming a longer duration would not change the safety strategy applied to this event. This occurs because a principal SSC is present to mitigate any potential release. This principal SSC, the C4 confinement system, is credited in the CAR for meeting the performance requirements of 10 CFR 70.61. It prevents the entrainment event from occurring. Consequently, there are no entrainment events that are screened on the basis of low unmitigated consequences (i.e., they are not listed in Table 5.5-25). Thus, even if the entrainment duration is assumed to be much longer than one hour and results in the entire MAR being involved in the event, the specified principal SSCs would not change, and the mitigated consequences would be acceptable. Note that the unmitigated consequences associated with backflow events are orders of magnitude below the unmitigated consequences associated with the bounding events.

Action:

The CAR will be revised to reflect this information.

44. <u>Section 5.4.4.1.2, pp. 5.4-10 thru 5.4-12</u>

Calculate the effluent concentration ratio without taking credit for the respirable fraction.

SRP Section 9.1.4.6.3.A recommends that the applicant use appropriate and verified assessment methods, computer codes, and literature values. Equation 5.4-3, the equation for 24-hour average effluent concentration ratio, contains a term for source term (ST) which is the same term as that used in Equation 5.4-2 for total effective dose equivalent to human receptors. The definition of source term, which is provided in Equation 5.4-1, has a term for the respirable fraction (RF). However, the inclusion of RF in source term derivations for demonstrating compliance with 10 CFR 70.61(c)(3) ~~is not appropriate. This~~ performance requirement relates to protection of the environment, not to protection of human health. Therefore, the applicant should demonstrate that the performance requirement is met for the entire range of particle sizes released to the environment, not just the respirable particle sizes.

Response:

Equation 5.4-3 will be revised to address the full range of particle sizes associated with the event and effluent concentration calculations will be revised accordingly. See the response to Question 42 for additional information related to the effluent concentration calculation.

Action:

The CAR will be revised to reflect this information.

45. Section 5.4.4.1.2, pp. 5.4-10 thru 5.4-12

Clarify how dose conversion factors from Federal Guidance Report No. 11 were chosen with due consideration for the chemical forms of radionuclides involved in accident scenarios.

Section 9.1.4.6.3.A of the SRP recommends that the applicant use appropriate and verified assessment methods, computer codes, and literature values. Section 5.4.4.1.2, "Dose Evaluation," describes the assumptions for calculating bounding total effective dose equivalent to individuals exposed during accidents, including the use of Federal Guidance Report No. 11 as the source of dose conversion factors used in the analysis. In many cases, Federal Guidance Report No. 11 provides dose conversion factors for more than one chemical form (or solubility) of the radionuclides listed. These multiple forms are represented by the transportability classes D, W and Y, where, for plutonium, the more limiting dose conversion factors are generally associated with class W compounds (such as plutonium nitrate). The application does not contain a description in Section 5.4.4.1.2 of how the solubility of various chemical forms of plutonium and americium were considered in performing the dose assessments.

Response:

Selection of the dose conversion factors (DCFs) from Federal Guidance Report No. 11 is based on the form of the potential releases from the MFFF when received by the dose receptor. For the MFFF, dose receptors are conservatively assumed exposed to oxides of unpolished plutonium, polished plutonium, and/or uranium. Thus, Y clearance class DCFs have been used for all nuclides except americium, which only has a W clearance class DCF.

This is a conservative assumption based on the following:

The assumed release forms (i.e., oxides) have specific activities (molecular) greater by approximately a factor of 2 than those of other potential release forms (e.g., plutonium oxalates and nitrates), including those that are considered more soluble.

The difference between the DCFs of the various clearance classes (< 2 for effective dose conversion factors) is not as significant as the difference in the specific activities (molecular).

Thus, the following relation exists:

$$[(specific\ activity)\times(DCF)]_{oxide} \geq [(specific\ activity)\times(DCF)]_{nitrate,oxalate,et\,al.}$$

Releases of soluble materials are further bounded by those of the insoluble form because the amount of material at risk in the bounding events for soluble releases is smaller than the amount of material at risk for the insoluble releases (by approximately a factor of 6).

Action:

The CAR will be revised to reflect this information.

46. Section 5.4.4.1.3. pp. 5.4-12 thru 5.4-13

Provide the hourly meteorological data for the period from January 1, 1987 through December 31, 1996 that was collected from the H-area meteorological tower. Include the standard deviation of the horizontal wind direction fluctuations (sigma-theta), derived stability class, wind direction, wind speed and accumulated precipitation for each hour. Include a description of how stability classes are derived using sigma-a and sigma-theta.

Section 5.4.3.2.B.v of the SRP recommends that the applicant provide a scientifically correct and reasonable estimate of the consequences from analyzed accidents. Several radiological accident consequence models that NRC may use to verify the applicant's dose calculations require hourly measurements of meteorological data. Therefore, the NRC staff must have the actual hourly data, rather than statistical summaries, to verify the correctness and reasonableness of the applicant's estimates.

Response:

All downwind transport wind direction, wind speed, stability class, and sigma-theta data were extracted from two separate five-year data bases (1987 to 1991 and 1992 to 1996), which contain a complete sequential record of quality-assured SRS hourly meteorological data monitored on the H-Area meteorological tower.

All values of an empirical atmospheric turbulence intensity parameter that can be related to a Pasquill stability category are automatically derived by a microprocessor on the meteorological tower. Stability classes were based on the measured values of the standard deviation of fluctuation about the mean horizontal wind direction.

Stability categories A through G were assigned according to the range of magnitudes of sigma-theta as summarized in Table 1. Hourly precipitation data were obtained from records collected by the National Weather Service Office in Augusta, Georgia (Bush Field) and published by the National Climate Data Center (NCDC). All values of mixing height were calculated from data sets of twice daily mixing height supplied by NCDC. The daily values were determined by NCDC from a standard algorithm that used radiosonde ascents for Athens, Georgia (January 1987 through August 1994) and Atlanta, Georgia (September 1994 through December 1996), and concurrent surface data from Bush Field as input data.

The attached Meteorological Data File (on CD) contains ten files that contain hourly meteorological data used as input to the MELCOR Accident Consequence Code System (i.e., MACCS2). These files were extracted from the SRS database and contain a one-year data set for each of the years 1987 through 1996. The MACCS2 data sets consist of hourly-averaged values of plume transport sector (22.5-degree sector toward which the wind blows), wind speed (tenths of meters per second), Pasquill atmospheric stability category (1-7), and precipitation (hundredths of inches). In addition, the last line of each file gives seasonally-averaged values of morning and afternoon mixing height. Wind transport sectors in the MACCS2 files were assigned according to the transport direction ranges given in Table 2. The MACCS2 file format is described in Table 3.

The attached Meteorological Data File (on CD) contains ten files used as input into the ARCON96 computer code, one for each of the calendar years 1987 through 1996. Each file consists of a five-character station identifier (SRSOH), Julian day, hour (local time), wind direction (i.e., the direction the wind blows from) in degrees azimuth, wind speed in meters per second, and Pasquill stability classes (1-7 or A-G). All data were extracted from two separate five-year databases (1987 to 1991 and 1992 to 1996), which contain a complete sequential record of quality-assured hourly data. ARCON96 file format is described in Table 4.

The attached Meteorological Data File (on CD) contains eight files of hourly meteorological data used as input to the Environmental Protection Agency's Industrial Source Complex (ISC) atmospheric dispersion model. These files contain a one-year data set for each of the years 1989 through 1996. The ISC data sets consist of hourly values of transport direction (i.e., degrees toward which the wind blows), wind speed (meters per second), mixing height (meters), ambient temperature (degrees Kelvin), and Pasquill atmospheric stability category. The ISC file format is described in Table 5.

Action:

None

Table 1. Determination of Stability Class from Sigma-Theta Measurements

Sigma-theta* Range (degrees)	Pasquill Stability Category	Stability Identifier
Greater than 22.4	A	1
17.5-22.4	B	2
12.5-17.4	C	3
7.5-12.4	D	4
3.8-7.4	E	5
2.1-3.7	F	6
Less than 2.1	G	7

* Sigma-theta range assignments are based on criteria contained in ANSI/ANS-3.11 (2000).

Table 2. Determination of Downwind Transport Direction Sector*

Downwind Transport Direction Range (degrees)	Downwind Transport Sector	Sector Identifier
348.75-11.25	North	1
11.25-33.75	North-northeast	2
33.75-56.25	Northeast	3
56.25-78.75	East-northeast	4
78.75-101.25	East	5
101.25-123.75	East-southeast	6
123.75-146.25	Southeast	7
146.25-168.75	East-southeast	8
168.75-191.25	South	9
191.25-213.75	South-southwest	10
213.75-236.25	Southwest	11
236.25-258.75	West-southwest	12
258.75-281.25	West	13
281.25-303.75	West-northwest	14
303.75-326.25	Northwest	15
326.25-348.75	North-northwest	16

* downwind transport direction is the direction the wind is blowing toward.

Table 3. Format of the MACCS Meteorological Data Files

Column	Format	Description
2-4	I3	Julian day of the year
6-7	I2	Hour of the Day (GMT)
9-10	I2	Transport direction sector (direction wind blows toward)
11-13	I3	Wind speed (10ths of meters/sec)
14	I3	Stability Class (coded 1-7)
15-17	I3	Total precipitation (100ths of inches)

Table 4. Format of the ARCON96 Meteorological Data Files

Column	Format	Description
3-7	A5	Location identifier (SRSOH)
11-13	I3	Julian day of the year
14-15	I2	Hour of the Day (local time)
18-20	I3	Wind direction (degrees, direction from which the wind blows)
21-24	I4	Wind speed (nearest tenth of a reporting unit without the decimal, i.e., a wind speed of 5.3 m/s would be entered as 53)
26	I2	Stability class (coded 1 - 7)

Table 5. Format of the ISC Meteorological Data Files

Column	Format	Description
7-9	I3	Julian day of the year
11-12	I2	Hour of the Day (Greenwich Mean Time)
18-25	F8.0	Transport direction (degrees azimuth)
26-33	F8.2	Wind speed (meters/sec)
34-41	F8.0	Mixing height (meters)
42-49	F8.0	Temperature (Degrees Kelvin)
50-57	I8	Stability Class (coded 1-7)

Attachment 46
Additional Meteorological Data

Attached Meteorological Data File (on CD)

47. <u>Section 5.5.1.1.1, p. 5.5-1; Table 5.5-2, pp. 5.5-60 thru 5.5-67</u>

Explain why high enriched uranium (HEU) is not included in Table 5.5-2 of the application as a hazardous material ("Haz Mat").

Section 5.4.3.1.D of the SRP recommends that the applicant show that the hazard identification method provide a list of material or conditions that could result in hazardous situations. The HEU waste stream is described in several sections throughout the application. For example, Section 8.1.1.2.3 briefly describes the dilution of the HEU and subsequent storage in a vessel, but neither the HEU nor its storage vessel is explicitly described in Table 5.5-2 as a hazard.

Response:

Table 5.5-2 will be revised to incorporate quantities of HEU (separated from the plutonium) and depleted uranium contained in the AP process. These values are less than 3 kg and less than 750 kg, respectively. The AP process will process approximately 20 kg of HEU per year. Systems handling HEU are described in response to Question 135.

Action:

The CAR will be revised to reflect this information.

48. <u>Section 5.5.1.1.1, p. 5.5-1; Table 5.5-2, pp. 5.5-60 thru 5.5-67; and Table 1.2-1, p. 1.2-7</u>

Resolve the discrepancy between the possession limit Table 1.2-1 and that which would be expected to be in the facility based on the "Non-Polished Plutonium Sources" fraction of 70 year Final Isotopic Composition found in Table 9-3, as > 95 percent plutonium-239. ***Text removed under 10 CFR 2.390.**

10 CFR 70.22(a)(4) requires that the applicant provide the name, amount and specification of the special ~~nuclear material the applicant~~ proposes to use. From Table 9-3, the 70 year Final Isotopic Composition in non-polished plutonium sources would be expected to include uranium-235 per gram of plutonium plus americium. Given that the possession limit for plutonium this would amount to a potential maximum uranium-235 in the form of >90% enriched uranium-235,

***Text removed under 10 CFR 2.390.**

Response:

Table 1.2-1 will be revised to address these concerns.

The isotopic quantities ~~in Chapter 9~~ (specifically Table 9-2) are conservatively established for shielding calculations and ~~should not be~~ used to establish possession limits. The value used to establish the actual U-235 content in unpolished PuO_2 is 0.5 weight percent.

The possession limit for uranium, any enrichment, will be revised to account for that in the unpolished plutonium. Also, the possession limits for MOX will be revised to include the uranium content and the possession limit for source material will be revised to include the maximum amount of material that may be stored in the secured warehouse. Lastly, the americium possession limit will be revised to include all plutonium decay products, except uranium.

Action:

Table 1.2-1 is to be revised as indicated below:

Table 1.2-1. Byproduct Material, Source Material, and Special Nuclear Material

Type of Material	Form of Material	Possession Limit
Source Material (Natural and/or Depleted Uranium)	Any chemical or physical form	
Plutonium, with ≤ 95 wt% ^{239}Pu	Any chemical or physical form	
MOX (mixture of UO_2 and PuO_2), with ≤ 22 wt% PuO_2	Any chemical or physical form	
MOX, with ≤6.3 wt% PuO_2	Any chemical or physical form	
Enriched Uranium, any enrichment	Any chemical or physical form in unpolished plutonium and waste	
Plutonium Decay Products, except Uranium	Any chemical or physical form in unpolished plutonium and waste	

Possession limits for other byproduct material, such as ^{252}Cf, ^{137}Cs, ^{60}Co, etc. will be submitted with the license application for possession and use of special nuclear material.

Text removed under 10 CFR 2.390.

49. Section 5.5.1.2, p. 5.5-3

Provide calculations and design bases to demonstrate that passive heat removal is adequate.

Section 8.3 of the SRP states, "Information contained in the application should be of sufficient quality and detail to allow for an independent review, assessment, and verification by the reviewers. Some information may be referenced to other sections of the application, or incorporated by reference, provided that these references are clear, specific, and essentially complete." SRP Section 8.4.3.1 states that an application would be acceptable if it addresses the baseline design criteria for chemical safety and includes information on the chemicals, process, equipment, inventories, ranges, and limits. At the construction permit stage, this would be expected to include design bases and values for these items, with sufficient system description to allow verification of the design bases and values. Sections 8.4.3.5 B, C, D, and F recommend that design bases, process safety features, and IROFS be included in the application.

On page 5.5-3, the application indicates that passive removal of decay heat is adequate for all areas except for the 3013 canister storage area. Calculations to justify that passive heat removal is adequate need to be provided before a safety determination can be made.

Response:

This response provides a summary of the design bases for decay heat and provides a summary of the thermal analyses performed to support preliminary design. Calculations to support final design will be summarized in the ISA.

The thermal power generated by the decay of nuclear material was calculated as follows:

- Unpolished Pu: 2.899 W/kg of unpolished PuO_2 powder
- Polished Pu: 2.181 W/kg of polished PuO_2 powder.

These values or higher (more conservative) values have been used in the thermal analyses of the different areas of the MFFF.

Although the gloveboxes are seismically qualified, air is conservatively assumed to enter the glovebox and cause oxidation of the UO_2 powder. The specific power of oxidation released by this exothermic reaction is taken into account using the following values:

- If $T < 74°C$ (165.2°F) then $P_{ox} = 0$ W/kg (0 W/lb) of UO_2,
- If 74°C (165.2°F) $< T <$ 340°C (644°F) then $P_{ox} = 1.1$ W/kg (0.499 W/lb) of UO_2,
- If $T > 340°C$ (644°F) then $P_{ox} = 4.63$ W/kg (2.1 W/lb) of UO_2

Where T is the powder temperature.

Two conditions were evaluated:

- Normal operating conditions in which the ventilation and cooling systems of equipment are in operation. During normal conditions, operation of the ventilation systems to remove thermal energy was considered in calculating the maximum temperatures of the structural confinement components.

- Hypothetical accident conditions in which the ventilation and cooling systems of equipment are assumed to be shutdown. In this conservative case, the maximum temperatures calculated are compared to the thermal design criteria of the hypothetical situation.

The design basis temperature criteria have been defined for relevant materials in normal and hypothetical accident conditions. The primary temperature criteria for the materials in the MFFF are summarized in Table 1. Although not required to meet the performance criteria of 10 CFR 70.61, additional temperature criteria have been defined in order to support worker radiation protection (ALARA) and to provide personnel protection from high temperatures during normal operations. These values are provided in Table 2.

Units with potentially large heat loads include the following:

- Storage rooms (rod storage, assembly storage, PuO_2 storage)

- Storage gloveboxes (PuO_2 buffer storage, handling and storage tunnel, pellet storage)

- High-capacity production units (homogenization units, and hopper and mixer in the primary dosing glovebox and the hoppers in the final dosing glovebox).

The results of the analysis indicate the following:

- During normal operation, the thermal design criteria are met for all units

- For the hypothetical condition, the analysis shows that after several days, the thermal design criteria could be exceeded in the PuO_2 storage area. Thus, cooling is required to be established within several days to ensure the thermal criteria are met.

- For the hypothetical condition, the temperature of the upper lateral polycarbonate panel in the handling and storage tunnel exceeds the thermal design criteria by approximately 10°F. However, more detailed calculations are expected to show that material limits are not exceeded. If necessary, cooling will be provided. This information will be provided in the ISA.

As stated in the CAR, the high depressurization exhaust (HDE) system is specified as the principal SSC necessary to establish cooling for the 3013 canister storage area during post-earthquake conditions. Preliminary sizing calculations for the HDE system show that the system has the capability to perform this function. Final calculations that demonstrate the HDE can perform this function, and to determine the specific time requirements, will be described in the ISA summary.

Action:

The CAR will be revised to reflect this information.

Table 1. Design Basis Temperature Criteria

Material	Situation	
	Normal	**Hypothetical**
Ordinary concrete	60°C	100°C
Stainless Steel	425°C	425°C
PPB #9 @	80°C	100°C
PPB #10 @	100°C	100°C
NS41 Silicone Elastomer	180°C	180°C
Polycarbonate (Lexan)	35°C * 50°C	70°C

* 35°C applies to gloveboxes that are subject to thermal cycling
@ Boronated Polyethylene Plaster (BPP)

Table 2. Additional Temperature Criteria for Personnel Protection

Material	Normal Operation
Borate (colemanite) concrete	80°C
Kyowaglas	80°C (storage) 35°C (operating)*
Fuel rods and pellets (Cladding)	60°C

* A higher temperature criterion is given for storage gloveboxes since personnel are not continuously located near them.

50. <u>Section 5.5.2.1.6.1, pp. 5.5-5 and 5.5-6</u>

Clarify the events and design bases for the electrolyzer. Section 8.3 of the SRP states, "Information contained in the application should be of sufficient quality and detail to allow for an independent review, assessment, and verification by the reviewers. Some information may be referenced to other sections of the application, or incorporated by reference, provided that these references are clear, specific, and essentially complete." SRP Section 8.4.3.1 states that an application would be acceptable if it addresses the baseline design criteria for chemical safety and includes information on the chemicals, process, equipment, inventories, ranges, and limits. At the construction permit stage, this would be expected to include design bases and values for these items, with sufficient system description to allow verification of the design bases and values. Sections 8.4.3.5 B, C, D, and F recommend that design bases, process safety features, and IROFS be included in the application.

Section 5.5.2.1.6.1 describes a fire event involving the electrolyzer. The over-temperature results in the boiling of the electrolyzer contents and breach of the glovebox. Ultimately, solution containing unpolished plutonium is released into the C3b area. The application indicates a prevention-type strategy is used to protect the worker, based upon the process safety I&C system. The safety function involves shutting down the electrolyzer system and allowing passive convection cooling down to ambient temperature (i.e., no safety cooling system). Protection of the site worker and the public involve the filters on the C3 confinement system. However, no design basis information is provided to support these scenarios and proposed control approach. In addition, it is likely that there will be significant electrical currents and/or other energy input during an off-normal condition of the electrolyzer that metallic components could be heated significantly above ambient temperature. This could result in additional scenarios, such as hydrogen generation and metal combustion, that might not be addressed by the proposed control schemes. A determination of adequate assurances of safety cannot be made without this additional description and design basis information.

Response:

The hazard events associated with the electrolyzer (dissolution unit) are described in Appendix A of Section 5.5 of the CAR. Principal SSCs have been identified for loss of confinement events, nuclear criticality events, and potential explosion events. Correspondingly, safety strategies and principal SSCs to address these may also be found in Section 5.5 of the CAR. These events result in the establishment of design basis to address the performance requirements of 10 CFR 70.61 for the electrolyzer.

The loss-of confinement hazard event involving the electrolyzer hazard described in detail in the CAR is an over-temperature event. Potential causes may include control system failure, electric isolation failure, or loss of cooling. For the unmitigated event, the over-temperature event is assumed to result in damage to the electrolyzer and the electrolyzer glovebox, and result in a release of radioactive material. The release is modeled as a boiling event with an ARF *RF of 0.002, involving the maximum amount of Pu in the electrolyzer, 14 kg of unpolished Pu. The unmitigated results of this event could produce consequences that exceed the low consequence criteria. Thus, the event must either be prevented or mitigated. The specific prevention/mitigation strategies for this event are described in CAR Section 5.5.2.1.6.1.

A HAZOP study for the electrolyzer will be performed to identify all potential accident sequences that may lead to an over-temperature event. This analysis is performed by a multi-disciplinary team of safety, operations, and process experts. The team will investigate all potential upset conditions as required by the formal HAZOP method that DCS is using. Thus, additional specific causes will be identified while performing HAZOP studies for the ISA. Prevention features will be identified as a result of these causes and labeled IROFS. DCS expects that the IROFS will be temperature controls and process shutdown controls. The temperature controls will detect high temperature, and the process shutdown controls will shut down the process prior to exceeding any design limits or chemical control limits. During the detailed design process, an appropriate design for these IROFS will be created. In the particular case of the electrolyzer, the IROFS temperature channels will be assigned to the emergency control system, and either channel of the emergency control system channels will be capable of shutting down the electrolyzer in the event of an over temperature condition.

Once the process is shutdown, natural convection cooling is sufficient to cool the system down. Thus, the proposed controls will terminate the event prior to any SSCs being damaged, prior to any radioactive material release, and prior to any additional scenarios such as metal combustion or excess hydrogen generation events.

Thus, the design bases for the electrolyzer associated with the over-temperature event is to detect high temperature and shutdown the electrolyzer and related processes prior to exceeding a specified temperature. This temperature will consider all material and chemical reactions. The limiting criterion for the electrolyzer is associated with controlling chemical reactions and is above 70°C, to preclude plutonium solution boiling. Final calculations and specific temperature setpoints will be performed during final design.

To preclude a nuclear criticality event, the electrolyzer is geometrically safe. NCSEs will be performed to identify credible event sequences that could compromise the geometry. Currently, it is known that corrosion in the electrolyzer is one such event sequence that could potentially challenge the safe geometry of the unit. Consequently, to minimize the potential for corrosion the electrolyzer is fabricated from titanium. In addition, the anode and cathode of the electrolyzer are isolated from ground. An isolation monitoring system comprised of a 10-ohm resistor between the electrolyzer (anode section) and earth, a 1000-ohm resistor between the electrolyzer (cathode section) and earth, and voltage measurements between the different parts of the electrolyzer (anode, anolyte, cathode, and catholyte) is used to monitor the isolation of the equipment.

To preclude the possibility of explosions associated with the electrolyzer, a scavenging air system is present. In the event of a failure of the scavenging air system, scavenging is provided by the emergency air system. The required flowrate of the scavenging air system is based on the rate of radiolysis in the electrolyzer. In addition, the potential for additional formation of hydrogen exists due to potential breakdown of hydrogen due to the presence of the voltage. Consequently, the voltage in the electrolyzer is limited.

The proposed controls will meet the single failure criteria; will be designed, procured and operated in accordance with a QA program that meets 10 CFR 50 App B and NQA-1 requirements; will meet appropriate codes and standards as described in CAR Section 11.6; and will be appropriately surveilled. This combination of commitments ensures that an over-temperature event involving the electrolyzer that results in consequences exceeding the low consequence criteria is highly unlikely. The ISA summary will demonstrate this by providing a description of the IROFS, demonstrating that the single failure criterion is satisfied, and by providing a description of the surveillance method.

Action:

Revise the CAR to reflect this information.

51. <u>Section 5.5.2.3.6.3, pp. 5.5-22 thru 5.5-24</u>

Clarify whether the 3013 canister is either a principal SSC or a defense-in-depth SSC for protection of the site worker from a load handling event.

Section 5.4.3.1.F of the SRP recommends that the applicant clearly describe the principal SSCs. Section 5.5.2.3.6.3 states that the 3013 canister is a both a principal SSC and defense-in-depth SSC for the site worker. This feature could be either a principal SSC or defense-in-depth SSC, but not both.

Response:

The 3013 Canister is a principal SSC for the site worker.

The "principal SSC" and "defense-in-depth" designations are made on an event/receptor basis. An SSC designated as "principal SSC" to protect the facility worker for any given event may also be designated as "defense-in-depth" to protect the site worker and public for the same event. SSCs designated as defense-in-depth are also principal SSCs (and fall under the 10 CFR 50 App B, NQA-1 QA program), but are not required or credited in the analysis for this event/receptor to meet the performance criteria of 10 CFR 70.61.

Action:

The CAR will be revised to reflect this information.

52. Section 5.5.2.3.6.3, pp. 5.5-22 thru 5.5-24

Clarify whether principal SSCs are applied to protect the site worker from a load handling event involving a MOX Fuel Transport Cask.

Section 5.4.3.1.F of the SRP recommends that the applicant clearly describe the principal SSCs. Section 5.5.2.3.6.3 does not mention whether or not principal SSCs are applied to mitigate the consequences to a site worker from a load handling event involving a MOX Fuel Transport Cask.

Response:

Principal SSCs are not required to protect the site worker for this event (low consequence).

Action:

The CAR will be revised to reflect this information.

53. Section 5.5.2.4.6.2, p. 5.5-28; Table 5.5-19, p. 5.5-101

Resolve the discrepancy between Table 5.5-19, which shows that waste containers are principal SSCs for the protection of site workers only, and Section 5.5.2.4.6.2, which describes waste containers as principal SSCs for protection of the public, site worker and facility worker.

Section 5.4.3.1.F of the SRP recommends that the applicant clearly describe the principal SSCs. It is not clear whether the waste containers are principal SSCs solely for the protection of site workers or for the public, site workers and facility workers because the discussion in Section 5.5.2.4.6.2 appears to contradict the entry in Table 5.5-19 for this SSC.

Response:

The waste container is required to provide protection for the facility and site workers. The safety function of the waste container is to ensure that hydrogen buildup in excess of explosive limits does not occur while providing appropriate confinement of radioactive material.

Action:

The CAR will be revised to reflect this information.

54. Section 5.5.2.4.6.10, p. 5.5-31

Verify that the delivery of chemicals does not present additional hazards to the facility.

Section 8.3 of the SRP states, "Information contained in the application should be of sufficient quality and detail to allow for an independent review, assessment, and verification by the reviewers. Some information may be referenced to other sections of the application, or incorporated by reference, provided that these references are clear, specific, and essentially complete." SRP Section 8.4.3.1 states that an application would be acceptable if it addresses the baseline design criteria for chemical safety and includes information on the chemicals, process, equipment, inventories, ranges, and limits. At the construction permit stage, this would be expected to include design bases and values for these items, with sufficient system description to allow verification of the design bases and values. Sections 8.4.3.5 B, C, D, and F of the SRP mention that design bases, process safety features, and Items Relied on for Safety (IROFS) should be included in the application.

Section 5.5.2.4.6.10 of the application briefly mentions outside explosions and mentions a mitigation strategy based upon maintaining the structure of the MOX fuel fabrication building and the emergency diesel generator building, and protecting the waste transfer line. However, the facility uses multiple chemical reagents that will require deliveries to replenish the supplies, and it is not clear if the explosion analyses have considered the potential hazards from deliveries. In particular, deliveries of hydrogen and fuel oil would seem to present predominantly explosion concerns, delivery of compressed gas cylinders might present primarily missile concerns, and deliveries of nitric acid and inert gases would present habitability concerns. In addition, events from operator activities during deliveries (e.g., overfilling) may also be of concern. The NRC would anticipate some analyses of these delivery situations, off-normal events, safe distances, their design bases and values, reliabilities, etc. In the absence of such information, it is not possible to make a safety determination.

Response:

Delivery of chemicals has been considered in the analyses to support the CAR and will be considered in the analyses to support the ISA.

A Safety Assessment (SA) has been performed to assess the safety basis of the principal SSCs at the MFFF. The SA includes an analysis of internal, man-made-external, and natural phenomena hazards. A subset of the events analyzed within the SA involve chemicals that have the potential to create event sequences that may affect nuclear safety. These chemical event sequences may originate as a result of chemical delivery, handling, storage, processing, recycling, and/or disposal. The methodology utilized to evaluate the chemical impact of these potential event sequences has been described in response to Question 113. Application of this methodology demonstrates that the release of chemicals utilized in the MOX fuel fabrication process, including the storage, delivery, and transport of these chemicals, will not affect the site worker or the public.

The analysis of potential exposures to workers performing a potential safety function is in progress and will be completed for the ISA. As stated in Section 5.5.2.10, workers mainly perform a monitoring function during emergency conditions. To ensure that workers can perform this function, the Emergency Control Room Air Conditioning system is designated as a principal SSC. Its function is to ensure habitable conditions for workers in the emergency control rooms are maintained. Upon completion of the chemical consequence analysis, measures will be provided, if necessary, to ensure that emergency control room workers are protected from potential chemical releases. Based upon the design at this point, no immediate actions have been identified for the facility worker to perform to ensure that the performance criteria of 10 CFR 70.61 are satisfied for the site worker or the public. Thus, a chemical release originating in the storage yard from any source will not require operator actions to ensure the performance criteria of 10 CFR 70.61 are satisfied. The only immediate action for a facility worker identified to date is to take action to limit their own individual dose by evacuating the area or taking other protective measures (e.g., don mask).

To address potential explosions from activities on the MFFF site external to the MOX building including delivery of chemicals, the MOX fuel fabrication building, the emergency diesel generator building, and the waste transfer line are designated as principal SSCs. These principal SSCs ensure that a high consequence event will not result from an external explosion. Explosion analyses to demonstrate that these principal SSCs provide adequate protection will be described in the ISA summary. All aspects of chemical delivery, handling, storage, processing, recycling and disposal will be considered for normal operations and off-normal and accident conditions. The analyses will include the effect from an explosion, including pressure waves, possible missiles, and habitability concerns. The response to Question 57 provides additional information related to external explosions.

Action:

The CAR will be revised to reflect this information.

55. <u>Section 5.5.2.6, pp. 5.5-33 thru 5.5-38</u>

Provide your philosophy/approach for combining independent (and dependent) natural phenomena events as well as natural events that are not "highly unlikely" with process events which are also not "highly unlikely." Parameters such as event duration and time to repair or mitigate may be taken into account to select credible combinations.

The regulatory acceptance criteria as provided in Section 5.4.3.1 (D)(ii)(a) of the MOX SRP states that the applicant's approach for the Process Hazard Analysis (PHA) should allow the applicant to examine selected accidents including natural phenomena and other types of boun~~ding accidents. In the discussion of~~ Natural Phenomena in Section 5.5.2.6, all events are listed as independent events.

Response:

For all events (excluding the design bases natural phenomena events, but including NPH events with an occurrence frequency exceeding the design bases event), the initiator is assumed to occur. Thus, the initiating event probability can be considered to be one. Therefore, combining NPH events that are not highly unlikely with process events that are also not highly unlikely is not required since the process event is assumed to occur. However, NPH events are considered as initiators of other events such as explosions or leaks.

Action:

The CAR will be revised to reflect this information.

56. Section 5.5.2.6, pp. 5.5-34 thru 5.5-38

Provide the rationale for choosing the widely varying annual exceedance probabilities for natural phenomena, for example, tornado ($2x10^{-6}$) and snow and ice events (1×10^{-2}). The explanation of rationale should include extreme local precipitation, snow and ice events, tornadoes, extreme winds, lightning, seismic events, external fires, and temperature extremes as listed in Section 5.5.2.6 of the application. Also provide a demonstration of how the selection of these various events for facility design will satisfy the performance requirements of 10 CFR 70.61. If design details are not yet available for such a demonstration, describe the approach or logic for incorporating future design information into such a demonstration.

The regulatory acceptance criteria as provided in Section 5.4.3.1 (D)(ii)(a) of the MOX SRP states that the applicant's approach for the PHA should allow the applicant to examine selected accidents including natural phenomena and other types of bounding accidents.

Response:

The MFFF selection of annual exceedance probabilities for natural phenomena events is based on the criteria established for reactors licensed under 10 CFR 50. The applicable regulatory guides specify recurrence intervals for each design bases event. The recurrence intervals are specific to each design bases event, thus resulting in different recurrence intervals for the design bases events. A discussion of each design bases event is provided below.

Demonstration that the MOX structures satisfy these requirements (i.e., structural evaluations to demonstrate the building capability during these events) as applicable will be provided as part of the ISA summary.

Earthquake

DCS has selected a design earthquake recurrence interval that is similar to the requirements for a reactor licensed in accordance with 10 CFR 50. This is consistent with the MOX Standard Review Plan (NUREG-1718), which refers to Division 1 Regulatory Guides for NPH guidance (i.e., the use of reactor-based approaches) as adapted for the risk-informed approach of 10 CFR 70 Subpart H. See the response to Question 20 for information regarding the selection of the design earthquake recurrence interval.

Wind

DCS has selected a design basis straight wind recurrence interval that is similar to the requirements for a reactor licensed in accordance with 10 CFR 50. Similar to the approach selected for earthquakes, a 1E-04 per year recurrence frequency is selected to determine the magnitude of the design (straight) wind loads. Below this probability of exceedance, tornado winds control the design associated with winds.

Tornado

DCS has selected a design basis tornado recurrence interval that is similar to the requirements for a reactor licensed in accordance with 10 CFR 50. Standard practice at nuclear reactors is to select a design basis tornado with a lower frequency than that associated with the design basis earthquake. NRC-sponsored research has indicated that some relaxation in the existing Division 1 tornado design guidance is justified (e.g., NUREG/CR-4661; see the response to Question 9). On this basis, the recurrence frequency for MFFF design tornado events is selected to be 2E-06 per year. See the response to Question 9 for additional information regarding the selection of the design basis tornado recurrence interval.

Flooding and Precipitation

DCS has selected a design basis flood and precipitation recurrence interval that is similar to the requirements for a reactor licensed in accordance with 10 CFR 50. Reactor requirements are based on the maximum probable flood. For the MFFF, the recurrence frequency for flooding and precipitation events is selected to be 1E-05 per year. Note that as described in CAR Chapter 1, the MFFF site is above the flood level associated with the 1E-05 per year flood and the maximum probable flood for the MFFF site, thus satisfying reactor requirements for this event.

Snow and Ice

DCS has selected a design basis snow and ice recurrence interval that is similar to the requirements for a reactor licensed in accordance with 10 CFR 50. For definition of snow and ice loads, building codes are typically applied. On this basis, the recurrence frequency is selected to be 1E-02 per year.

Note that as described in CAR Chapter 1, the design load associated with this event is approximately 10 psf, which corresponds to approximately 2 inches of ice. The live load allocated to the MOX building is 50 psf. Thus, significant design margin exists for this event.

Lightning

The MFFF is designed in accordance with NFPA 780-1997 and is designed for all lightning activity. Thus, no recurrence interval is selected.

Extreme Temperature

Observed temperature extremes for the Savannah River Site over the period 1961 to 1996 ranged from 107°F (42°C) to -3°F (-20°C). Temperature extremes for the Savannah River Site are postulated to occur on or near the MFFF occasionally and are considered in the design of the MFFF systems and structures.

External Fire

The MFFF is designed to account for credible external fires including forest fires, fires occurring at nearby facilities, and fires occurring on the MFFF site. Thus, no recurrence interval is selected.

Action:

The CAR will be revised to reflect this information.

57. Section 5.5.2.7.6.2, p. 5.5-40

Provide the basis of the statement that the impacts of explosions in F area are bounded by the impacts accounted for in the MFFF structures for safeguards and security reasons, including expected explosion overpressures, the basis of these estimated overpressures, and the ability of the facility to withstand the estimated overpressures.

Section 5.4.3.1 (D)(ii)(b) of the MOX SRP states that the applicant's approach for the probabilistic hazard assessment should allow the applicant to examine selected accidents including natural phenomena and other events such as fires, explosions, criticalities, radiological (or hazardous chemical as applicable under 10 CFR Part 70) exposures, and loss of containment. In Section 5.5.2.7.6.2 of the application it is stated that the impacts of explosions in F area are bounded by the impacts accounted for in the MFFF structures for safeguards and security reasons.

Response:

Explosions originating from F Area are bounded by the impacts of other events considered in the MFFF Design. This conclusion is based on a review of Savannah River Site safety documentation and MFFF design basis information, chemical inventories at F Area and MFFF, chemical deliveries to F Area, transportation routes, and distances between F-Area Process Buildings and the MFFF.

Utilizing Regulatory Guide 1.91, the safe distance between the explosion source and the MFFF can be determined from the TNT equivalent weight of a potential explosive material.

Structural loading evaluations have not been completed for the MOX facilities; however, preliminary estimates indicate the MOX building (and the emergency diesel generator building) can withstand a dynamic external overpressure greater than This is based on the structural requirements associated with other design bases events including earthquakes, tornadoes, and straight winds.

Utilizing the methodology described in Regulatory Guide 1.91, the minimum quantity of TNT equivalent material that would produce a dynamic external overpressure of 10 psi at the MOX Building was determined. Results are provided in Table 1. The distances in Table 1 represent the F-Area road and F-Area facility, respectively, that are closest to the MOX building.

As shown in Table 1, greater than of explosive material is required to reach a dynamic external overpressure of at the MOX building. According to WSRC safety documentation, a year's supply of potentially explosive material is approximately pounds. A year's supply of explosive material requires many shipments to the F-Area facility. Additionally, it is a low probability event for a transportation accident to occur at the nearest point to the MFFF. Therefore, it is concluded that a transportation accident resulting in an explosion that can produce a dynamic external overpressure greater than at the MOX building is highly unlikely.

For the facility analysis, F-Area processes have no tanks containing over of explosive material. Therefore, any explosion at an F-Area facility would produce a dynamic external overpressure at the MOX building less than 10 psi.

Utilizing the same approach, the peak pressure waves affecting the emergency diesel generator building will not exceed a dynamic external overpressure of

Final peak pressure calculations and the ability of the facility to withstand overpressures will be demonstrated during final design calculations. These will be described in the ISA summary.

Table 1. Bounding Weights Based on 10 psi Peak Over Pressure Wave and Known Distances

Weight (lb):		
Range:	**Text removed under 10 CFR 2.390.**	
Peak Reflected Press:		

Note that SRP Section 5.4.3.1 (D)(ii)(b) relates to a preliminary or process hazards analysis and not to a probabilistic hazard assessment as stated in the comment.

Action:

The CAR will be revised to reflect this information.

58. <u>Section 5.5.3.2, pp. 5.5-46 thru 5.5-47</u>

Clarify the choice of $6x10^{-4}$ as the respirable release fraction (ARFxRF) for the bounding accident consequence assessment in Section 5.5.3.2.

SRP Section 9.1.4.6.3.A recommends that the applicant use appropriate and verified assessment methods, computer codes, and literature values. Section 5.5.3.2, "Internal Fire," describes a bounding consequence assessment in which the respirable release fraction (ARF x RF) is $6x10^{-4}$. However, the reference for this value (NUREG/CR-6410, Nuclear Fuel Cycle Facility Accident Analysis Handbook) cites an ARF = $6x10^{-3}$ and an RF = 0.01 for solid, noncombustible powders exposed to thermal stress (i.e., an ARFxRF = $6x10^{-5}$).

Response:

In the evaluation of consequences at the MFFF, a single set of airborne release fractions (ARF) and respirable fractions (RF) were used for all events within a specific release mechanism (e.g., fire or drop) for a specific form (e.g., powder or liquid). To ensure that conservative consequences are estimated, bounding release fractions were established based on all the potential release events established in the Preliminary Hazards Analysis (PHA). In this specific case, for fires involving non-reactive powders, NUREG/CR-6410 does cite an ARF of 6E-03 and an RF of 0.01. However, under the technical basis (p. 3-72 of the NUREG), it is noted that some tests done with PuO_2 in a calcining furnace noted higher RF values based on temperature of the furnace. Since the MFFF has a similar calcining furnace and the PHA includes a fire event involving this furnace, the release fractions were adjusted to a more conservative value to assure bounding consequences were established for this event. Thus, the RF was increased by a factor of 10 per the technical discussion in NUREG/CR-6410.

Action:

The CAR will be revised to reflect this information.

59. **<u>Section 5.5.3.2, pp. 5.5-46 thru 5.5-47</u>**

Justify the use of a leak path factor of 10^{-4} for two banks of HEPA filters under accident conditions.

Under 10 CFR 70.22(f), the application for a license for a plutonium fuel fabrication plant must provide a description and safety assessment of the design bases of the principal structure, systems, and components of the plant. In the application the ventilation filtration system is assumed to operate and mitigate releases of radioactive material following accidents. The application states that the leak path factor for two banks of HEPA filters is assumed to be 10^{-4}. The basis for this assumption is not presented. NRC guidance in "Nuclear Fuel Cycle Facility Accident Analysis Handbook," NUREG/CR-6410, recommends that removal efficiencies of 99 percent to 95 percent be used of a series of HEPA filters that are not protected by prefilters, sprinklers, and demisters under severe accident conditions.

Response:

Introduction

DCS has evaluated the NRC guidance related to the efficiency of HEPA filters in NUREG/CR-6410, Appendix F, Section F.2.1.3, *Nuclear Grade HEPA Filters*, and believes the guidance is not applicable to the MFFF during potential accident conditions. DCS believes that NUREG/CR-6410, Appendix F, Section F.2.2 provides the appropriate guidance for determining HEPA filter efficiency during accident conditions. Following the guidance provided in Section F.2.2 and its associated references, a HEPA filter removal efficiency of greater than 99% for each stage is achievable if the HEPA filters are appropriately protected. DCS has performed initial calculations to demonstrate the HEPA filters will be appropriately protected during potential fires. Final calculations will be performed as part of final design and summarized in the ISA. These calculations will demonstrate that the HEPA filters are protected from all impacts related to fires, thus the HEPA filter removal efficiency during these conditions will be greater than 99%. These conclusions are discussed in the following paragraphs.

Part 1 - DCS has evaluated the NRC guidance related to the efficiency of HEPA filters in NUREG/CR-6410, Appendix F, Section F.2.1.3, *Nuclear Grade HEPA Filters*, and believes the guidance is not applicable to the MFFF during potential accident conditions. This conclusion is based on the following positions.

1. NUREG 6410 Section F.2.1.3 references Regulatory Guide 1.52 (NRC, 1978) for the recommendation to use HEPA filter efficiencies of 99 to 95% if conditions are severe or the filters are unprotected. This use of Regulatory Guide 1.52 for the MFFF is not applicable because the guide applies specifically to light water cooled nuclear power plants and associated accident conditions. The typical accident conditions listed in Table 1 of Regulatory Guide 1.52 show the inlet to the cleanup system has a maximum temperature of 280°F and 180°F in the primary and secondary reactor containment systems. The corresponding values for relative humidity are 100% plus condensing moisture for the primary and 100% for the secondary. The 2001 revision of Regulatory Guide 1.52 (NRC,

2001) also includes BWR Main Steam Line Break, Fuel Damage or Pre-incident Spike, Equilibrium Iodine Activity, BWR Rod Drop Accident, PWR Steam Generator Tube Rupture, Fuel Damage or Pre-incident Spike, Coincident Iodine Spike, PWR Main Steam Line Break, Fuel Damage or Pre-incident Spike, Coincident Iodine Spike, PWR Locked Rotor Accident, PWR Rod Ejection Accident, and Fuel Handling Accident. These reactor events are associated with water vapor, and the predominant source of the radioactive releases are noble gases and iodine gases. These conditions are not the same that would exist during a potential fire event at the MFFF. Part 2 of this response describes potential conditions associated with a fire at the MFFF.

Additionally, the guide recommends a leak path factor of 10^{-2} for particle removal for a cleanup system consisting of two HEPA filters in the cleanup train. The description of the credit taken for HEPA filters is ambiguous. Section B, page 1.52-2 states the upstream "HEPA filters remove the fine discrete particulate matter and pass the air stream to the adsorber" while "HEPA filters downstream of the adsorption units collect carbon fines and provide redundant protection against particulate release in case of failure of the upstream HEPA bank." Section C5 on page 1.52-5 states that "HEPA filter sections should be tested in place....to confirm a penetration of less than 0.05% at rated flow." It is not clear which HEPA filter or if both are credited for the allowed 99% particle removal efficiency in accident dose evaluations.

The ambiguity is cleared up in the 2001 revision of Regulatory Guide 1.52 (NRC, 2001). It states on page 1.52-4 that "The HEPA filters remove the fine discrete particulate matter from the air stream. A HEPA filter or a medium efficiency post filter...downstream from the adsorption units collects carbon fines and provides additional protection against particulate matter release in case of failure of the upstream HEPA filter bank. It is not necessary to perform in-place leak testing on postfilters or HEPA filters downstream from the carbon adsorbers." Thus, it is clear from the revised NUREG-1.52 that the first HEPA filter stage is credited with a 99% particle removal efficiency, not the entire system. The guide does not discuss the value of additional stages. Further, the value of additional stages is provided by Bergman et al., 1995. Studies referenced by Bergman have shown that each filter stage in a system with multiple filter stages provides a removal efficiency of greater than 99%. The report concludes that it is appropriate to credit each stage of a system if the filters are adequately protected from the potential effects of a fire.

2. NUREG/CR-6410 , Section F.2.1.3, page F-7 states "If a series of HEPA filters is protected by pre-filters, sprinklers, and demisters, efficiencies of 99.9% for the first filter and 99.8% for all subsequent filters is recommended for accident analysis (Elder et al., 1986)." Although this guide is an easy prescription to use for determining HEPA efficiencies, it is not based on experimental studies. The Elder et al., 1986 report is based on a meeting between officials from the Atomic Energy Commission and Albuquerque Operations Office held in Albuquerque, NM on December 9, 1971 (US AEC, 1971). "These guidelines represented the opinions of the meeting attendees, and were not supported by technical data." (Bergman et al., 1995).

3. The final sentence of NUREG/CR-6410, Section F.2.1.3 is not correct as written. It states: "If conditions are severe or the filters are unprotected, efficiencies as low as 99 to 95 percent are recommended (USNRC 1978)." The USNRC reference is Regulatory Guide 1.52. However, Regulatory Guide 1.52 does not state any efficiency value other than 99%. Elder et al., 1986, which is also cited in the same section, does not give efficiency values less than 99.8%.

   It appears that the 95% to 99% efficiency values were incorrectly attributed to Regulatory Guide 1.52, but were taken instead from Bergman et al., 1995, based on NUREG/CR-6410, Section F.2.2, page F-11: "The limiting assumption is that the filter efficiency drops to zero upon failure, but data is cited that indicates filter efficiency can remain at 94 percent to 99 percent for partial failure (separation of media from frame, tears in the media)." According to Bergman et al., 1995, "Previous publications had assigned a single value for the efficiency of HEPA filtration systems under all accident conditions. However, the publications reviewed in this paper demonstrate that the efficiency of HEPA filters will vary greatly depending on the operating conditions, thereby requiring a case-by-case analysis."

   (Note that DCS incorrectly cited the use of Regulatory Guide 1.52 (NRC 1978) in Section 5.5.3.2 of the CAR, stating "The bounding leak path factor associated with these systems is 1×10^{-4} (NRC 1978a)." DCS made an editorial error in the CAR report and should have cited NUREG/CR-6410, Section F.2.2. This error will be corrected.)

4. NUREG/CR–6410 also implies that sprinklers should be used to protect the HEPA filters. The decision not to incorporate HEPA filter water sprays into the MFFF design is based on the following discussion.

   DCS understands that *Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility* (NUREG-1718, Appendix E) recommends that DOE Standard *Fire Protection Design Criteria*, DOE-STD-1066-99, be used for the HEPA filter plenums. This standard calls for the use of water sprays and demisters to reduce the potential fire damage to HEPA filters. Considering the use of water sprays and demisters requires weighing the advantages of the decreased exhaust temperature with the increased potential for HEPA filter damage due to filter wetting. The potential for ~~HEPA~~ filter damage from the water spray system is recognized in Section 14.8.1.4 of DOE-STD-1066-99.

   Bergman et al., (1997) questioned the efficacy of using water sprays in protecting final HEPA filters from fires. They showed a series of full-scale controlled studies where the use of water sprays, even with demisters, can lead to structural damage to HEPA filters. Water sprays add moisture to the HEPA filters and thereby decrease the filter media strength and also increase the filter pressure drop. The increased pressure drop is exacerbated by smoke and other particle deposits. With the decreased strength and higher pressure drop, the filter media can burst if the fan pull is sufficient. Bergman et al., (1995) determined that the differential pressure threshold for rupture of wet filters is 10 inches of water (Bergman et al., 1995; NUREG/CR-6410, Section F.2.2, Table F-5.).

Bergman et al., (1997) also reviewed studies conducted in the 1970s that showed water spray systems were effective in lowering the exhaust gas temperature, but did not show the benefit in protecting HEPA filters from the temperature and smoke environment of fires. Based on these early studies, fire protection engineers incorporated water sprays and demisters into HEPA filter plenums to protect HEPA filters against fires. These practices have been incorporated into DOE Standard on Fire Protection Design Criteria (DOE-STD-1066-99). Bergman et al., (1997) analyzed a fire that occurred in 1980 at the Rocky Flats Plant inside a HEPA plenum and concluded that the greatest damage to the HEPA filters was due to the water spray that resulted in three of the four filter stages being blown out.

DOE Standard-1066-99 Appendix B, *Operating Temperatures for HEPA Filters* indicates that there is a rapid decrease in the tensile strength of the HEPA filters above 234°C. For the reasons stated above, DCS believes that if HEPA filter temperatures are less than 210°C during potential fire accidents without the aide of sprinklers, and sprinklers should not be incorporated into the design.

Part 2- Following the guidance provided in Section F.2.2 and its associated references, a HEPA filter removal efficiency of greater than 99% for each stage is achievable if the HEPA filters are appropriately protected. DCS has performed initial calculations to demonstrate the HEPA filters will be appropriately protected during potential fires.

NUREG/CR-6410 Section F.2.2 provides a systematic method for determining the efficiency of HEPA filters that are subjected to a large variety of environmental conditions. By computing the environmental conditions associated with specific accident scenarios, the filter efficiencies that correspond to these conditions can be obtained from tables of experimental values of filter efficiency at various environmental conditions. Two steps are involved: (1) determine if the HEPA filters will be structurally damaged, and (2) if not, determine the appropriate removal efficiency.

A preliminary assessment is made to determine whether the potential conditions could result in structural damage to the HEPA filter using Table F-5 of NUREG/CR-6410. If the filter is structurally damaged, Bergman et al., (1995) recommends a 0% efficiency be conservatively assigned to the filter because of the large variability in data from small damage to major damage. NUREG/CR-6410 recommends the use of actual residual efficiency values instead of 0%. Calculations performed for the CAR (described below) indicate that the final HEPA filters will not be structurally damaged during a fire event.

In determining the filter efficiency, each event must be analyzed to determine its effect on the respective HEPA filters. In the case of an area fire at the MFFF, three mechanisms that may affect MFFF HEPA filter efficiency were identified. These include:

- Branding (i.e., hot embers burning through filter media);
- Excessive soot loading; and
- Prolonged exposure to high temperatures.

MFFF systems are designed to protect the HEPA filters from these effects as described in the following paragraphs:

Branding - Spark arresters are installed upstream of the HEPA filters to prevent hot particles from impacting the filter media. Additionally, large particles will be removed from the ventilation stream by gravitational settling, while smaller particles will cool and reach thermal equilibrium prior to impacting the filter media.

Soot Loading - The nature of the soot aerosols generated during a fire depends on the type of fuel being burned and the ventilation state of the fire. In general, the dynamics of a chamber fire that represents a typical room in a nuclear fuel fabrication facility changes initially from a well-ventilated fire that generates small, relatively solid aerosols to a ventilation limited fire in which larger and more liquid aerosols are generated (Alvares et al., 1979, 1981). The impact of these changes on filter clogging is dramatic with the aerosols produced in well-ventilated fires having little clogging potential compared to the aerosols produced in under-ventilated fires. The type of fuel affects the combustion process as well. Liquid pool and solid crib fires have been modeled to allow general estimates of the combustion process and the aerosol emission. Filter clogging models are also available to estimate clogging by the different aerosols (Bergman et al., 1979, 1983).

An analysis was performed by DCS to address filter soot loadings produced by specific area fires. The area with the largest combustible loading was selected, and the soot produced was determined considering the combustible types and fire efficiency. Various materials of construction and solvent were considered. Soot deposition in the ducting was also considered. Soot deposition, filter soot loadings, and plugging conditions were determined in accordance with Ballinger (1988). The analysis concluded that the worst area fire did not result in plugging of MFFF HEPA filters. See the response to Question 148 for additional information.

The soot analysis did not credit the effect of prefilters. However, the system design uses high strength prefilters as necessary. High strength prefilters (including spark arresters) reduce the load (minimize the pressure drop) on HEPA filters since they have been shown to be very effective in protecting HEPA filters from the worst-case smoke loading (Alvares et al., 1979, 1981). The high strength prevents prefilter collapse during the filter clogging.

High Temperature - An analysis of fire area exhaust flow dilution was performed by DCS to determine possible maximum temperatures at the HEPA filter during an area fire. The analysis assumed an affected room exhaust temperature of 2,300°F and no heat transfer in the ducting between fire and filter. Maximum dilution air temperature was also assumed (104°F room/119°F glovebox). Possible adjacent areas participating in the fire were also considered. The analysis concluded that the diluted flow temperature for any of the four MFFF exhaust systems is less than the HEPA filter continuous exposure higher temperature of 400°F (204°C). See the response to Question 146 for additional information.

This temperature is less than the 234°C where there is a rapid decrease in tensile strength as specified in Appendix B, Operating Temperatures for HEPA filters of DOE-STD-1066-99. It is also at a temperature at which there is no affect on filter penetration as shown in Table F-3 of NUREG/CR-6410. By ensuring the temperature at the HEPA filters during accidents is below 210°C, Table F-3 shows that each HEPA filter will have an efficiency of 99.8%.

The analyses discussed above support a conclusion that MFFF final HEPA filters will not breach as a result of the heat and soot/smoke produced in an area fire. In accordance with Table F-3 of NUREG/CR-6410, Appendix F, the expected temperatures during a fire support a removal efficiency of 99.8%. As a result, a removal efficiency of 99% for each of two stages of HEPA filters is justified for use in the CAR.

Conclusion

Each of the MFFF HEPA final filters credited in the safety analysis are designed and tested to be at least 99.97% efficient at 0.3 micrometer diameter particles. The filters are tested in accordance with ASME N510-1995 (Testing of Nuclear Air-Treatment Systems). During a potential fire, the final HEPA filters are protected from the effects of the fire to ensure that they are not structurally weakened and to ensure that they maintain a high efficiency. These measures include prefilters, spark arresters, location of filters as far as practical from postulated fires; separation of redundant trains; small fire areas which limits soot loading on filters; and adequate mixing of exhaust air to ensure filter inlet temperatures do not challenge the filters. Additionally, the prefilters, spark arresters, and HEPA filters are made of non-combustible materials, and no ignition source is located inside the filter housing to initiate a fire. Calculations for the CAR have shown that the filters can be protected from the effects of a potential fire. Thus, using the guidance of NUREG 6410, Section F.2.2 and the associated reference (Bergman 1995), a removal efficiency of greater than 99% (99.8%) for each of two stages of HEPA filters is justified. Final calculations to demonstrate that the final HEPA filters are protected from the effects of a potential fire will be provided in the ISA.

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NRC, 1978. *Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants*, Regulatory Guide 1.52, Revision 2, U.S. Nuclear Regulatory Commission, 1978.

NRC, 2001. *Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants*, Regulatory Guide 1.52, Revision 3, U.S. Nuclear Regulatory Commission, 2001.

US AEC, 1971. "Operational Safety Division/Engineering and Construction," Unnumbered report.

Action:

Revise Section 5.5.3.2 to refer to NUREG/CR-6410 in lieu of Regulatory Guide 1.52 as mentioned in item 3 above.

60. <u>Section 5.5.3.5, pp. 5.5-48 and 5.5-49</u>

Describe the "Explosion Event" in more detail.

Section 8.3 of the SRP states, "Information contained in the application should be of sufficient quality and detail to allow for an independent review, assessment, and verification by the reviewers. Some information may be referenced to other sections of the application, or incorporated by reference, provided that these references are clear, specific, and essentially complete." SRP Section 8.4.3.1 states that an application would be acceptable if it addresses the baseline design criteria for chemical safety and includes information on the chemicals, process, equipment, inventories, ranges, and limits. At the construction permit stage, this would be expected to include design bases and values for these items, with sufficient system description to allow verification of the design bases and values. Sections 8.4.3.5 B, C, D, and F recommend that design bases, process safety features, and IROFS be included in the application.

Section 5.5.3.5, "Explosion Event," discusses an internal explosion from either over-pressurization or due to potentially explosive mixtures. No energies, pressures, or other parameters are given. No reliability values are provided to demonstrate that the event would be highly unlikely. The application states the radioactive materials are filtered prior to their release, and uses a bounding leak path factor of $1x10^{-4}$. However, HEPA filters are unlikely to survive an explosion of the magnitude implied by the text, and the leak path factor is likely to be orders of magnitude larger. In addition, there may be a substantial release of highly respirable material from the damaged or destroyed HEPA filters. More information and design bases are needed before a safety determination can be made.

Response:

As described in Chapter 5, Section 5.5.2.4 of the CAR, explosions are identified as potential events within the MOX building from process operations, from operations external to the MOX building from nearby facilities and the storage of chemicals on the MFFF site, and from laboratory operations.

The design basis for potential explosions that could occur within the MOX building from process operations is to prevent them. Chapter 5 identifies several potential categories of internal process explosions and provides the principal SSCs identified to prevent them. HAZOP evaluations will be performed as part of the ISA to identify specific causes of these explosions and to identify specific controls to prevent them. As described in response to Question 39, these controls are IROFS and will ensure that internal explosion events are highly unlikely.

The design bases for potential explosions that could occur external to the MOX building is to ensure that principal SSCs are protected from the impact of these explosions. Chapter 5 identifies several potential categories of external explosions and provides the principal SSCs identified to ensure that other principal SSCs are protected from the impacts of external explosions. Calculations involving energies, pressures, distances, building structures, etc. will be performed as part of the ISA to demonstrate the effectiveness of the principal SSCs specified for this event.

Pages 61-17 through 61-21 removed under 10 CFR 2.390.

The design bases for potential explosions in the laboratory are to ensure that they do not impact process operations (from a safety perspective), and to ensure the ventilation system can mitigate any direct radioactive material release from the laboratory. This commitment will be demonstrated in the ISA. Calculations will be performed to demonstrate that laboratory explosions and the resultant pressure waves will not impact process operations, and to demonstrate the effectiveness of the ventilation system following a laboratory explosion.

The explosion event discussed in Section 5.5.3.5 is a hypothetical explosion event that occurs within the MOX building. It is labeled as a hypothetical event since as described above, internal process related explosions will be made highly unlikely and potential external and laboratory explosions will not impact process operations. The discussion is presented to provide a commitment to defense in depth and is beyond what is required to satisfy the requirements of 10 CFR 70.61. The evaluation shows that if an internal explosion were to occur and involve the maximum quantity of radioactive material within a process cell, the consequences would be low. As part of the ISA, this commitment will be demonstrated. Potential bounding internal explosions similar to those described in Chapter 5 will be identified and evaluated during the ISA process to determine potential pressure waves and their impact on the facility and the ventilation systems. The evaluations are expected to show that the facility structure, process cell walls, and final HEPA filters are not significantly damaged by the explosion event.

Action:

The CAR will be revised to reflect this information.

61. <u>Table 5.5-2, pp. 5.5-60 thru 5.5-67</u>

Provide information to correlate specific events or compartments containing events as identified in Appendix 5A with the radioactive inventory as listed in Table 5.5-2. Location descriptions or identifiers should also be adequate to locate events on drawings showing fire barriers, ventilation zones and components, and other features affecting safety. For example, from an integrated safety perspective, the staff will need to evaluate events similar to FW-1 and FW-2 (Table 5.5-12), which pertain to fires involving more than one fire area and systems that cross fire areas, respectively.

Section 5.4.3.1 (E)(i) of the MOX SRP recommends that the applicant's safety assessment of the design bases includes a hazards analysis that identifies the approximate location and quantities of Special Nuclear Material (SNM) and other hazardous materials. The staff will need to locate analyzed events and areas for potential events in order to evaluate the applicant's analysis and to determine if there is any potential for additional consequences due to common mode failures, fire spread, or other effects that will need to be determined from an integrated assessment.

Response:

The events in Appendix 5A apply for each process unit or workshop identified in the item labeled "specific location." Events that impact individual locations are evaluated for each glovebox, AP vessel, or other sub-unit within the specified process unit or workshop based on the MAR provided in Table 5.5-2. For fire events, the evaluation is based on the total MAR within a fire area.

Additional information is attached to more clearly correlate events in Appendix 5A with the information in Table 5.5-2 and specific facility locations. Table 5.5-2 has been revised and now includes room numbers and fire areas for cross-reference to the process conceptual layouts provided in CAR Chapter 11.1 and the confinement zone maps provided in CAR Chapter 11.4 (Confinement zones for each room are identified on the process conceptual layouts provided in CAR Chapter 11.1). Tables providing the damage ratio and the ARF x RF values used in the consequence evaluation are attached. A table providing the MAR for each fire area is attached and is consistent with the fire area information provided in CAR Chapter 7. An updated chemical inventory is provided in response to Question 113.

Note that Table 5.2-2 has also been updated to include the depleted uranium inventory in the secured warehouse, and the uranium inventory in the AP process once it has been separated from the plutonium, and to more precisely list the potential location of material in the receiving and shipping areas. Also note that these numbers have been revised after the response to the NRC's Request for Additional Information concerning the Environmental Report was submitted.

Action:

The CAR will be revised to reflect this information.

Pages 61-2 through 61-14 removed under 10 CFR 2.390.

Table 2. Release Fractions Used in SA

Release Fractions:							
		Release Mechanism					
Release Form	Release Fraction	Explosive Detonation	Explosive Over-Pressurization	Fire/Boil	Drop	Entrainment (/hr)
Solution	ARF	1.0	5.0E-05	2.0E-03	2.0E-05	4.0E-07
	RF	0.01	0.8	1.0	1.0	1.0
	ARF x RF	1.0E-02	4.0E-05	2.0E-03	2.0E-05	4.0E-07
Powder	ARF	1.0	5.0E-03	6.0E-03	2.0E-03	4.0E-05
	RF	0.2	0.3	0.1	0.3	1
	ARF x RF	2.0E-01	1.5E-03	6.0E-04	6.0E-04	4.0E-05
Pellet	ARF	0.01	5.0E-03	5.0E-04	1.0	NA
	RF	1.0	0.3	0.5	1.1E-05	
	ARF x RF	1.0E-02	1.5E-03 *	2.5E-04	1.1E-05	
Rod	ARF	0.01	0.1	0.0	1.0	NA
	RF	1.0	0.5	1.0	1.1E-05	
	ARF x RF	1.0E-02	5.0E-02 **	0.0E+00	1.1E-05	
Filter (un-encased)	ARF	2.0E-06	1.0E-02	1.0E-04	1.0E-02	NA
	RF	1.0	1.0	1.0	1.0	
	ARF x RF	2.0E-06	1.0E-02	1.0E-04	1.0E-02	

* Pellet damage ratio used below for pellet exposed to over-pressurization flows/forces

** Rod damage ratio used below for breach of pressurized rod

ARF – Airborne Release Fraction

RF – Respirable Fraction

Table 3. Damage Ratios used in SA

Damage Ratios:		
Release Mechanism	**Form**	**DR**
Explosive Over-Pressurization	pellet	0.01
Explosive Over-Pressurization	Rod	0.001

Damage Ratio of one used for other materials and events.

62. Table 5.5-9, p. 5.5-86; Table 5.5-12, p. 5.5-90; Table 5.5-15, p. 5.5-95; and 5.5-18, p. 5.5-100; Appendix 5A

Explain the difference between the events provided in the tables in Appendix 5A and those listed in Tables 5.5-9, 5.5-12, 5.5-15, and 5.5-18. Explain if the events listed in Tables 5.5-9, 5.5-12, 5.5-15, and 5.5-18 are a subset of those tables in Appendix 5A.

Section 5.4.3.1 (E) of the MOX SRP recommends that the applicant's safety assessment of the design basis shows that the design and design bases will result in a facility that will meet the performance requirements of 10 CFR 70.61 and the defense-in-depth requirements of 10 CFR 70.64(b). The SRP also recommends that the safety assessment of the design basis should include a hazards analysis. Clarification of the lists of events is necessary to understand the hazard analysis which is included to support the application.

Response:

Events in Tables 5.5-9, 5.5-12, 5.5-15, 5.5-18, 5.5-23, and 5.5-25 are subsets of the total list of events from the hazard assessment provided in Appendix 5A. These tables provide a cross-reference to the events and event types in Appendix 5A. All events listed in Appendix 5A are discussed in the corresponding sections of Section 5.5 according to event type, except for low consequence events. Low consequence events are identified in Table 5.5-25 and discussed in Section 5.5.2.11.

Action:

The CAR will be revised to reflect this information.

63. <u>Table 5.5-9, p. 5.5-86; Table 5.5-12, p. 5.5-90; Table 5.5-15, p. 5.5-95; and 5.5-18, p. 5.5-100;</u>

Provide the calculated consequences for all hazard assessment events as listed in Tables 5.5-9, 5.5-12, 5.5-15, and 5.5-18. Also provide the parameters used in equations 5.4-1 and 5.4-2 to calculate the consequences.

Section 5.4.3.1 (D)(iv) of the MOX SRP recommends that the applicant's approach for assessing the consequences of accidents be consistent with the acceptance criteria in Section 5.4.3.2 (B)(v)(c). The consequences and parameters need to be provided for the staff to evaluate the approach.

Response:

For the CAR, hazard assessment events are categorized by their unmitigated consequence into one of two categories: low consequence or above low consequence. The consequence threshold is based on the performance criteria of 10 CFR 70.61 as described in Section 5.4 of the CAR. For low consequence events, no principal SSCs are required or identified. For events whose consequences have the potential to exceed the low consequence criteria of 10 CFR 70.61, principal SSCs are identified to mitigate the consequences or make the event highly unlikely. Section 5.5.2 of the CAR identifies hazard assessment events that require the application of principal SSCs, and thus identifies the consequence category for that event.

Numerical values for the bounding mitigated consequences are provided in CAR Section 5.5.3. These consequences are determined by calculating the potential unmitigated consequences for all hazard assessment events. The events with the potential to produce the largest unmitigated consequences for each event type are designated as the bounding events. The bounding consequence event then applies HEPA filters to determine the mitigated consequences. As part of a conservative deterministic approach to establishing a safety basis, no credit for any other mitigating features and no other reduction in the source system is applied. Thus, these calculations can be used to determine the upper bound for all unmitigated event consequences (note that the dose reduction factor for the HEPA filters is less than a factor of 100 for the criticality event as over 99% of the mitigated consequence is based on the dose associated with noble gases).

The methodology for performing the consequence calculations is provided in Section 5.4. All terms are identified and values are either referenced in the CAR, or provided, except for damage ratios, and ARFxRF values. These values are provided in response to Question 61. In addition, clarified source term information has been provided in response to Question 61.

It is important to note that the results associated with the bounding events presented in the CAR are based on conservative, deterministic assumptions. Use of best estimate methodology would produce results that are several orders of magnitude lower than those presented in the CAR.

Action:

The CAR will be revised to reflect this information.

64. Table 5.5-10, pp. 5.5-87 and 5.5-88; Table 5.5-13, pp. 5.5-91 and 5.5-92; Table 5.5-16, pp. 5.5-96 and 5.5-97

Provide a description of the training and procedures to be relied on as SSCs and provide estimates of the likelihood of these procedures to be incorrectly followed or to fail to provide the intended mitigation.

Section 5.4.3.1 (D)(iii) of the MOX SRP recommends that the applicant's approach for applying likelihoods to the accidents should show how each principal SSC acts to prevent or mitigate the accident. The staff will need to know the likelihood of events with consequences that could exceed regulatory limits to determine if the plant will meet the performance requirements of 10 CFR 70.61. Tables 5.5-10, 5.5-13, and 5.5-16 list principal SSCs for facility worker protection from various event categories. Some of the event groups within the categories contain training and procedures as their principal SSCs.

Response:

A safety assessment has been performed as part of the CAR to identify hazards, event sequences, and principal SSCs to meet the performance requirements of 10 CFR 70.61. For some events, the principal SSC of "Training and Procedures" has been identified as being required to meet the performance requirements of 10 CFR 70.61 for the facility worker. In all cases, "Training and Procedures" are meant to address the actions that the facility workers take to protect themselves by evacuating the area or otherwise minimizing dose (e.g., donning of mask) following the occurrence of an event. See the response to Question 54 for additional information.

Event likelihood is discussed in response to Question 39.

The human factors engineering program, and training and procedures are described in CAR Chapters 12 and 15, respectively. These programs provide the framework for the development of specific procedures and training to be provided to meet the performance requirements of 10 CFR 70.61.

Action:

The CAR will be revised to reflect this information.

65. <u>Table 5.5-10, pp. 5.5-87 and 5.5-88; Table 5.5-11, p. 5.5-89; Table 5.5-13, pp. 5.5-91 and 5.5-92; Table 5.5-14, pp. 5.5-93 and 5.5-94; Table 5.5-16, pp. 5.5-96 and 5.5-97; Table 5.5-17, pp. 5.5-98 and 5.5-99; Table 5.5-19, pp. 5.5-101 and 5.5-102</u>

In addition to information provided in response to the comment above, provide failure or reliability estimates (ranges would be sufficient) for the principal SSCs as listed in Tables 5.5-10, 5.5-11, 5.5-13, 5.5-14, 5.5-16, 5.5-17, and 5.5-19. If a failure or reliability value can not be determined from the description of the SSC, then a target value for desired maximum failure probability or minimum reliability should be provided.

Section 5.4.3.1 (D)(iii) of the MOX SRP recommends that the applicant's approach for applying likelihoods to the accidents should show how each principal SSC acts to prevent or mitigate the accident. The staff will need to know the likelihood of events with consequences that could exceed regulatory limits to determine if the plant will meet the performance requirements of 10 CFR 70.61.

Response:

The response to this question is provided in the general response to Question 39.

Action:

None

66. <u>Table 5.6-1, pp. 5.6-5 thru 5.6-11</u>

Describe the safety functions that are allocated directly or indirectly to software components.

Per Section 11.3 of the SRP, the Safety Analysis should address how potential failure modes are analyzed, including consideration of communication failures, common-mode failures, and human errors. Software is a potential common-mode and it is not clear from the table if it was considered. Software is frequently overlooked during system hazard analyses, but this should not occur in safety applications. Software hazard analysis controls software hazards and hazards related to interfaces between the software and the system interfaces between the software and the system (including hardware and human components.) It includes analyzing the requirements, design, implementation, user interfaces and changes. The description of methods, practices, or standards concerning the allocation of system hazards to software components and their inclusion in the requirements is missing in Table 5.6-1 of the application.

Per section 11.3 and 11.5.1 of the SRP, a determination should be made if the design basis adequately addresses the specific criteria of how potential failure modes are analyzed to include the effects of communication failures, common mode failures, and human errors. Information is needed in order to evaluate the applicant's commitment to provide plant systems that satisfy the acceptance criteria.

Response:

HAZOPs and other process hazard analyses techniques are being performed as part of the ISA to identify specific causes of events and associated prevention features (IROFS). Software failures including communication failures, common mode failures and human errors are included in the analysis. To date, criticality prevention related to material inventory control is the only safety function that has been allocated to software.

In the design of the control systems for the MFFF, DCS will minimize the use of software programmable electronic devices in applications that are IROFS. Where software programmable electronic systems are used as IROFS, the software programmable electronic system will not be the single element of a protection scheme. The configuration of the software driven control system for the MFFF is a multi-level system. The different levels use different software, different hardware, different communication links, and different control links. The different levels are not dependent upon the other to perform their intended functions.

Part of the design criteria for the control system of the MFFF is that the control system will include redundant and/or diverse instrument channels. The instrument channels will be designed with separation to ensure that there is no single failure vulnerability.

Methods, practices, codes and standards applicable to MFFF software systems are described in CAR Chapter 11.6. Application of the codes and standards identified ensure that the design adequately addresses communication failures, common mode failures and human errors.

Action:

The CAR will be revised to reflect this information.

CHAPTER 6, NUCLEAR CRITICALITY ANALYSIS

67. Chapter 6, General

Explain the words "as practical" or "as needed" as used throughout this chapter. Provide explicit criteria explaining who makes the determination whether following a design principle is practical or necessary, and how the determination is made.

Explicit identification of how the design philosophy will be applied is necessary to ensure that the design bases will provide reasonable assurance of protection against a criticality accident.

Response:

The determination of whether following a design principle is practical is made in the nuclear criticality evaluation program during the NCSEs. These analyses are subjected to numerous technical and managerial reviews. Specifically, there is an independent reviewer, a technical supervisor review, a design verification review, plus management reviews.

Specific instances of the words "practical" and "needed" and their explanation are as follows:

- Page 6-2: "Management measures provide reasonable assurance that items relied on for safety (IROFS) will be available and reliable to perform their designated safety functions *when needed*."

  The phrase, "when needed" here refers to the need for IROFS being available whenever they are required to perform their intended safety function.

- Page 6-4: "These reviews will be conducted, in consultation with operating personnel, by MFFF staff who are knowledgeable in nuclear criticality safety and who (*to the extent practicable*) are not immediately responsible for operations."

  The phrase "to the extent practicable" refers to criticality safety personnel that are independent of operations. However, sometimes the most knowledgeable personnel are those who are involved in operations. This will be handled on a case-by-case basis.

- Page 6-5: "*Where practicable*, reliance is placed on equipment design that uses passive engineered controls rather than on administrative controls."

  See response to Question 72.

- Page 6-8: "*Where practicable*, reliance is placed on equipment design that uses passive engineered controls rather than on administrative controls."

  This is the same sentence as the previous one and is a restatement of general principles. See response to Question 72.

- Page 6-9: "In terms of assumed reliability of criticality controls, NCSEs for the MFFF will consider controls of higher hierarchical preference, *to the extent practical*, to provide correspondingly higher reliability when assessing criticality risks and demonstrating compliance with the double contingency principle."

  The phrase "to the extent practical" here refers to the hierarchical preference in the design of the MFFF of passive engineered controls, such as geometry, over active engineered controls, to be the most reliable and, therefore, preferable. However, there are some situations, such as in the powder mixing units, where a fixed geometry can not be assured. In those cases, active controls are used. The use of such controls will be justified in the NCSEs.

- Page 6-14: "Justification for the use of moderation control, *when needed*, is provided in NCSEs and the ISA Summary."

  The phrase, "when needed" here refers to the fact that the use of moderation control will be justified in the NCSEs. This will be done whenever, the criticality safety can not be demonstrated independently from all conditions of moderation.

- Page 6-27: "Moderation control is used for some process equipment when *its needed capacity* is not compatible with mass control alone, such as equipment in the Powder Area and some units in the Pellet Process Area and Fuel Rod Process Area."

  The phrase, "its needed capacity" refers to the design, or required, capacity of the process.

- Page 6-30: "Two types of homogeneity *can be needed*."

  The phrase, "can be needed" here refers to the fact that in some MOX Process (MP) units, it is important to adequately mix PuO_2 and powders and adequately mix the powder blend and the small quantities of organic additives. Process controls are involved here and the unit will be demonstrated safe by the NCSEs.

- Page 6-35: "Benchmark experiments are selected that resemble *as closely as practical* the systems being evaluated in a design application in all characteristics, such as system configuration (i.e., rod lattice versus homogeneous solution), moderator characteristics, fuel material composition (e.g., ^{239}Pu content) and density, moderator-to-fuel ratio, multiple fuel unit interaction, presence and form of strong neutron-absorbing materials, and reflector characteristics."

  The phrase, "as closely as practical" here refers to the goal that the selected benchmark experiments match the corresponding MFFF situation. However, there are only a limited set of benchmark experiments available (i.e., in the international benchmark handbook). While there are many benchmarks applicable to the MFFF, DCS will use the most appropriate set of experiments. In the case where there is some difference between the design application and the benchmark experiments, this difference will be discussed and the use of the benchmark set justified.

- Page 6-36,-37: "*Where practical*, criticality is precluded by demonstrating that the design is subcritical without the need to implement controls, or by making appropriate design changes to render criticality non-credible. In those cases in which it *is not practical* to make criticality non-credible, criticality control parameters are selected and limits on these parameters are established."

  The phrase "where practical" here refers to the MFFF project goal of using passive features, such as geometry, to ensure criticality safety. The use of such controls usually ensures that a criticality will not be credible and therefore is desirable. The phrase, "is not practical" here refers to the situations where criticality safety can not be assured using such passive features. The use of non-passive features to demonstrate criticality safety will be justified in the NCSEs.

- Page 6-39: "*To the extent practical*, process designs will incorporate sufficient features such that they can be demonstrated subcritical under both normal and credible accident conditions."

  The phrase, "to the extent practical" here should be deleted. MFFF process designs <u>will</u> incorporate sufficient features such that they can be demonstrated subcritical under both normal and credible accident conditions.

Action:

The phrase "to the extent practical" on page 6-39 will be deleted in the next update of the CAR. Otherwise, no change.

68. Section 6.1, pp. 6-1 and 6-2

Describe the qualifications and duties of the Nuclear Criticality Safety (NCS) staff during the design phase.

10 CFR 70.22(f) states, in part: "Each application for a license to possess and use special nuclear material in a plutonium processing and fuel fabrication plant shall contain...a description and safety assessment of the design bases of the principal structures, systems, and components of the plant..." 10 CFR 70.62(a) states that each licensee or applicant shall establish and maintain a safety program that demonstrates compliance with the performance requirements of 10 CFR 70.61. Nuclear criticality safety is an important area for the safety assessment of the design bases of the principal structures, systems, and components and for the safety program that demonstrates compliance with the 10 CFR 70.61performance requirements. SRP Section 6.4.3.1.A recommends that the applicant describe positions, responsibilities, experience, and qualifications of persons responsible for NCS.

The NCS Organization and Administration is described for the operations phase, but not for the design phase except for the statement that the NCS function is within the "design engineering organization". Qualified staff are necessary to ensure that the design bases will provide reasonable assurance of protection against a criticality accident.

Response:

The qualifications of the criticality safety function during the design phase are analogous to those during the operations phase (see Section 6.1 of the CAR).

The MFFF Facilities Design Manager is responsible for the design of the facility and site-related interfaces for the MFFF, including the nuclear safety discipline that encompasses the criticality safety function, during the design phase. The criticality safety function is responsible for the following during the design phase:

- Establish the Nuclear Criticality Safety design criteria;
- Provide criticality safety support for integrated safety analyses and configuration control;
- Assess normal and credible abnormal conditions;
- Determine criticality safety limits for controlled parameters;
- Develop and validate methods to support nuclear criticality safety evaluations (NCSEs);
- Perform criticality safety calculations and write NCSEs;
- Specify criticality safety control requirements and functionality;
- Provide advice and counsel on criticality safety control measure; and
- Support emergency response planning.

The minimum qualifications for a criticality safety function manager are a Bachelor of Science (BS) or Bachelor of Arts (BA) degree in science or engineering with at least two years of nuclear industry experience in criticality safety. A criticality safety function manager must understand and have experience in the application and direction of criticality safety programs. A criticality safety function manager has the authority and responsibility to assign and direct activities for the criticality safety function.

The minimum qualifications for a senior criticality safety engineer are a BS or BA degree in science or engineering with at least two years of nuclear industry experience in criticality safety. A senior criticality safety engineer has the authority and responsibility to conduct activities assigned to the criticality safety function.

The minimum qualifications for a criticality safety engineer are a BS or BA degree in science or engineering with at least one year of nuclear industry experience in criticality safety. A criticality safety engineer has the authority and responsibility to conduct activities assigned to the criticality safety function, with the exception of independent verification of NCSEs.

The MFFF implements the administrative practices for criticality safety, as contained in Section 4.1 of American National Standards Institute/American Nuclear Society (ANSI/ANS)-8.1-1983 (R1998), Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.

The DCS criticality safety staff meet or exceed the minimum qualifications described above.

Action:

Section 6.1 will be revised to indicate the qualifications and duties described above in the next update of the CAR.

69. <u>Section 6.1, pp. 6-1 and 6-2</u>

Revise the application (including pages 6-2 and 6-4) to provide the correct reference to the American National Standards Institute/American Nuclear Society (ANSI/ANS)-8.1 standard.

Page 6-2 refers to ANSI/ANS-8.1-1983, when the correct reference should be ANSI/ANS-8.1-1983 (R1988).

Response:

The references in the CAR Chapter 6 will be revised to the latest revision of the ANSI/ANS 8.1 standard (R1998).

Action:

The ANSI/ANS-8.1 date of applicability will be corrected in the next update of the CAR.

70. <u>Section 6.1, pp. 6-1 and 6-2</u>

Justify the absence of a commitment to ANSI/ANS-8.19-1986, "Administrative Practices for Nuclear Criticality Safety", when describing the commitment to administrative practices in ANSI/ANS-8.1-1983 (R1988). Clarify what is being committed to with regard to administrative practices.

10 CFR 70.22(f) states, in part: "Each application for a license to possess and use special nuclear material in a plutonium processing and fuel fabrication plant shall contain...a description and safety assessment of the design bases of the principal structures, systems, and components of the plant..." 10 CFR 70.62(a) states that each licensee or applicant shall establish and maintain a safety program that demonstrates compliance with the performance requirements of 10 CFR 70.61. Nuclear criticality safety is an important area for the safety assessment of the design bases of the principal structures, systems, and components and for the safety program that demonstrates compliance with the 10 CFR 70.61 performance requirements. SRP Section 6.4.3.1.b recommends that ....

It appears that the applicant has explicitly chosen to commit to some standard administrative practices and implicitly commits to others. Clearly describing the criteria used in standards during the design is necessary to ensure that the design bases will provide reasonable assurance of protection against a criticality accident.

Response:

A commitment to ANSI/ANS-8.19-1986, "Administrative Practices for Nuclear Criticality Safety", is in Section 6.4 of the CAR. ANSI/ANS 8.19 was not specifically listed in the referenced pages.

Action:

A statement that DCS implements the administrative practices for criticality safety, as contained in ANSI/ANS-8.19-1996, will be added to Section 6.1 in the next update of the CAR.

71. Section 6.3.1, pp. 6-4 and 6-5

State whether the Nuclear Criticality Safety Evaluations (NCSEs) will be completed and submitted to the NRC prior to construction. Justify your response.

Page 6-4 of the application states that, "The NCSEs are used to develop the design basis of the facility and to demonstrate compliance with the double contingency principle." Information used to develop the design basis should be submitted for review.

Response:

The current schedule is for the Nuclear Criticality Safety Evaluations (NCSEs) to be prepared by the time of the submittal of the license application for possession and use of special nuclear material.

NCSEs will be summarized in the ISA Summary submitted with the license application. DCS understands that the NCSEs are part of the detailed Integrated Safety Analysis (ISA) documentation, which, in accordance with 10 CFR 70.66(c)(2), are available to NRC at the licensee's site.

Action:

None

72. Section 6.3.1, pp. 6-4 and 6-5

On page 6-5, third bullet, define exactly what is meant by the statement, "Where practicable [...]," when referring to the preferred hierarchy of controls. State whether there is a specified procedure for making the determination of practicability.

10 CFR 70.22(f) states, in part: "Each application for a license to possess and use special nuclear material in a plutonium processing and fuel fabrication plant shall contain...a description and safety assessment of the design bases of the principal structures, systems, and components of the plant..." 10 CFR 70.62(a) states that each licensee or applicant shall establish and maintain a safety program that demonstrates compliance with the performance requirements of 10 CFR 70.61. Nuclear criticality safety is an important area for the safety assessment of the design bases of the principal structures, systems, and components and for the safety program that demonstrates compliance with the 10 CFR 70.61performance requirements. SRP Section 6.4.3.3.2.0 recommends that passive geometry control is preferred and justification should be provided for other controls.

Section 6.3.3 describes each type of control and its relative preference for use as a criticality control. The term "practicable" is rather loose in this context. This information is necessary to ensure that the design bases will provide reasonable assurance of protection against a criticality accident.

Response:

"Where practicable" is meant to describe a determination based on engineering design and throughput considerations. See the response to Question 67 for information regarding how the determination of practicability is made.

The preference for passive engineered control is consistent with the design criteria stated in CAR Section 6.3.1. Passive engineered control inherently incorporates passive, designed-in forms of control, which generally result in higher levels of control reliability than other forms of control such as active engineered, enhanced administrative or simple administrative controls. However, several instances exist in the MFFF design where active engineered or administrative control is implemented because the physical processes do not lend themselves to passive control. For example, mass control is always performed by non-passive control. Mass control is often used in the powder areas where the geometrical shape of the powder can not be guaranteed. Similarly, in the laboratory where in actuality the mass is small and located in various locations, the mass is manually weighed by administrative controls where dual independent means are used. In this case, two different scales are used by two different personnel. Another example is the administrative prohibition of not bringing hydrogenous material into the normally dry areas.

Further clarification defining favored criticality safety design approach commitments will be added to the end of the third bullet in CAR Section 6.3.1, by indicating that specific safety principles incorporated during the development of the MFFF design in order to enhance the inherent reliability of criticality controls are summarized as follows: (a) the preferred use of passive engineered features over active engineered features, (b) the preferred use of engineered features over administrative controls, and (c) the preferred use of enhanced administrative controls over simple administrative controls. (See also the response to Question 80.)

Action:

The additional information discussed above will be included in the next update of the CAR.

73. <u>Section 6.3.1, pp. 6-4 and 6-5</u>

On page 6-5, fourth bullet, specify any additional facility management measures that may be used to flow down controlled parameters.

10 CFR 70.22(f) states, in part: "Each application for a license to possess and use special nuclear material in a plutonium processing and fuel fabrication plant shall contain...a description and safety assessment of the design bases of the principal structures, systems, and components of the plant..." 10 CFR 70.62(a) states that each licensee or applicant shall establish and maintain a safety program that demonstrates compliance with the performance requirements of 10 CFR 70.61. Nuclear criticality safety is an important area for the safety assessment of the design bases of the principal structures, systems, and components and for the safety program that demonstrates compliance with the 10 CFR 70.61performance requirements. SRP Section 6.4.3.3.2.D recommends that the applicant [commit to] incorporate controls into facility management measures as required by 10 CFR 70.62(d).

If there are additional management measures that are relied on for safety, they should be identified here. This information is necessary to ensure that the design bases will provide reasonable assurance of protection against a criticality accident.

Response:

Section 6.2 and Chapter 15 address management measures that will be relied on for safety. Sections 6.2.1 through 6.2.4 address specific management measures, which are related to criticality safety. Any additional criticality related management measures identified during the ISA will be documented in the LA. The 4th bullet under Section 6.3.1 will be amended to indicate that the controlled parameters are flowed into facility management measures, such as procedures and maintenance requirements, as specified in Section 6.2.

Action:

Section 6.3.1 will be clarified to reflect the information above in the next update of the CAR.

74. Section 6.3.2, pp. 6-6 thru 6-8

Provide a list of those specific areas and/or operations for which an exemption is sought, with justification.

The basis for criticality accident alarm system (CAAS) exemptions is stated as follows: "CAAS coverage will be exempted from areas that are (1) limited to less than half of a minimum critical mass with no potential for double batching, and (2) used for storage of closed shipping containers." Under 10 CFR 70.24, an exemption must be approved by NRC. The CAAS is one of the principal SSCs of the facility mitigating the consequences of a criticality accident.

Response:

Specific areas (if any) requiring exemption from criticality accident monitoring requirements will be identified in the LA. The basis for such exemptions will be provided.

Action:

Section 6.3.2 will be clarified to reflect the information above in the next update of the CAR.

75. Section 6.3.2, pp. 6-6 thru 6-8

Describe whether CAAS detectors will be gamma or neutron detectors or whether they will provide dual alarm coverage of all non-exempt areas.

Dual alarm coverage is required under 10 CFR 70.24. This information does not appear to be provided in the application. The CAAS is one of the principal SSCs of the facility mitigating the consequences of a criticality accident.

Response:

As required by 10 CFR 70 and Regulatory Guide 3.71, the MFFF CAAS will provide two criticality monitors/alarms. Standard gamma/neutron criticality detectors are planned to be used. Actual detector selection will be made for final design.

Action:

None

76. Section 6.3.3.1, pp. 6-8 and 6-9

Clarify under what set of conditions in Section 6.3.3.1 neutron interaction is to be considered.

This section could be interpreted to mean that neutron interaction is only considered where single parameter limits are applied. This information is necessary to ensure that the process remains subcritical under both normal and credible abnormal conditions in accordance with 10 CFR 70.61(d).

Response:

Neutron interaction is considered for all conditions to ensure that the process remains subcritical under all normal and credible accident conditions. Section 6.3.3.1 will be modified to indicate that the potential for neutron interaction between units is fully evaluated to ensure that the process remains subcritical under all normal and credible accident conditions, and additional controls on spacing are identified and incorporated into facility management measures as necessary.

Action:

Section 6.3.3.1 will be modified as identified above in the next update of the CAR.

77. Section 6.3.3.1, pp. 6-8 and 6-9

Explain the special status afforded to fixed neutron absorbers in Section 6.3.3.1 and state whether the use of other types of neutron absorbers are considered.

This section 6.3.3.1 singles out fixed neutron absorbers specifically in the section on general criticality control modes. This information is necessary to ensure that the design bases will provide reasonable assurance of protection against a criticality accident as required by 10 CFR 70.61(d).

Response:

Fixed neutron absorbers are afforded special status in the third paragraph of CAR page 6-9 since fixed neutron absorbers represent a very reliable means of criticality control (i.e., passive control method) often used in conjunction with geometry control. When incorporated into the MFFF design along with a commitment to implement ANSI/ANS-8.21-1995 guidance (see response to Question 90), criticality control provided with fixed neutron absorbers is considered comparable to geometry control. This position is consistent with the CAR Section 6.3.3.1 commitment to rely on passive engineered controls where practicable.

Other types of neutron absorber control (i.e., soluble boron) are not considered as inherently reliable as fixed neutron absorbers and are not currently envisioned to be employed in any MFFF process unit or area. Any such applications of neutron absorber control in the final MFFF design will be clearly identified in the LA and ISA. In any case, compliance with the double contingency principle of ANSI/ANS-8.1-1983 (R1998) will be demonstrated in the NCSEs.

Action:

None

78. <u>Section 6.3.3.2, pp. 6-10 and 6-18</u>

In Section 6.3.3.2, "Available Method of Control", for all controlled parameters (especially mass, volume, and geometry), commit to consider the most reactive combinations of tolerances on the dimensions and material specifications.

SRP Section 6.4.3.3.2.9.A allows for the use of borosilicate glass raschig rings provided the applicant commits to ANSI/ANS 8.5-1996; soluble absorbers could also be used, but no mention of raschig rings or soluble absorbers is made. Section 6.3.3.2.1, "Geometry Control", states that "tolerances on nominal design dimensions are treated conservatively", but does not state that the most reactive combination of tolerances will ~~be determined~~ and used. This should also be generally true for other controlled parameters. This is necessary to ensure that the process remains subcritical under both normal and credible abnormal conditions in accordance with 10 CFR 70.61(d).

Response:

As stated in Section 6.3.4.3, General Design Approach, "the design approach with respect to criticality … for each controlled parameter, assume the credible optimal condition (i.e., most reactive condition physically possible) for the parameter, or calculate the allowed range for the parameter." Criticality calculations and nuclear criticality safety evaluations are performed assuming the most reactive physical condition to ensure that the process remains subcritical under all normal and abnormal conditions, in accordance with 10 CFR 70.61(d).

Action:

None

79. Section 6.3.3.2.4, p. 6-12; Section 6.3.3.2.5, p. 6-13

Provide a demonstration for the following assumptions: (1) the assumption that the presence of ^{241}Pu can be neglected, in Section 6.3.3.2.4; and (2) the assumption that 1-inch of water can be used to conservatively represent reflection, in Section 6.3.3.2.5.

Demonstration of these assumptions, used in designing the facility, is necessary to ensure that the process remains subcritical under both normal and credible abnormal conditions in accordance with 10 CFR 70.61(d).

Response:

(1) Specifications of plutonium isotopics used in the MFFF include concentration of fissile and nonfissile plutonium isotopes (e.g., ^{239}Pu, ^{240}Pu, ^{241}Pu), as well as the relative abundance of plutonium to uranium. A calculation will be prepared that will show that 96% ^{239}Pu and 4% ^{240}Pu as used in criticality calculations bounds the presence of ^{240}Pu (5% to 9%) and ^{242}Pu (<0.02%) and offsets any contribution from ^{241}Pu (<1%) such that it can be neglected. A clarifying statement after the third sentence in CAR Section 6.3.3.2.4 will be added to indicate that these statements will be demonstrated in a MFFF criticality calculation and referenced in NCSEs.

(2) As stated in SRP Section 6.4.3.3.2.5, at a minimum, reflection conditions equivalent to a 1-inch tight fitting water jacket are assumed to account for personnel and other transient incidental reflectors not evaluated in unreflected models. However, it is MFFF criticality calculation practice that ranges of water reflection up to and including 12 inches (30 cm) be employed. The calculations as referenced by the NCSEs shall provide a demonstration of the full range of reflection. Any exceptions to this principle, such as in the moderation controlled, normally dry areas, shall be fully justified in the NCSEs.

A clarifying statement at the end of the second bullet in CAR Section 6.3.3.2.5 will be added to indicate that the quantity of water reflection will be demonstrated to be conservative in the calculations/NCSEs.

Action:

The clarifying statements indicated above will be included in the next update of the CAR.

80. Section 6.3.4, pp. 6-18 thru 6-33

Provide a commitment to the effect that two-parameter control is preferred over single-parameter control and show how this principle is applied in Tables 6-1 and 6-2.

10 CFR 70.22(f) states, in part: "Each application for a license to possess and use special nuclear material in a plutonium processing and fuel fabrication plant shall contain...a description and safety assessment of the design bases of the principal structures, systems, and components of the plant..." 10 CFR 70.62(a) states that each licensee or applicant shall establish and maintain a safety program that demonstrates compliance with the performance requirements of 10 CFR 70.61. Nuclear criticality safety is an important area for the safety assessment of the design bases of the principal structures, systems, and components and for the safety program that demonstrates compliance with the 10 CFR 70.61performance requirements. SRP Section 6.4.3.3.5.B recommends that two-parameter control should be preferred over one-parameter control. There does not appear to be a statement that reliance on controls on two independent parameters is preferable to two controls on one parameter. This should be part of the preferred design approach. Connected with this, several process units only appear to have a single controlled parameter defined. Knowledge of the design criteria to be used is necessary to ensure that the design bases will provide reasonable assurance of protection against a criticality accident.

Response:

The preference for two-parameter control over one-parameter control is consistent with the safety principles stated in CAR Section 6.3.4.2. Two-parameter control inherently incorporates diverse forms of control, which generally result in higher levels of control reliability than single-parameter control. However, such a criticality control scheme incorporating the preferred MFFF hierarchy of criticality controls is not feasible for a MFFF. Therefore, the MFFF design preference is to rely on passive geometry control as the preferred criticality safety control, followed by reliance on dual independent controls on control parameters.

The purpose of Tables 6-1 and 6-2 is to show the criticality control methods for the main criticality control units in the MFFF. In all cases of parameters indicated as used for criticality control in the MFFF ("YES" in the tables), the design of the MFFF is such that no single credible failure will result in a criticality. As shown in Tables 6-1 and 6-2, criticality control in many locations in the MFFF is by the preferred passive geometry control that is implemented by design. In other cases, such as shown in Table 6-2 in the powder area, geometry control is not practical due to the changing geometry that results from the process. That is, there exists a variety of hoppers, scales, conveyors, mixers, and locations of material containers each with a varying geometry. In those cases, both mass and moderation is each controlled such that no single failure will result in a criticality. However, it is obviously important to control both of these parameters.

Further clarification defining favored criticality safety design approach commitments will be added to the end of CAR Section 6.3.4.2. The clarification will indicate that specific safety principles incorporated during the development of the MFFF design in order to enhance the inherent reliability of criticality controls are summarized as follows: (a) the preferred use of passive engineered features over active engineered features, (b) the preferred use of engineered features over administrative controls, (c) the preferred use of enhanced administrative controls over simple administrative controls, and (d) the preferred use of two-parameter control over single parameter control.

Action:

The above clarification to Section 6.3.4.2 will be included in the next update of the CAR.

81. <u>Section 6.3.4, pp. 6-18 thru 6-30</u>

For the Aqueous Polishing Process, for those Criticality Control Units (CCUs) where there is only one control defined, state the design approach to establishing double contingency protection, including whether there will be dual independent controls on the one parameter.

10 CFR 70.22(f) states, in part: "Each application for a license to possess and use special nuclear material in a plutonium processing and fuel fabrication plant shall contain...a description and safety assessment of the design bases of the principal structures, systems, and components of the plant..." 10 CFR 70.62(a) states that each licensee or applicant shall establish and maintain a safety program that demonstrates compliance with the performance requirements of 10 CFR 70.61. Nuclear criticality safety is an important area for the safety assessment of the design bases of the principal structures, systems, and components and for the safety program that demonstrates compliance with the 10 CFR 70.61 performance requirements. SRP Section 6.4.3.3.5.A recommends that the applicant commit to the double contingency principle as stated in ANSI/ANS 8.1-1983. Knowledge of the design philosophy is necessary to ensure that the design bases will provide reasonable assurance of protection against a criticality accident.

Response:

As stated in response to Question 80, the MFFF design preference is for passive geometry control, followed by dual independent controls on controlled parameters. The purpose of Tables 6-1 and 6-2 was to show the criticality parameter control methods for the main criticality control units in the MFFF. In all cases of parameters indicated as used for criticality control in the MFFF ("YES" in the tables), the design of the MFFF is such that no single credible failure will result in a criticality.

Action:

The revised Tables 6-1 and 6-2 will be included in the next update of the CAR. (See revised tables as part of the response to Question 83.)

82. <u>Section 6.3.4, pp. 6-18 thru 6-33</u>

Explain whether the physicochemical forms discussed are controlled programmatically in the same manner as other criticality control modes. Describe why they have been separated out and how they are treated differently than other parameter limits, if any.

10 CFR 70.22(f) states, in part: "Each application for a license to possess and use special nuclear material in a plutonium processing and fuel fabrication plant shall contain...a description and safety assessment of the design bases of the principal structures, systems, and components of the plant..." 10 CFR 70.62(a) states that each licensee or applicant shall establish and maintain a safety program that demonstrates compliance with the performance requirements of 10 CFR 70.61. Nuclear criticality safety is an important area for the safety assessment of the design bases of the principal structures, systems, and components and for the safety program that demonstrates compliance with the 10 CFR 70.61performance requirements. SRP Section 6.4.3.3.2.D recommends that incorporating controls into facility management measures. Knowledge of the design philosophy to be used is necessary to ensure that the design bases will provide reasonable assurance of protection against a criticality accident. Whenever process controls are relied on for criticality safety, they should be programmatically controlled.

Response:

Physicochemical forms are controlled in the same way as other criticality control parameters. SRP Section 6.4.3.3.2 describes 12 methods of criticality control. Therefore, CAR Section 6.3.4.3.2 has been consistent with those definitions. However, the physicochemical form of the fissile material is equally important. Therefore, it has been included separately in CAR Section 6.3.4.3.1.

Additionally, the importance of controlling physicochemical characteristics has been shown by listing it in Tables 6-1 and 6-2 (as revised in the response to Question 83) along with the other 12 methods of criticality control. It is also noted in the first sentence of CAR Section 6.3.4.3.2, "Choice of Criticality Control Mode," that criticality safety in the MFFF is ensured by application of one or more of the 12 control modes listed in the SRP, as well as by the control of the physicochemical forms of the fissile material.

Action:

None

83. <u>Section 6.3.4, pp. 6-18 thru 6-30</u>

Several CCUs do not have any parameters identified. Describe the criticality safety design basis for all of these units in more detail.

10 CFR 70.22(f) states, in part: "Each application for a license to possess and use special nuclear material in a plutonium processing and fuel fabrication plant shall contain...a description and safety assessment of the design bases of the principal structures, systems, and components of the plant..." 10 CFR 70.62(a) states that each licensee or applicant shall establish and maintain a safety program that demonstrates compliance with the performance requirements of 10 CFR 70.61. Nuclear criticality safety is an important area for the safety assessment of the design bases of the principal structures, systems, and components and for the safety program that demonstrates compliance with the 10 CFR 70.61performance requirements. SRP Section 6.4.3.3.2.E recommends that the applicant describes controlled parameters for each process used as NCS controls. Knowledge of the dominant controlled parameters upon which the facility design will be based is necessary to ensure that the design bases will provide reasonable assurance of protection against a criticality accident.

Response:

CAR Tables 6-1 and 6-2 have been revised to indicate that criticality control of these units is by concentration control, via upstream units. Material is only allowed to enter these units after concentration measurements/indications are made and verified by an independent means to satisfy the double contingency principle.

Only if the concentration is determined to be low, and verified by an independent means to satisfy the double contingency principle, is material allowed to enter these units. The allowance of material to enter the units will be demonstrated to meet the double contingency principle in the NCSEs.

Action:

The attached Tables 6-1 and 6-2 will be included in the next update of the CAR.

Table 6-1. Preliminary Definition of Reference Fissile Medium and Control Methods for Principal AP Process Units

Criticality Control Unit	Control Method													Comments
	Physicochemical Characteristics (PC)	Mass (M)	Geometry (G)	Density (D)	Isotopics (I)	Reflection (R)	Moderation (MN)	Concentration (C)	Interaction (IN)	Neutron absorber (A)	Volume (V)	Heterogeneity (H)	Process variable	
Decanning Unit														
PuO_2 dosing hopper	NO $PuO_2 + H_2O$	NO	YES	NO [1] $d \leq 7$	NO [1] $^{240}Pu \geq 4\%$	NO	NO	NO	TBD [2]	YES Cd coating	NO	NO	NO	Plutonium coming from the PuO_2 container storage vault; see Table 6-2.
Dissolution Unit														
Electrolyzer	NO $PuO_2 + H_2O$	NO	YES	NO [1] $d \leq 7$	NO [1] $^{240}Pu \geq 4\%$	NO	NO	NO	TBD [2]	YES Cd coating	NO	NO	NO	
Reception tank	NO $PuO_2 + H_2O$	NO	YES slab	NO [1] $d \leq 7$	NO [1] $^{240}Pu \geq 4\%$	NO	NO	NO	TBD [2]	YES Cd coating	NO	NO	NO	
PuO_2 filter	NO $PuO_2 + H_2O$	NO	YES Cylin-der	NO [1] $d \leq 7$	NO [1] $^{240}Pu \geq 4\%$	NO	NO	NO	TBD [2]	NO	NO	NO	NO	Double control to guarantee absence of PuO_2 in downstream equipment.
Dilution and sampling tank	YES $Pu(NO_3)_3 + H_2O$[3]	NO	YES Slab	NO	NO [1] $^{240}Pu \geq 4\%$	NO	NO	NO	TBD [2]	Cole-manite concrete	NO	NO	PC	Colemanite concrete is a type of borated concrete.

Table 6-1. Preliminary Definition of Reference Fissile Medium and Control Methods for Principal AP Process Units (Continued)

Criticality Control Unit	Control Method													Comments
	Physicochemical Characteristics (PC))	Mass (M)	Geometry (G)	Density (D)	Isotopics (I)	Reflection (R)	Moderation (MN)	Concentration (C)	Interaction (IN)	Neutron absorber (A)	Volume (V)	Heterogeneity (H)	Process variable	
Dissolution Unit (Continued)														
Buffer Tank	YES $Pu(NO_3)_3$ + H_2O [3,6]	NO	YES Annul-ar	NO	NO [1] $^{240}Pu \geq$ 4%	NO	NO	NO	TBD [2]	YES Cole-manite concrete	NO	NO	NO	Colemanite concrete is a type of borated concrete.
Purification Unit														
Feeding Tank	YES $Pu(NO_3)_3$ + H_2O [3,6]	NO	YES Annul-ar	NO	NO [1] $^{240}Pu \geq$ 4%	NO	NO	NO	TBD [2]	YES Cole-manite concrete	NO	NO	NO	Colemanite concrete is a type of borated concrete.
Purification pulsed columns: Extraction Scrubbing Pu stripping	YES $Pu(NO_3)_3$ + H_2O [3,6]	NO	YES Cylin-der	NO	NO [1] $^{240}Pu \geq$ 4%	NO	NO	NO	TBD [2]	NO	NO	NO	NO	
Diluent washing pulsed columns	YES $Pu(NO_3)_3$ + H_2O [3,6]	NO	YES Cylin-der	NO	NO [1] $^{240}Pu \geq$ 4%	NO	NO	NO	TBD [2]	YES Cd coating	NO	NO	NO	
Pu barrier mixer settlers	YES $Pu(NO)_3$ + H_2O [3,6]	NO	YES	NO	NO [1] $^{240}Pu \geq$ 4%	NO	NO	NO	TBD [2]	YES Cd coating	NO	NO	NO	

Table 6-1. Preliminary Definition of Reference Fissile Medium and Control Methods for Principal AP Process Units (Continued)

Criticality Control Unit	Control Method													Comments
	Physicochemical Characteristics (PC))	Mass (M)	Geometry (G)	Density (D)	Isotopics (I)	Reflection (R)	Moderation (MN)	Concentration (C)	Interaction (IN)	Neutron absorber (A)	Volume (V)	Heterogeneity (H)	Process variable	
Purification Unit (Continued)														
U stripping + diluent washing mixer settlers	YES $UO_2(NO_3)_2 + H_2O$ [6]	NO	YES slab	NO	NO $^{235}U \leq 93.5\%$	NO	NO	YES	TBD [2]	NO	NO	NO	C	Nominal $UO_2(NO_3)_2$ concentration well below value used for geometry calculation.
Oxidation columns	YES $Pu(NO_3)_3 + H_2O$ [3,6]	NO	YES Cylin-der	NO	NO [1] $^{240}Pu \geq 4\%$	NO	NO	NO	TBD [2]	NO	NO	NO	NO	
Pu Rework Tank	NO $Pu(NO_3)_3 + H_2O$ [3]	NO	YES Slab	NO	NO [1] $^{240}Pu \geq 4\%$	NO	NO	NO	TBD [2]	YES Cd coating	NO	NO	NO	
Rafinates Reception, and Recycling, Control Tanks	NO $Pu(NO_3)_3 + H_2O$ [3]	NO	YES Annul-ar	NO	NO [1] $^{240}Pu \geq 4\%$	NO	NO	NO	TBD [2]	YES Cole-manite concrete	NO	NO	NO	Colemanite concrete is a type of borated concrete.

Table 6-1. Preliminary Definition of Reference Fissile Medium and Control Methods for Principal AP Process Units (Continued)

Criticality Control Unit	Control Method													Comments
	Physicochemical Characteristics (PC)	Mass (M)	Geometry (G)	Density (D)	Isotopics (I)	Reflection (R)	Moderation (MN)	Concentration (C)	Interaction (IN)	Neutron absorber (A)	Volume (V)	Heterogeneity (H)	Process variable	
Oxalic Precipitation and Oxidation Unit														
Reception tank	YES $Pu(NO_3)_3 + H_2O$ [3,6]	NO	YES Annul-ar	NO	NO [1] $^{240}Pu \geq 4\%$	NO	NO	NO	TBD [2]	YES Cole-manite concrete	NO	NO	NO	Colemanite concrete is a type of borated concrete.
Preparation tank	YES $Pu(NO_3)_3 + H_2O$ [3,6]	NO	YES Annul-ar	NO	NO [1] $^{240}Pu \geq 4\%$	NO	NO	NO	TBD [2]	NO	NO	NO	NO	
Precipitators	YES $PuO_2F_2 + H_2O$ [4,6]	NO	YES	NO	NO [1] $^{240}Pu \geq 4\%$	NO	NO	NO	TBD [2]	NO	NO	NO	NO	
Flat filter	YES $PuO_2F_2 + H_2O$ [4,6]	NO	YES Slab	NO	NO [1] $^{240}Pu \geq 4\%$	NO	NO	NO	TBD [2]	NO	NO	NO	NO	
Calcination furnace	NO $PuO_2 + H_2O$	NO	YES Cylin-der	NO [1] $d \leq 3.5$	NO [1] $^{240}Pu \geq 4\%$	NO	NO	NO	TBD [2]	NO	NO	NO	NO	
Homogenization Unit														
Homogenizing hoppers	NO $PuO_2 + H_2O$	NO	YES Slab	NO [1] $d \leq 3.5$	NO [1] $^{240}Pu \geq 4\%$	NO	NO	NO	NO [5]	YES Cd coating	NO	NO	NO	

Table 6-1. Preliminary Definition of Reference Fissile Medium and Control Methods for Principal AP Process Units (Continued)

Criticality Control Unit	Control Method													Comments
	Physicochemical Characteristics (PC)	Mass (M)	Geometry (G)	Density (D)	Isotopics (I)	Reflection (R)	Moderation (MN)	Concentration (C)	Interaction (IN)	Neutron absorber (A)	Volume (V)	Heterogeneity (H)	Process variable	
Canning Unit														
Canning feeding head	NO PuO_2 + H_2O	NO	YES	NO [1] d ≤ 3.5	NO [1] $^{240}Pu \geq$ 4%	NO	NO	NO	TBD [2]	NO	NO	NO	NO	
Oxalic Mother Liquor Recovery Unit														
Oxalic mother liquors recovery	YES PuO_2F_2 + H_2O [4,6]	NO	YES	NO	NO [1] $^{240}Pu \geq$ 4%	NO	NO	NO	TBD [2]	YES Cole-manite concrete	NO	NO	NO	Colemanite concrete is a type of borated concrete.
Solvent Recovery Unit														
Solvent recovery mixer settlers	YES $UO_2(NO_3)_2$ + H_2O [6]	NO	NO	NO	NO $^{235}U \leq$ 93.5%	NO	NO	YES [7]	TBD [2]	NO	NO	NO	NO	
Acid Recovery Unit														
Acid recovery	YES $Pu(NO_3)_3$ + H_2O [3,6]	NO	NO	NO	NO [1] $^{240}Pu \geq$ 4%	NO	NO	YES [7]	TBD [2]	NO	NO	NO	NO	
Silver Recovery Unit														
Silver recovery	YES $Pu(NO_3)_3$ + H_2O [3,6]	NO	NO	NO	NO [1] $^{240}Pu \geq$ 4%	NO	NO	YES [7]	TBD [2]	NO	NO	NO	NO	

Table 6-1. Preliminary Definition of Reference Fissile Medium and Control Methods for Principal AP Process Units (Continued)

Criticality Control Unit	Control Method													Comments
	Physicochemical Characteristics (PC)	Mass (M)	Geometry (G)	Density (D)	Isotopics (I)	Reflection (R)	Moderation (MN)	Concentration (C)	Interaction (IN)	Neutron absorber (A)	Volume (V)	Heterogeneity (H)	Process variable	
Offgas Treatment Unit														
Offgas Treatment	NO	NO	NO	NO	NO	NO	NO	YES [7]	TBD [2]	NO	NO	NO	NO	
Liquid Waste Reception Unit														
Liquid Waste Reception	NO	NO	NO	NO	NO	NO	NO	YES [7]	TBD [2]	NO	NO	NO	NO	
Sampling Unit														
Sampling Unit	NO	NO	NO	NO	NO	NO	NO	YES [7]	TBD [2]	NO	NO	NO	NO	

NOTES:

[1] Parameter value ranges indicated are selected for use in criticality design calculations to encompass credible optimum conditions without reliance on process variable controls.
[2] To be determined (TBD). Analysis of interaction between components to be evaluated to confirm spacing requirements, or determine if additional criticality control design features or management measures are required to address interaction.
[3] Actual chemical form of Pu Nitrate is $Pu(NO_3)_4$ for most process steps, which is less reactive than $Pu(NO_3)_3$.
[4] Actual chemical form is a mixture of Pu Oxalate and Pu Nitrate. Either chemical form is less reactive than PuO_2F_2.
[5] Interaction limited by geometry (hopper spacing) and cadmium coating of hoppers.
[6] The absence of a more restrictive material is controlled in an upstream unit, which prevents any means of adverse chemical form change.
[7] Concentration controlled by upstream units.

Table 6-2. Preliminary Definition of Reference Fissile Medium and Control Methods for MP Process Units

Criticality Control Unit	Control Method													Comments
	Physicochemical Characteristics (PC)	Mass (M)	Geometry (G)	Density (D)	Isotopics (I)	Reflection (R)	Moderation (MN)	Concentration (C)	Interaction (IN)	Neutron absorber (A)	Volume (V)	Heterogeneity (H)	Process variable	
Receiving Area														
PuO_2 3013 storage pit	NO $PuO_2 + H_2O$	NO $M \leq 5$ kg / container	YES	NO [1] $d \leq 7$	NO [1] $^{240}Pu \geq 4\%$	NO [2]	NO [1] $H_2O \leq 1\%$ inside containers	NO	NO [2]	NO	NO	NO	NO	-Incoming plutonium container I.D. is verified to confirm mass, isotopics, and powder moisture assumptions listed.
PuO_2 Container Opening and Handling Unit	NO $PuO_2 + H_2O$	YES	NO	NO [1] $d \leq 7$	NO [1] $^{240}Pu \geq 4\%$	NO	YES	NO	TBD [3]	NO	NO	NO	NO	
PuO_2 buffer storage	NO $PuO_2 + H_2O$	NO	YES	NO [1] $d \leq 3.5$	NO [1] $^{240}Pu \geq 4\%$	NO [2]	NO	NO	NO [2]	YES Borated concrete	NO	NO	NO	
Primary dosing (including master blend homogenizing)	NO $PuO_2 + H_2O$,	YES	NO	NO [1] $d \leq 3.5$	YES $^{240}Pu \geq 4\%$[1]	NO	YES [4]	NO	TBD [3]	NO	NO	TBD	M,I, MN [4]	-Mass of PuO_2 per jar is controlled. -The relative quantity of PuO_2 and UO_2 is controlled; used in downstream units. -Homogeneity of master blend will be controlled if required by downstream units.

Table 6-2. Preliminary Definition of Reference Fissile Medium and Control Methods for MP Process Units (Continued)

Criticality Control Unit	Control Method													Comments
	Physicochemical Characteristics (PC)	Mass (M)	Geometry (G)	Density (D)	Isotopics (I)	Reflection (R)	Moderation (MN)	Concentration (C)	Interaction (IN)	Neutron absorber (A)	Volume (V)	Heterogeneity (H)	Process variable	
Powder Area														
Primary blend ball milling Scrap milling	NO Master blend	YES	NO	NO [1] $d \leq 5.5$	YES $^{240}Pu \geq 4\%$[1]; $M_{Pu}/(M_U + M_{Pu}) \leq 22\%$ [5]	NO	YES	NO	TBD [3]	NO	NO	NO	M	-U metal balls are present in the ball-mill and are accounted for as reflector in the criticality calculations
Final dosing	NO Master blend	YES	NO	NO [1] $d \leq 5.5$	YES $^{240}Pu \geq 4\%$[1]; $M_{Pu}/(M_U + M_{Pu}) \leq 22\%$ [5] $M_{Pu}/(M_U + M_{Pu}) \leq 6.3\%$ in jar	NO	YES	NO	TBD [3]	NO	NO	NO	M,I	The relative quantity of master blend and UO_2 is controlled; used in downstream units.

Table 6-2. Preliminary Definition of Reference Fissile Medium and Control Methods for MP Process Units (Continued)

Criticality Control Unit	Control Method													Comments
	Physicochemical Characteristics (PC)	Mass (M)	Geometry (G)	Density (D)	Isotopics (I)	Reflection (R)	Moderation (MN)	Concentration (C)	Interaction (IN)	Neutron absorber (A)	Volume (V)	Heterogeneity (H)	Process variable	
Powder Area (Continued)														
Homogenizing and pelletizing	NO Master blend	YES	NO	NO [1] d ≤ 5.5	YES $^{240}Pu \geq$ 4%[1]; %Pu ≤ 22% [5,7]	NO	YES	NO	TBD [3]	NO	NO	NO	M	-Final blend heterogeneity process variable control used in downstream units. -Physicochemical characteristics control applied to control pellet dimensions to extent used in downstream units.
	NO Final blend	YES	NO	NO [1] d ≤ 3.5	YES $^{240}Pu \geq$ 4%[1]; %Pu ≤ 6.3% [5]	NO	YES [4]	NO	TBD [3]	NO	NO	YES	M,MN [4], H	
	NO Pellets	YES	NO	NO	YES $^{240}Pu \geq$ 4%[1]; %Pu ≤ 6.3% [5]	NO	YES	NO	TBD [3]	NO	NO	NO [8]	M	

Table 6-2. Preliminary Definition of Reference Fissile Medium and Control Methods for MP Process Units (Continued)

Criticality Control Unit	Control Method													Comments
	Physicochemical Characteristics (PC)	Mass (M)	Geometry (G)	Density (D)	Isotopics (I)	Reflection (R)	Moderation (MN)	Concentration (C)	Interaction (IN)	Neutron absorber (A)	Volume (V)	Heterogeneity (H)	Process variable	
Powder Area (Continued)														
Jar storage and handling tunnel	NO Arrays of J60 and J80 Jars	YES [13] J60 master blend ≤ 65 kg; J60 $PuO_2 \leq$ 15 kg, J80 total ≤ 90 kg, J80 Master blend ≤ 30 kg	YES	NO [1] $PuO_2 \leq$ 3.5; $UO_2 \leq$ 3.5; Master blend ≤ 5.5; Scraps ≤ 11;	YES $^{240}Pu \geq$ 4%[1]; J60 %Pu ≤ 22% [5]; J80 Master blend %Pu ≤ 22% [5]; J80 Scraps %Pu ≤ 6.3% [5]	NO [2]	YES [14] %H2O <= 5% in the jars	NO	NO [2]	NO	NO	YES [5,7]	NO	

Table 6-2. Preliminary Definition of Reference Fissile Medium and Control Methods for MP Process Units (Continued)

Criticality Control Unit	Control Method													Comments
	Physicochemical Characteristics (PC)	Mass (M)	Geometry (G)	Density (D)	Isotopics (I)	Reflection (R)	Moderation (MN)	Concentration (C)	Interaction (IN)	Neutron absorber (A)	Volume (V)	Heterogeneity (H)	Process variable	
Powder Area (Continued)														
Scrap Processing Unit	NO Scrap pellets and powder	YES	NO	NO $d \leq 11$	YES $^{240}Pu \geq 4\%$[1]; $\%Pu \leq 22\%$ [5]	NO [2]	YES [14]	NO	NO [2]	NO	NO	YES [5,7]	NO	
Powder Auxiliary Unit	NO Any MOX Powder	YES	NO	NO $d \leq 11$	YES $^{240}Pu \geq 4\%$[1]; $\%Pu \leq 22\%$ [5]	NO [2]	YES [14]	NO	NO [2]	NO	NO	YES [5,7]	NO	

Table 6-2. Preliminary Definition of Reference Fissile Medium and Control Methods for MP Process Units (Continued)

Criticality Control Unit	Control Method: Physicochemical Characteristics (PC)	Mass (M)	Geometry (G)	Density (D)	Isotopics (I)	Reflection (R)	Moderation (MN)	Concentration (C)	Interaction (IN)	Neutron absorber (A)	Volume (V)	Heterogeneity (H)	Process variable	Comments
Pellet Process Area														
Pellet storage	YES Array of pellets [9]	NO	YES	NO $d \leq 11$	YES $^{240}Pu \geq$ 4%[1]; %Pu $\leq$ 6.3% [5]	NO [2]	NO	NO	NO [2]	YES	NO	YES [8]	NO	-Isolation shields provided for interaction control between boats.
Sintering furnace	YES Array of pellets [9]	NO [10]	YES	NO $d \leq 11$	YES $^{240}Pu \geq$ 4%[1]; %Pu $\leq$ 6.3% [5]	NO [2]	NO [10]	NO	NO [2]	NO	NO	YES [8]	NO	
Grinding	YES Array of pellets [9]	YES	NO	NO $d \leq 11$	YES $^{240}Pu \geq$ 4%[1]; %Pu $\leq$ 6.3% [5]	NO	YES	NO	TBD [3]	NO	NO	YES [8]	M	

Table 6-2. Preliminary Definition of Reference Fissile Medium and Control Methods for MP Process Units (Continued)

Criticality Control Unit	Control Method													Comments
	Physicochemical Characteristics (PC)	Mass (M)	Geometry (G)	Density (D)	Isotopics (I)	Reflection (R)	Moderation (MN)	Concentration (C)	Interaction (IN)	Neutron absorber (A)	Volume (V)	Heterogeneity (H)	Process variable	
Pellet Process Area (Continued)														
Pellet inspection and sorting, Quality Control and Manual Sorting	YES Array of pellets [9]	NO [10]	YES	NO $d \leq 11$	YES $^{240}Pu \geq 4\%$[1]; $\%Pu \leq 6.3\%$ [5]	NO	NO [10]	NO	NO [2]	NO	NO	YES [8]	NO	-Physicochemical characteristics control applied to verify pellet dimensions.
Pellet tray-baskets storage	YES Array of pellets [9]	NO	YES	NO $d \leq 11$	YES $^{240}Pu \geq 4\%$[1]; $\%Pu \leq 6.3\%$ [5]	NO [2]	NO	NO	NO [2]	YES	NO	YES [8]	NO	-Interaction between storage units controlled by isolation shields.
Scrap pellet storage	YES Final blend pellet scraps [9]	NO	YES	NO $d \leq 11$	YES $^{240}Pu \geq 4\%$[1]; $\%Pu \leq 6.3\%$ [5]	NO [2]	NO	NO	NO [2]	YES	NO	YES [8]	NO	-Interaction between storage units controlled by isolation shields.
Scrap box loading, Pellet Repackaging, Pellet Handling	YES Array of pellets [9]	YES	NO	NO $d \leq 11$	YES $^{240}Pu \geq 4\%$[1]; $\%Pu \leq 6.3\%$ [5]	NO [2]	NO	NO	NO [2]	NO	NO	YES [8]	NO	

Table 6-2. Preliminary Definition of Reference Fissile Medium and Control Methods for MP Process Units (Continued)

Criticality Control Unit	Control Method: Physicochemical Characteristics (PC)	Mass (M)	Geometry (G)	Density (D)	Isotopics (I)	Reflection (R)	Moderation (MN)	Concentration (C)	Interaction (IN)	Neutron absorber (A)	Volume (V)	Heterogeneity (H)	Process variable	Comments
Fuel Rod Process Area														
Rod cladding and decontamination	YES Array of pellets/rods [11]	NO [10]	YES	NO $d \leq 11$	YES $^{240}Pu \geq 4\%$[1]; $\%Pu \leq 6.3\%$ [5]	NO [2]	NO [10]	NO	TBD [3]	NO	NO	YES [8]	NO	
Rod controls (decontamination, helium leak testing, x-ray inspection, rod scanning, rod inspection, and sorting units, decladding, dry cleaning) Rod Tray Loading	YES Array of rods [11]	NO [10]	YES	NO $d \leq 11$	YES $^{240}Pu \geq 4\%$[1]; $\%Pu \leq 6.3\%$ [5]	NO [2]	NO [10]	NO	TBD [3]	NO	NO	YES [8]	NO	
Rod storage	YES Array of rods [11]	NO	YES	NO $d \leq 11$	YES $^{240}Pu \geq 4\%$[1]; $\%Pu \leq 6.3\%$ [5]	NO	NO	NO	NO [2]	YES	NO	YES [8]	NO	-Interaction between storage units controlled by isolation shields.

Table 6-2. Preliminary Definition of Reference Fissile Medium and Control Methods for MP Process Units (Continued)

Criticality Control Unit	Control Method													Comments
	Physicochemical Characteristics (PC)	Mass (M)	Geometry (G)	Density (D)	Isotopics (I)	Reflection (R)	Moderation (MN)	Concentration (C)	Interaction (IN)	Neutron absorber (A)	Volume (V)	Heterogeneity (H)	Process variable	
Assembly Area														
Assembly mock-up loading	YES Array of rods [11]	NO	YES	NO $d \leq 11$	YES $^{240}Pu \geq$ 4%[1]; %Pu $\leq$ 6.3% [5]	NO [2]	YES	NO	TBD [3]	NO	NO	YES [8]	NO	
Assembly mounting	YES Array of rods [11]	NO [10]	YES	NO $d \leq 11$	YES $^{240}Pu \geq$ 4%[1]; %Pu $\leq$ 6.3% [5]	NO [2]	TBD	NO	TBD [3]	NO	NO	YES [8]	PC	-Completeness of assembly controlled for use in downstream process unit

Table 6-2. Preliminary Definition of Reference Fissile Medium and Control Methods for MP Process Units (Continued)

Criticality Control Unit	Control Method													Comments
	Physicochemical Characteristics (PC)	Mass (M)	Geometry (G)	Density (D)	Isotopics (I)	Reflection (R)	Moderation (MN)	Concentration (C)	Interaction (IN)	Neutron absorber (A)	Volume (V)	Heterogeneity (H)	Process variable	
Assembly Area (Continued)														
Assembly handling and inspection Assembly dry cleaning	YES Assembly [12]	NO	YES	NO $d \leq 11$	YES $^{240}Pu \geq$ 4%[1]; %Pu $\leq$ 6.3% [5]	NO [2]	YES	NO	YES	NO	NO	YES [8]	NO	-Each inspection station handles only one fuel assembly at a time, and there is no interaction between stations. -All rod positions in an assembly being repaired may not be occupied; requires moderation control.
Assembly storage	YES Array of assemblies [12]	NO	YES	NO $d \leq 11$	YES $^{240}Pu \geq$ 4%[1]; %Pu $\leq$ 6.3% [5]	NO [2]	TBD	NO	NO [2]	NO	NO	YES [8]	NO	-Moderation control may be required to preclude flooding of storage area with full density water (>0.9 g/cm^3) assuming full loading of maximum Pu content rods.
Assembly packaging	YES Array of assemblies [12]	NO	YES	NO $d \leq 11$	YES $^{240}Pu \geq$ 4%[1]; %Pu $\leq$ 6.3% [5]	NO [2]	TBD	NO	NO [2]	NO	NO	YES [8]	NO	-Moderation control may be required to preclude flooding of packaging area with full density water (>0.9 g/cm^3) assuming full loading of maximum Pu content rods.

Table 6-2. Preliminary Definition of Reference Fissile Medium and Control Methods for MP Process Units (Continued)

NOTES:
[1] Parameter value ranges indicated are selected for use in criticality design calculations to encompass credible optimum conditions without reliance on process variable controls.
[2] Reflection and interaction addressed by geometry control.
[3] To be determined (TBD). Analysis of interaction between components to be evaluated to confirm spacing requirements, or determine if additional criticality control design features or management measures are required to address interaction.
[4] Moderation control related to introduction of moderator (organic additives) into equipment for process reasons (see Section 6.3.2.6) (process variable control).
[5] Relative quantity of U and Pu ($M_{Pu}/(M_U + M_{Pu})$) process variable control implemented by upstream process units.
[6] To be determined (TBD). Homogeneity of master blend controlled by primary dosing/master blend homogenizing if required.
[7] Scrap isotopic composition (%Pu) and homogeneity controlled by upstream units (i.e., scraps are recycled MP process product).
[8] Isotopics (including U-Pu homogeneity) and diameter of pellets controlled by Homogenization and Pelletizing Unit.
[9] Diameter of pellets controlled by upstream process units.
[10] Mass and moderation control may be used in some off-normal situations (e.g., seismic).
[11] Pellet diameter controlled by upstream process units; clad characteristics guaranteed by supplier.
[12] Assembly characteristics, including dimensions of pellets, controlled by upstream process units or guaranteed by supplier, as applicable.
[13] Mass process variable control implemented by upstream process units.
[14] Moderation (additive addition) process variable control implemented by upstream process units.