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NUCLEAR REGULATORY COMMISSION

Title: Private Fuel Storage, LLC

Docket Number: 72-22-ISFSI; ASLBP No. 97-732-02-ISFSI

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of:	)	
	)	
PRIVATE FUEL STORAGE, LLC,	)	Docket No. 72-22
(Independent Spent Fuel	)	ASLBP No.
Storage Installation)	)	97-732-02-ISFSI
	)	

U. S. Nuclear Regulatory Commission
Sheraton Hotel, Wasatch Room
Salt Lake City, Utah 84114

On May 11, 2002 the above-entitled matter came on for hearing, pursuant to notice, before:

MICHAEL C. FARRAR, CHAIRMAN
Administrative Judge
Atomic Safety & Licensing Board Panel

DR. JERRY R. KLINE
Administrative Judge
Atomic Safety & Licensing Board Panel

DR. PETER S. LAM
Administrative Judge
Atomic Safety & Licensing Board Panel

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1 Saturday, May 11, 2002 9:00 a.m.

2

3 P R O C E E D I N G S

4

5 JUDGE FARRAR: Good morning, everyone.
6 We're here on a Saturday morning. We had Friday
7 afternoon off after the oral arguments yesterday.
8 Had the afternoon off because no witnesses were
9 available, so we'll be here on Saturday. I hope
10 this will give a chance for some members of the
11 public who might not have been able to come during
12 the workweek to show up. I know the Salt Lake
13 Tribune had run a little story on Friday mentioning
14 today's hearings, so I hope during the course of
15 the day, we'll see some members of the public here.
16 Having failed to complete Section D of
17 the Unified Contention, we'll move onto Section E,
18 and finish up Section D later on.
19 Any preliminary matters from the
20 parties? Then we have -- Mr. Gaukler, you're going
21 to do this?
22 MR. GAUKLER: Yes, I am.
23 JUDGE FARRAR: Then we have a PFS
24 witness -- well, let me swear him first.
25 Dr. Cornell, would you stand and raise your right

1 hand, please.

2 ALLIN CORNELL,

3 called as a witness, for and on behalf of the

4 Applicant, being first duly sworn, was examined and

5 testified as follows:

6 JUDGE FARRAR: Thank you.

7 MR. GAUKLER: Your Honor, PFS presents

8 Dr. Cornell on Section E of the Unified Contention.

9 Dr. Cornell is a recent professor at Stanford and

10 an independent consultant. He has extensive

11 professional expertise in earthquake engineering,

12 probabilistic engineering analysis of seismic and

13 other loads on structures and structural responses.

14 He also has been actively involved in the

15 development of structural design guidelines, codes

16 and standards, including determining the

17 appropriate level of earthquake design required to

18 achieve a desired level of safety.

19

20 DIRECT EXAMINATION

21 BY MR. GAUKLER:

22 Q. Doctor, you have in front of you your

23 testimony entitled Testimony of C. Allin Cornell

24 dated April 1, 2002?

25 A. Yes, I do.

1 Q. And this testimony comprises 53 pages,
2 plus a four-page attachment, a list of references
3 and resume?
4 A. Correct.
5 Q. Have you made any changes or corrections
6 to the testimony since it was filed?
7 A. Yes, I have given counsel several
8 typographical errors and other items.
9 Q. And those changes have been included in
10 the copy we have handed out this morning?
11 A. Yes.
12 Q. This testimony was prepared by you or
13 under your supervision and direction?
14 A. Correct.
15 Q. With these changes, do you accept your
16 testimony as true and correct?
17 A. Yes, I do.
18 Q. For this proceeding?
19 A. Yes, I do.
20 MR. GAUKLER: Your Honor, I ask that the
21 testimony of C. Allin Cornell be included in the
22 transcript as if read.
23 JUDGE FARRAR: Any objections?
24 MS. NAKAHARA: No objection, Your Honor.
25 MR. TURK: No, Your Honor.

1 JUDGE FARRAR: Then we'll have the
2 testimony bound in the transcript at this point as
3 if read.
4
5 (Prefiled testimony of Dr. Allin Cornell
6 follows.)
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April 1, 2002

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
PRIVATE FUEL STORAGE L.L.C.	)	Docket No. 72-22
	)	
(Private Fuel Storage Facility)	)	ASLBP No. 97-732-02-ISFSI

TESTIMONY OF C. ALLIN CORNELL

I. BACKGROUND – WITNESS

Q1. Please state your full name

A1. Allin Cornell.

Q2. By whom are you employed and what is your position?

A2. I am currently a professor (research) at Stanford University in Stanford, California and an independent engineering consultant. In the former capacity I perform research and supervise several Ph.D.-level graduate students in the areas of probabilistic analysis of structural engineering and earthquake engineering. As a consultant, I assist engineering and earth sciences firms, industrial concerns, and government agencies in developing and applying methodologies and standards for probabilistic seismic hazard analysis, engineering safety assessments, natural hazards analyses, and earthquake engineering.

Q3. What are your areas of professional expertise?

A3. Through my education, teaching, research and consulting activities (described below) I have developed professional expertise in earthquake engineering, probabilistic engineering analysis of seismic and other loads on structures, and structural responses to such loads. By virtue of my exper-

tise in these areas, I have been actively involved in the development of structural design guidelines, codes and standards, including the appropriate level of earthquake design required to achieve a desired level of safety. I have been involved in establishing earthquake standards of design for nuclear power plants, radiological waste facilities, offshore oil platforms, and buildings.

Q4. Please summarize your educational and professional qualifications.

A4. My professional and educational experience is summarized in the *Curriculum Vitae* attached to this testimony. My graduate education was in civil structural engineering. After nearly two decades as a faculty member at M.I.T., I entered about twenty years ago into an arrangement with Stanford University whereby I could continue conducting research and supervising advanced graduate students while devoting half-time to a professional practice as an independent consultant. A primary objective of this arrangement was to use my consulting activities to encourage and guide the rapidly emerging practice of employing probabilistic methods in engineering applications, while also being able to return to the university to study at an academic level some of the challenging technical problems identified in that practice. A focus of my efforts has been to address, through the common language of probability, the problems that arise at the interface between the scientists who characterize the natural hazards that threaten facilities and the structural and other engineers responsible for designing those facilities in a safe and cost-effective way. The majority of this work has been with earth scientists and structural engineers engaged in earthquake engineering.

Q5. Please describe your studies and professional experience in structural engineering and earthquake hazard analysis.

A5. I have been studying structural engineering since about 1956 as an undergraduate in architecture, methods of probability and statistics since graduate school, and the earth sciences through almost four decades of research and practice. My Ph.D. dissertation, which was entitled "Stochastic Process Models in Structural Engineering," included studies of earthquake engineering. I have subsequently published more than 150 papers in both

engineering and scientific journals and conference proceedings. In 1970, I co-authored the first textbook designed to educate civil engineers in probability, statistics and decision theory under uncertainty. Major recognition for my professional contributions includes election to the National Academy of Engineering in 1981, several medals of the American Society of Civil Engineering, a number of invited annual lectures (for example, that of the Earthquake Engineering Research Institute in 1999) and, most recently announced, the 2007 Medal of the Seismological Society of America. Various other accomplishments and studies relevant to this matter include the following:

- In 1968, I published a seminal paper in the Bulletin of the Seismological Society on characterizing earthquake hazards using probabilistic seismic hazard analysis ("PSHA"). Improved and elaborated by more than thirty years of subsequent application and research (by myself and by many others), PSHA has become the standard method for earth scientists to characterize and report the earthquake threat at a site. For example, the USGS has used the method for two decades to study the entire US and to produce maps of seismic hazard that appear in all model building codes.

- I have participated directly, commonly as a senior advisor, in many prominent PSHA studies. These include the PSHA for the Diablo Canyon Nuclear Power Plant ("NPP"), the major EPRI Seismic Owners Group PSHA of the Central and Eastern US ("CEUS") NPP sites, the Caltrans-sponsored PSHA studies of all major California bridges, and PSHAs for the INEEL and LLNL DOE national lab sites and the Yucca Mountain site. I was also a member of the Senior Seismic Hazard Analysis Committee (SSHAC) (sponsored jointly by NRC, EPRI and DOE) to establish "standards" for conducting PSHAs at nuclear facility sites.

- As documented in a brief history of the field in ASCE 4-98 [Ref. 32 (ASCE 4-98 Appendix A)], I was one of the originators of

seismic probabilistic risk analysis ("SPRA")[1] for nuclear power plants, beginning with informal advice to MIT colleague Norman Rasmussen who directed the first nuclear power plant PRA, WASH 1400. I was co-author with Nathan Newmark of the first published SPRA paper (presented by invitation at the annual meeting of the American Nuclear Society); this was followed by a second paper (co-authored by several structural and nuclear engineers) based on the first practical application to a specific NPP (Oyster Creek).

- I have been involved in a number of SPRA studies for nuclear facilities, including the Diablo Canyon NPP, and was a member of the NRC-sponsored Senior Seismic Margins Research Project committee responsible for directing a major project conducted by the LLNL studying the fragility curves of NPP SSCs.

- I have also served as an engineering consultant on the seismic safety assessment of major individual structures, including recently the Golden Gate Bridge, the new Pac Bell baseball park in San Francisco, the Keenleyside Dam in British Columbia, and offshore platforms in California and around the world.

Q6. Please describe your involvement in the research and development of industry codes and standards, including earthquake design standards.

A6. I have had extensive involvement in the research and development of industry codes and standards. This involvement has included activities as:

- Developer of methods to facilitate the introduction of probabilistic safety assessment directly into professional engineering codes of practice, including development of the methodology adopted by the American Institute of Steel Construction ("AISC") in the first probability-based structural code introduced in the US.

[1] SPRA couples the results of a PSHA with seismic "fragility curves" (that is, curves that depict the vulnerability of plant structures, systems, and components ("SSCs") to various levels of earthquake excitation) and a PRA model of the plant SSC interactions to produce results such as the mean annual seismically-induced core damage frequency (CDF). (The CDF is used as a subsidiary safety goal by the NRC.)

- Co-author of report for specifying loads for building design that became the basis for the American National Standards Institute ("ANSI") model building loads code.

- Member of an NRC-sponsored committee that produced the recommended guidelines for conducting the seismic margins studies of existing NPPs in the IPEEE (Individual Plant Evaluation for External Events) program.

- Member of an advisory committee to the NRC on replacement of Part 100 Appendix A with 10 C.F.R. 100.23 and Regulatory Guide 1.165, providing for probabilistic seismic standards for NPPs and setting the recommended annual probability level.

- Member of a DOE committee responsible for producing guidelines for seismic evaluation of the high-level radioactive waste tanks at DOE nuclear weapons facilities. This group worked in parallel with the DOE committee that produced DOE Standard 1020-94 for seismic evaluation of all DOE facilities. The two committees shared a key member and co-authored many concepts.

- Member of a four-person panel of senior earthquake engineers requested by the American Petroleum Institute to prepare the bases and recommendations for the selection of the mean return period of the design basis earthquake for offshore structures.

- Developer of new probability-based seismic code procedures adopted for use in the 2000 FEMA-sponsored guidelines for the design and assessment of steel-moment resisting frame buildings (a common structural system that behaved unexpectedly badly in the 1994 Northridge earthquake).

- Co-author of 2000 draft of the International Standards Organization guidelines for seismic design of offshore oil production platforms.

- Member of a National Science Foundation-sponsored, multi-university earthquake engineering research center that is studying "performance-based earthquake engineering," which will couple

PSHA, modern scientifically-based predictions of highly nonlinear dynamic building behavior, and risk-cost-benefit analysis.

Q7. What is your experience with nuclear facilities and the NRC's requirements for the design and licensing of dry cask storage systems?

A7. As indicated by the above description of my background, nuclear power plants and other nuclear facilities have been a major focus of my professional work on the development and application of methodologies and standards for evaluating earthquake hazards. My professional engagements in the area have included work for the NRC, the DOE and a number of commercial operators of nuclear power plants, defense reactors, and high level radioactive waste storage facilities. While working as a consultant to a company preparing material for ISFSI seismic rulemaking, I had the opportunity to become generally familiar with the technologies and issues applicable to the design of ISFSIs.

Q8. Are you familiar with the Private Fuel Storage Facility (PFSF) and the activities that will take place there?

A8. Yes.

Q9. What is the basis of your familiarity with the PFSF?

A9. In connection with the preparation of my earlier declaration and the preparation of this testimony, I have read relevant filings in this proceeding, reviewed portions of the Safety Analysis Report for the PFSF ("SAR") and the NRC Staff's Safety Evaluation Report ("SER"), reviewed a variety of related technical documents (such as DOE Standards 1020-94, 1020-2002, 1021-93, NUREG/CR-6728, etc., as cited herein) and have had multiple conversations with PFSF project personnel such as Mr. Bruce Ebbeson, Mr. Paul Trudeau, Dr. Robert Youngs, Dr. Alan Soler, and Dr. Krishna Singh. In addition, I attended the deposition of the State's expert witness Dr. Walter Arabasz, and have reviewed the declarations of the State's experts that were filed in support of the State's Opposition to PFS's Motion for Summary Disposition of Utah L, Part B (now Section E of Unified Contention Utah L/QQ). I have also reviewed the recent depositions of Drs. Farhang Ostadan and Steven Bartlett on Utah QQ, the earlier decla-

rations by Dr. Singh et al. of Holtec International ("Holtec") and Mr. Ebbeson of Stone & Webster, Inc. ("Stone & Webster") and the testimony being filed simultaneously by Mr. Ebbeson, Mr. Trudeau, Drs. Singh and Soler of Holtec, Dr. Robert Youngs of Geomatrix, and Dr. Wen Tseng of International Civil Engineering Consultants.

Q10. What is the purpose of your testimony?

A10. The purpose of my testimony is to respond to allegations raised by the State of Utah in Section E of Unified Contention Utah L/QQ involving the exemption requested by Private Fuel Storage ("PFS") to use the 2,000-year return period earthquake as the seismic design basis for the PFSF. In particular, I will discuss the appropriateness of using a probabilistic seismic hazard analysis as the basis for designing the PFSF and the sufficiency of the 2,000-year return period earthquake and the seismic related design procedures and criteria contained in NRC guidance documents, such as the Standard Review Plans ("SRPs") applicable to NRC-licensed facilities like the PFSF, as the standard for the PFSF seismic design. I shall also address specific issues raised by the State in Section E of the Unified Contention Utah L/QQ.

II. APPROPRIATENESS OF USING PROBABILISTIC SEISMIC HAZARD ANALYSIS METHODOLOGY FOR THE PFSF EARTHQUAKE DESIGN

Q11. Please describe how the current NRC regulations provide for the earthquake design of ISFSIs.

A11. The current regulations for the seismic design of ISFSIs at sites west of the Rocky Mountains (10 C.F.R. § 72.102(b)) call for the assessment of the design basis seismic ground motions based on the deterministic procedures formerly used for nuclear power plant design (Appendix A, 10 C.F.R. Part 100).

Q12. Please describe PFS's request for an exemption to use a 2,000-year return period earthquake as the design basis for the PFSF?

A12. PFS has requested an exemption from the deterministic methodology currently required by 10 C.F.R. Part 72 to use the Probabilistic Seismic Hazard Analysis methodology, accepted by the NRC for new nuclear power

plants, for establishing the design basis ground motions for the PFSF. Specifically, PFS proposes to set the design basis motions for the PFSF at a mean annual probability of exceedance ("MAPE") of $5x10^{-4}$. Another way of referring to these design basis motions is to say that they correspond to the 2,000-year mean return period ("MRP") level, or "the 2,000-year MRP earthquake."

Q13. What is meant by "deterministic" procedures for assessing earthquake design basis ground motions?

A13. Deterministic assessments of the seismic hazard at a site lead to one or a small set (of magnitudes and locations) of representative earthquakes that could affect a site and a corresponding set of ground motion response spectra. As it has been applied in the nuclear field, the deterministic procedure consists of associating a single event magnitude to each identified seismic source, based where possible on the dimensions of the active fault, or where such faults are ill-defined, on the historical seismicity in large regions of assumed uniform seismicity. Single locations (or distances to the site) are associated with each such event. A method of ground motion prediction is then used to project a single value of one or more ground motion measures (e.g., peak ground acceleration and/or spectral acceleration) to the site for each of the magnitude-location pairs. From these ground motion results, the dominant event pair (or set of two or three pairs) is identified together with its (or their) representative response spectra at the site. This becomes the design ground motion.

Q14. Please describe the Probabilistic Seismic Hazard Analysis (PSHA) methodology for assessing earthquake design basis ground motions and explain how does it differ from a deterministic" approach.

A14. A PSHA takes into account the entire range of potential events (magnitudes and locations) that could affect a site and resulting site ground motions (as measured by peak ground acceleration and spectral acceleration) with their corresponding frequencies of occurrence and uncertainties. The result is a curve of estimated annual probability of exceedance versus level of ground motion. This curve can be used to select the design ground mo-

tion at a level corresponding to a pre-specified mean annual probability of exceedance.

Q15. Is the PSHA methodology commonly used for determining design basis ground motions for earthquake design of building and structures?

A15. Yes. The use of PSHA methodology for establishing structural design basis ground motions is today the dominant nuclear power industry practice. Use of PSHA methodology is also prevalent in the design of other structures and facilities including buildings, bridges, offshore structures and U.S. Department of Energy ("DOE") facilities. Current regulations and guidelines based on probabilistic seismic hazard principles include those governing the design of buildings [Ref. 8 (97 Uniform Building Code ("UBC"), p. 2-17, § 1631.2) and [Ref. 9 (International Building Code ("IBC"), p. 353 § 1615.2.1], offshore structures [Ref. 10 (API RP2A, p. 125, § C.2.3.6b)], and DOE facilities [Ref. 11 (DOE-STD-1020-94, Table 2.1, pp. 2-4)].[2] In the building and offshore area, the use of PSHA-based designs dates to the early 1980s.

Q16. Why is the PSHA methodology so widely used and accepted?

A16. The PSHA methodology has become widely accepted and used today because there are several advantages to using a probabilistic approach to establish design ground motions. These advantages are: (1) the probabilistic approach captures more fully the current scientific understanding of earthquake forecasting than the deterministic method; (2) the probabilistic approach is capable of reflecting the uncertainties in professional knowledge of key elements of the seismic hazard; and (3) the probabilistic approach can be used to set design criteria that are consistent among different regions and among different failure consequences, thus allowing a rational and a equitable allocation of safety resources.

[2] Portions of DOE-STD-1020-94 are attached as PFS Exhibit DDD.

Q17. Has the NRC adopted the use of PSHA methodology?

A17. Yes. The NRC has recognized the advantages of the probabilistic approach and has replaced Appendix A, 10 C.F.R. Part 100, which was based on a deterministic hazard assessment methodology, with regulations and guidance documents that provide for use of PSHA methodology for the seismic design of new nuclear power plants. [Ref. 3 (10 C.F.R. §100.23) and Ref. 4 (Regulatory Guide 1.165)]. As stated in my background, I served on a committee of consultants that advised the NRC and its contractor in its development of these documents. The NRC has also used probabilistic seismic procedures in areas such as re-evaluation of existing nuclear power plants and norms for high-level waste geological repository design. This move towards probabilistic methodologies is consistent with the NRC's general policy of risk-informed regulations and decision making. [e.g., Ref. 5 (Reg. Guide 1.174 on Risk Informed Decisions) and Ref. 6 (Commission Direction Setting Issue 12, "Risk-Informed, Performance-Based Regulation")]. In accordance with this use of probabilistic procedures, the Commission is considering a proposed rulemaking to modify the current provisions of 10 C.F.R. § 72.102 to employ probabilistic procedures for the seismic design of ISFSIs [Ref. 7 (SECY-01-0178)].

Q18. Is it appropriate to use the PSHA methodology for assessing and determining the design basis ground motion for the PFSF as requested by PFS?

A18. Yes. The proposed use by PFS of a PSHA both to characterize the seismic hazard at the site and to set the seismic design basis of the PFSF is fully consistent with both current NRC policy and practices as well as broader engineering policy and practice. The State's seismic expert witness in this proceeding agrees that a PSHA should be used for the seismic analyses and design of the PFSF. Deposition of Walter J. Arabasz ("Arabasz Dep.") (October 31, 2001) at 44-45, attached as PFS Exhibit EEE.

III. APPROPRIATENESS OF USING A 2,000-YEAR RETURN PERIOD EARTHQUAKE FOR THE SEISMIC DESIGN OF THE PFSF

A. General Principles of Risk-Informed Seismic Design

Q19. Please describe the seismic design basis for the PFSF.

A19. PFS has performed the seismic analysis and design of important-to-safety structures, systems, and components at the PFSF using design basis earthquake (or "DBE") ground motions associated with a mean annual probability of exceedance of $5x10^{-4}$ (i.e., a 2,000-year mean annual return period, or 2,000-year MRP) and applying those ground motions to the design criteria and procedures of the NRC's SRPs for nuclear systems, structures, and components (SSCs).

Q20. Based on your work using PSHA and developing codes and standards for earthquake safety, are there any general principles that provide guidance on the adequacy of PFS's proposed seismic design basis for the PFSF?

A20. Yes. General principles of risk-informed seismic design can be used to judge the adequacy of the seismic design basis proposed for the PFSF. The first such general principle is that there should be a risk-graded approach to seismic safety which allows facilities and structures with lesser failure consequences to have larger mean annual probabilities of failure. A second general principle is that the adequacy of a design basis earthquake ("DBE") to provide the desired level of seismic safety is to be judged by considering both the mean annual probability of exceedance of the DBE and the level of conservatism incorporated into the design criteria and procedures.

Q21. Please describe the first principle that you identified, use of risk-graded approach for establishing seismic design standards.

A21. Most modern seismic design criteria are based on the principle that the probability of SSC failure (where failure is defined as reaching or exceeding a behavior mode that may preclude the SSC from fulfilling its intended function, e.g., containment of hazardous material,) that needs to be addressed in the design is inversely related to the consequences of such failure. In other words, the less severe the anticipated consequences of SSC failure, the larger the probability of failure that can be tolerated.

Thus, SSCs whose seismic failure would cause less severe consequences are designed to allow for higher probabilities of failure. The State's seismic expert witness in this proceeding agrees that it is appropriate to use the risk-graded approach underlying the use of PSHA for the seismic analysis and design of SSCs. Arabasz Dep. at 59-60.

Q22. What are the underlying reasons for applying a risk-graded approach to seismic safety?

A22. The fundamental reasons supporting the use of a risk-graded approach to seismic analysis and design are notions of equity and efficiency: the public should be provided comparable levels of safety for various societal activities, and the greatest overall safety is obtained if seismic safety resources are distributed rationally among different projects [Ref. 12 (Paté-Cornell, Structural Safety Journal)]. Examples of seismic standards that explicitly use this principle include the draft International Standards Organization ("ISO") guidelines for offshore structures [Ref. 13 (Banon et. al., OMAE 2001)], of which I am a co-author, Federal Emergency Management Agency ("FEMA") guidelines for building assessment [Ref. 14 (FEMA 273 pp. 2-5)], and DOE Standard 1020-94 [Ref. 11 (Table B-1, p. B-5)]. Further, the NRC Staff has stated, with respect to the seismic design of nuclear facilities: "The use of probabilistic techniques and a risk-graded approach are compatible with the direction provided by the Commission on Direction Setting 12, 'Risk-Informed, Performance-Based Regulation.'" [Ref. 15 (SECY-98-071 pp. 3-4)].

Q23. Has the NRC made any determination of the relative risk posed by ISFSIs, such as the PFSF, compared to those posed by operating nuclear plants?

A23. Yes, the NRC has stated that the potential consequences of failure of ISFSIs are much less severe than those for NPPs. For example, the Commission has rejected the notion that licensing standards should be as high for ISFSIs as for NPPs, noting that "[t]he potential ability of irradiated fuel to adversely affect the public health and safety and the environment is largely determined by the presence of a driving force behind dispersion. Therefore, it is the absence of such a driving force, due to the absence of high temperature and pressure conditions at an ISFSI (unlike a nuclear re-

actor operating under such conditions that could provide a driving force), that substantially eliminate the likelihood of accidents involving a major release of radioactivity from spent fuel stored in an ISFSI." [Ref. 16 (60 Fed. Reg. 20,883 (1995))]. Indeed, the Commission has stated in the context of the PFSF case that "Our flexible approach to financial assurance in nonreactor cases appropriately reflects differing levels of risk." Private Fuel Storage (Independent Spent Fuel Storage Installation), CLI-00-13, 52 NRC 23, 30 (2000). The Commission further supported "the Board's risk calculus [holding that a ISFSI presents safety risks more closely comparable to a uranium enrichment plant is] reasonable." Id. at 31. "[T]he Commission has previously stated that a spent fuel storage facility, which holds fuel that has been cooled for at least 1 year and is not subject to dispersive forces associated with high temperature and pressure, has a much smaller potential for serious accidents than a power reactor." Id. (citations omitted). Thus, the Commission has determined that an ISFSI, by virtue of the largely passive nature of its operation, poses much less risk than a nuclear power plant, which relies on active cooling and safe-shutdown systems to maintain the integrity of the high-pressure reactor coolant boundary and shut down after an earthquake.

Q24. In terms of the appropriate level of seismic safety, what is the significance of the Commission's determination that ISFSIs pose much less risk than an operating nuclear power plant?

A24. Because the Commission has determined that the potential consequences of seismic failure of ISFSIs are much less severe than those for nuclear power plants, under the risk-graded approach to the seismic design, ISFSIs, such as the PFSF, can be allowed higher annual probability of failure due to seismic events than NPPs.

Q25. Please elaborate on the second general principle stated above, that a combination of both the mean annual probability of exceedance of the DBE and the level of conservatism incorporated into the design criteria and procedures determine the adequacy of a DBE to provide the desired level of seismic safety.

A25. While the risk-graded approach is implemented in somewhat different ways in the various fields of seismic design, the standards of practice almost invariably utilize a DBE defined at some mean annual probability of

exceedance ("MAPE") and a set of design procedures and acceptance criteria. Both the design procedures and the acceptance criteria (e.g., applicable codes and standards) include conservatisms that implicitly or explicitly implement "performance goals" (e.g., target levels of the seismic failure probability for the SSCs), which are defined in a manner reflecting the anticipated consequences of the failure. These conservatisms are typically not explicitly stated, but are embedded in the design procedures and the various codes and standards pursuant to which the design of an SSC is accomplished.

Q26. Please describe how the MAPE of the DBE and the level of conservatism incorporated in the applicable codes and standards affect the failure probability of seismically-designed SSCs.

A26. The desired level of seismic safety can be achieved by adjusting either the MAPE of the DBE or the level of conservatism of the design procedures and acceptance criteria, or by adjusting both elements simultaneously. For example, a lower (or higher) failure probability can be achieved by keeping the design procedures and acceptance criteria fixed while reducing (or increasing) the MAPE of the DBE; or, alternatively, by fixing the MAPE while making the design procedures more or less conservative; or by adjusting both elements simultaneously. (A concrete example of the last approach is described below in association with a 2002 revision of DOE-STD-1020-94.) Whichever choice is made among these alternatives, it is important to understand that both the MAPE and the level of conservatism in the design procedures and acceptance criteria must be considered when assessing and comparing the safety implications of various seismic design standards. One fact remains true, however: because of the conservatisms incorporated in all seismic design procedures and acceptance criteria, the probability of failure of a seismically-designed facility or SSC is virtually always less than the MAPE of the governing DBE. In other words, virtually all facilities and SSCs designed against a given DBE have a mean return period to failure that is longer than the mean return period of the earthquake for which they are designed. In practical terms, this means that

seismically-designed SSCs are able to withstand a more severe, i.e., more infrequent, earthquake than that used as the DBE.

Q27. Can you give an example of the application of these principles of risk-graded seismic design?

A27. The application of these principles of risk-graded seismic design is perhaps most clearly and explicitly seen in the U.S. Department of Energy's Standard 1020-94. The basis for DOE Standard 1020-94 is a set of "performance categories" (1 to 4) for seismically designed[3] SSCs with increasing consequences of failure, and thus decreasing probabilities of failure as their performance goals [Ref. 11 (DOE-STD-1020-94, p. 1-2, Section B-2, and Table B-1)]. DOE is responsible for (1) facilities such as ordinary buildings (Performance Category 1 or PC1) designed to protect occupant safety, (2) essential facilities and buildings that should continue functioning after an earthquake with minimal interruption (PC2), (3) important facilities such as ISFSIs that contain hazardous materials[4] (PC3), and (4) critical facilities such as those involving nuclear reactors (PC4).

The performance goals for DOE structures, systems and components in the four performance categories PC1 to PC4 in DOE-STD-1020-94 are set as mean annual failure probabilities of 10^{-3}, $5x10^{-4}$, 10^{-4}, and 10^{-5}, respectively [Ref. 11 (DOE-STD-1020-94, p. Table B-1)] reflecting the increasing consequences of failure. On the other hand, the mean annual probability of exceedance (MAPE) for the design basis ground motions are set as $2x10^{-3}$, 10^{-3}, $5x10^{-4}$, and 10^{-4}, respectively. These values are uniformly larger than the performance goals.

To bridge the gap between the performance goals and the DBE MAPEs, the DOE-STD-1020-94 standards call for design procedures and acceptance criteria that vary among the categories, ranging from those "corresponding closely to model building codes" for PC1 and PC2 , to those for PC4 which "approach the provisions for commercial nuclear power plants" [Ref. 11 (DOE-STD-1020-94, p. 2-2, C-4 to C-5)]. The quantitative effect, in

[3] There is a fifth category, PC0, for which there are no seismic requirements.

[4] For PC3 SSCs the performance goal is set relative to "damage beyond which hazardous material confinement and safety-related functions are impaired" [Ref. 11 (DOE-STD-1020-94 at pg B-8)].

terms of reducing earthquake risk, of applying the conservatisms built into these various design procedures and acceptance criteria is reflected in the ratios between the MAPE of the design basis ground motions and the corresponding performance goal probabilities. These ratios are 2, 2, 5 and 10, respectively [Ref. 11 (DOE-STD-1020-94, p. C-5)]. The ratios are called "Risk Reduction Ratios", R_R, in DOE-STD-1020-94 parlance. The following table summarizes these three parameters, the DBE MAPE, the Performance Goal, and the R_R for the four performance categories PC1 through PC4 in DOE-STD-1020-94:

TABLE 1: DOE STD 1020-94 SEISMIC PERFORMANCE GOALS, DBE MAPES AND R_Rs

Performance Category	Target Seismic Performance Goal (P_F)	DBE Exceedance Probability (MAPE)	Risk Reduction Ratio (R_R)
PC1 (e.g., office building)	$1x10^{-3}$	$2x10^{-3}$	2
PC2 (e.g., essential building that should remain operational, such as hospital or police station)	$5x10^{-4}$	$1x10^{-3}$	2
PC3 (e.g., hazardous waste facilities such as ISFSIs)	$1x10^{-4}$	$5x10^{-4}$ (except $1x10^{-3}$ for Western sites near tectonic boundaries)[5]	5 (except 10 for Western sites near tectonic boundaries)[3]

[5] The actual value of R_R obtained from the design conservatisms for a given SSC is dependent to some degree on the shape or slope of the ground motion hazard curve. For example, the PC4 value of 10 cited in the table is representative of locations in the Central and Eastern United

Footnote continued on next page

PC4 (e.g., nuclear reactor facility)	$1x10^{-5}$	$1x10^{-4}$ (except $2x10^{-4}$ for Western sites near tectonic boundaries) [3]	10 (except 20 for Western sites near tectonic boundaries) [3]

Q28. Has a revised version of DOE-STD-1020-94 recently been issued?

A28. Yes. A revised version of DOE Standard 1020 was approved in January, 2002 [Ref. 18 (DOE-STD-1020-2002]. The modifications have no effect on the use made of the DOE-STD-1020-94 here. The primary change is that PC1 and PC2 are now based on the IBC 2000 building code instead of the older UBC model building code. This newer code calls for a considerably larger, 2500-year, DBE and, appropriately, much less conservative acceptance criteria (e.g., the ground motions are reduced by a new factor of 1.5) (which I discuss further below). This IBC 2000 code has not been based on an explicit Performance Goal or explicit risk reduction, R_R, values, however, and DOE has not made an effort to estimate them. As a result, the Performance Goals and the R_R values on this table have been left blank in DOE-1020-2002 in those categories.[6] A minor change has also

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States. However, higher risk reduction ratios, e.g., 20 for PC4 facilities, are achieved in western US sites near tectonic boundaries, where hazard curves are considerably steeper [Ref. 11 (DOE–STD-1020-94, Table C-3 p. C-5)]. The higher achievable R_R values have allowed the DOE to specify that higher DBE MAPE levels can be used for PC4 facilities as well as for PC3 facilities in these regions.

[6] Although the R_R column is left blank for PC1 and PC2, it can be shown (using the information in NERHP Recommended Provisions for Seismic Regulations for New Buildings and Other structures [Ref. 19 (FEMA-303 at p. 37)] and the procedures outlined in Attachment A hereto) that the net R_R is still about 2 for PC 2 and, now, because of the 1.5 reduction referred to above, the net value is only about 0.4 for PC1; it is still 2 before this adjustment. If so, then the performance goal achieved for PC1 has remained effectively unchanged at 10^{-3} and that for PC2 has perhaps been implicitly improved. DOE-STD-1020-2002 recognizes these issues stating that the original PC1 and PC2 goals (still cited in Appendix B, Table B-1) are "no longer exact" [Ref. 18 (DOE-STD-1020-2002 at pg C-6)].

been made to the PC3 category to permit the use for PC3 category structures and components of USGS national probabilistic seismic hazard maps. To meet building code needs, these maps are printed for this 2500-year level. Therefore, the DOE-STD-1020-2002 MAPE of PC3 is modified slightly to this $4x10^{-4}$ value. The PC3 performance goal remains 10^{-4}, however. Therefore, the R_R has been reduced from 5 to 4 by making the acceptance criteria somewhat less conservative.[7] This is the example of a conscious, simultaneous change of MAPE and conservatisms referred to above. For simplicity and clarity, because the DOE-STD-1020-94 and the PFSF both have a 2000-year DBE, I shall continue to refer to the original document.

Q29. How is the level of conservatism or risk reduction factors , R_Rs, for DOE-STD – 1020-94 achieved?

A29. In DOE-STD-1020-94, for most SSCs the overall conservatism levels are controlled through conventional "deterministic" acceptance criteria to achieve specific R_R levels [Ref. 11 (DOE-STD-1020-94, pg. 1-5)]. For the categories of more interest here, PC3 and PC4, this has been accomplished by specifying certain procedures, parameter values, and material standards [Ref. 11 (DOE-STD-1020-94, Chap. 2)] that permit calculation of a SSC's earthquake resistance capability ("capacity") versus earthquake and other loadings ("demand"). Capacity and demand are compared to determine whether compliance with the acceptance criteria is achieved. In DOE-STD-1020-94, the conservatisms have been "intentionally introduced and controlled" [Ref. 11 (DOE-STD-1020-94, at pg. C-6)]. For example, the seismic portion of the demands is obtained by estimating the force on the SSC due to the design basis earthquake and then multiplying this demand by a factor, SF, whose value has been carefully calibrated by probabilistic calculations (described in the document [Ref. 11 (DOE-STD-1020-94, Section C.2.2)]) to achieve the value of R_R appropriate to the

[7] A factor referred to as SF in Eq. 2-1 and 2-7 [Ref. 18 (DOE-STD-1020-2002)] has been reduced from 1.0 to 0.9 to accomplish this change.

DBE MAPE and performance goal of category PC3 and of category PC4 (5 and 10 respectively, for most regions).

Q30. Do the design acceptance criteria and procedures for NRC-licensed facilities contain similar conservatisms, or risk reductions factors, as those embodied in DOE-STD-1020-94?

A30. Yes. It is well established that the design acceptance criteria and procedures guidelines provided by the NRC SRPs contain many conservatisms that result in risk reduction factors as large as, or larger than, those for PC4 category facilities designed to DOE-STD-1020-94. NRC SRP standards share with DOE's PC3 and PC4 categories many procedures leading to design conservatism [Ref. 11 (DOE-STD-1020-94, pp. C-5, C-6)]. These conservatisms are introduced through prescribed analysis methods, specification of material strengths, limits on inelastic behavior, etc. The conservatism levels in NRC seismic SRPs are not explicitly keyed to values of R_R, however. Nonetheless, the risk reduction factors achieved through the use of NRC guidelines for typical SSCs have been found to be equal to, or higher than, those called for in DOE-STD-1020-94 for PC4 facilities.

Q31. Is this higher level of conservatism compared to DOE-STD-1020-94 provided by the design criteria embodied in the NRC SRPs expressed anywhere?

A31. Yes. DOE-STD-1020-94 acknowledges the higher R_R levels provided by the NRC SRPs by stating that the "[c]riteria for PC4 approach the provisions for commercial nuclear power plants". [Ref. 11 (DOE-STD-1020-94, p. 2-2, C-4 to C5). Further, there is recent independent technical support both for the general conclusion that NRC SRPs provide equal or greater levels of conservatism than DOE-STD-1020-94, and for the quantitative finding that the levels of the risk reduction factor, R_R, for typical systems, structures, and components designed to NRC SRPs are in the range of 5 to 20 or greater [Ref. 20 (NUREG/CR-6728 at Chapter 7)].[8]

Q32. What do you mean by typical systems, structures and components?

[8]Demonstration of these conclusions requires a somewhat detailed technical discussion, which is presented in Attachment A to this Testimony.

A32. By typical systems, structures and components I mean those SSCs which are representative of SSCs commonly found in commercial nuclear power plants. These are the SSCs that have been evaluated in the many seismic PRAs and seismic margins studies upon which the experience base has been built to reach these general conclusions about the 5 to 20 or greater range of NPP SSC R_R values. As used here, the term typical SSCs is restricted further to exclude brittle SSCs, which are not found in any case among those in the PFSF.

Q33. What would be expected for other components assuming that they were designed to NRC SRPs?

A33. Given the decades of NRC's concern about seismic safety, and given the code, standards and criteria they call for, one would expect *a priori* similar levels of conservatism in any SSC designed to their SRPs and hence a similar range of R_R levels. For a SSC such as a free-standing storage cask, which is not typical of commercial NPPs, the level of conservatism can be demonstrated by specific analysis. This has been done here by finding a lower bound on R_R based on beyond-design-basis analyses by Holtec and the NRC Staff with respect to the HI-STORM 100 storage system, as discussed further below.

B. Application of General Principles to the PFSF

1. Application of NRC SRP Risk Reduction Factors to ISFSIs

Q34. You stated earlier that PFS has performed the seismic design for important to safety SSCs at the PFSF using a 2,000-year mean annual return period earthquake and applying the design criteria and procedures of the NRC's SRPs for nuclear components. What do you mean by the NRC's SRPs for nuclear components?

A34. I mean the SRPs that the NRC has established for various facilities that it licenses. These SRPs set forth the acceptance criteria and procedures for designing the facility, typically referring to standards and codes specifically developed for the design and construction of nuclear components, such as the code for the Seismic Analysis of Safety-Related Nuclear Structures developed by the American Society of Civil Engineers, ASCE

4-86[9] and the "Code Requirements for Nuclear Safety Related Concrete Structures" of the American Concrete Institute, ACI 349,[10] to which the PFSF has committed. Specifically, the NRC has a Standard Review Plan for nuclear power plants, NUREG-0800,[11] which specifies the design procedure and acceptance criteria for nuclear power plants. Likewise, the NRC has a Standard Review Plan for Independent Spent Fuel ISFSIs, NUREG-1567,[12] and one for dry cask storage systems, NUREG-1536.[13]

Q35. Is the conclusion that the R_R levels for typical systems, structures, and components designed to NRC SRPs are in the range of 5 to 20 or greater premised on the application of any particular SRP?

A35. As stated above, the basis for this conclusion is the history of seismic PRA and margins studies conducted on commercial nuclear power plants designed to NUREG-0800, the SRP for such facilities. However, by virtue of the general commonality of the design procedures and acceptance criteria called for in other SRPs, that the conclusion is equally applicable to SSCs designed to the NRC dry storage SRPs cited above. This commonality is discussed below and in the testimony of other PFS witnesses.

Q36. What is your familiarity with these SRPs?

A36. I have been involved for most of my professional career with the evolution of key parts of the seismic portions of NUREG-0800, the SRP for commercial NPPs. In particular, I am very familiar with the assessment of vibratory ground motions (Section 2.5.2) and seismic design parameters

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[9] [Ref. 31 (American Society of Civil Engineers, ASCE 4-86, *Seismic Analysis of Safety-Related Nuclear Structures and Commentary for Seismic Analysis of Safety Related Nuclear Structures*, September 1986)].

[10] [Ref. 34 (American Concrete Institute, ACI-349, *Code Requirements for Nuclear Safety-Related Concrete Structures*, 1999)].

[11] [Ref. 2 (U.S. Nuclear Regulatory Commission, NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, August 1988)].

[12] [Ref. 1 (U.S. Nuclear Regulatory Commission, NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities*, March 2000)].

[13] [Ref. 38 (U.S. Nuclear Regulatory Commission, NUREG-1536, *Standard Review Plan for Dry Cask Storage Facilities*, January 1997)].

(Section 3.7.1), and the documents they refer to. As explained earlier, I participated in the development of Section 100.23 of 10 C.F.R. Part 100, Regulatory Guide 1.165, the EPRI and LLNL PSHA studies of CEUS (Central and Eastern U.S.) sites, and the Senior Seismic Hazard Analysis Committee (SSHAC) report. Other sections of NUREG-0800 relevant to seismic safety, e.g., those defining load combinations, acceptable codes (such as ASME Boiler and Pressure Vessel,[14] ACI 349, [15] AISC,[16]) etc., are similar in content if not in detail to other seismic criteria that I have worked with my entire career. I have reviewed recently the NPP SRP, NUREG-0800. My familiarity with NUREGs-1567 and 1536, the SRPs for dry storage systems, was limited before beginning my work on the PFSF, but I have reviewed them in the context of that work.

Q37. Based on your review of NUREGs-1536 and 1567, do you have any opinion on the similarity of conservatisms embodied in the acceptance criteria and procedures of 1536 and 1567 compared to those encompassed within NUREG-0800 as they relate to seismic design?

A37. Yes. That review confirmed the similarities in the seismic elements of the ISFSI and NPP SRPs. Some specific examples follow. Both set of requirements call for use of Regulatory Guide 1.165 [Ref. 4 "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," 1997] and accept Regulatory Guide 1.60 [Ref. 37 "Design Response Spectra for Seismic Design of Nuclear Power Plants"]. For damping levels, which introduce important conservatisms, both NUREG-1567 and NUREG-0800 reference the NRC Regulatory Guide 1.61 [Ref. 39 "Damping Values for Seismic Analysis for Nuclear Power Plants," 1974]. For reinforced concrete structures (other than the casks themselves, e.g., as would be used with a cask transfer building)

[14] [Ref. 35 (American Society of Mechanical Engineers, *ASME Boiler and Pressure Vessel Code-Nuclear Power Plant Components*, Section III, 1989)].]

[15] [Ref. 34 (American Concrete Institute, ACI-349, *Code Requirements for Nuclear Safety-Related Concrete Structures*, 1999)].

[16] [Ref. 36 (American Institute of Steel Construction, *Manual of Steel Construction, Allowable Stress Design*, 1989)].

the ISFSI SRPs, like that for NPPs, call for application of ACI-349 [Ref. 34 "Code Requirements for Nuclear Safety Related Concrete Structures"]. Finally all three SRPs cite frequently Section III of the ASME Boiler and Pressure Vessel Code. [Ref. 35] Such similarities explain why one can anticipate very similar levels of conservatism from both the NPP and IS-FSI SRPs.

Q38. Do you have any other basis on which to conclude that the SRPs for ISFSIs generally embody the same level of conservatism as NUREG-0800?

A38. Reviewing the testimony filed by PFS of Dr. Alan Soler, Dr. Krishna Singh, Mr. Bruce Ebbeson, Mr. Paul Trudeau, and Dr. Wen Tseng, I see that they used the standards and codes generally applicable for nuclear components, such as those cited above, which are the same standards and codes referenced in NUREG-0800. Further, they have stated that they generally used the same design criteria and procedures applicable to nuclear power plants.

Q39. What conclusion do you draw based on your review and understanding of the SRPs and the testimony of those responsible for the design of the PFSF structures and components?

A39. Because important-to-safety structures, systems and components at the PFSF are designed to the same codes and standards as those for nuclear power plants, the conclusion that the R_R levels for typical systems, structures, and components designed to NRC SRPs are in the range of 5 to 20 (or greater) would apply to such structures systems and components at the PFSF.

Q40. What SSCs important to safety at the PFSF would clearly fall under the rubric of "typical" SSCs designed to NRC SRPs for which a R_R of 5 to 20 (or greater) would apply?

A40. PFSF SSCs in the CTB, including the building itself, its roof, the cranes and the seismic struts, clearly fall under this category because the same (or very similar) SSCs occurring in the NPPs have been analyzed in the many seismic PRAs and margins studies that provided the experience upon which this general range of R_R values is based. Several projects have developed guideline procedures based on such general R_R observations (e.g.,

the NRC and EPRI margins methods, DOE-STD-1020-94, and most recently NUREG/CR-6728, in which, as cited above, ~~Answer Attachment A~~ one can find the quote that is the basis for the conclusion that typical NPP R_Rs are 5 to 20 or more). The results of these studies have been evaluated and/or collected and summarized in seismic PRA and margins projects I have been involved in the past, e.g., the Diablo Canyon seismic PRA, the LLNL Seismic Margins project, and the development of NRC seismic margins methodology.

Q41. What about the foundation to the Canister Transfer Building or the storage cask pads for the spent fuel casks?

A41. The NPPs whose seismic PRAs and margins studies form the basis of the R_R values cited have buildings with foundations generally analogous to that of the CTB. While I am personally less familiar with the foundation SPRA results, I am aware that they have been prepared for potential foundation failure modes such as overturning, bearing, and sliding. While it is not entirely clear whether the R_R range conclusion (based on NUREG-6728) was intended to apply to foundations, it can be presumed, nonetheless, that given the NRC's many years of concern for seismic safety and for margins beyond the design basis, that comparable levels of conservatism in foundations have been provided by their criteria and by practice in the field, and hence that comparable levels of R_R likely exist with respect to performance that might jeopardize hazardous materials containment.

Q42. What about the spent fuel storage casks themselves?

A42. As described in the testimony of Dr. Alan Soler, the spent fuel storage casks are designed to the ISFSI SRP NUREG-1536 [Ref. 38] discussed above. They are also designed for other SRP-dictated accident conditions, such as hypothetical drop and tip-over events. With respect to direct seismic inertial forces, it can be expected for the reasons cited above that their R_R values will equal or exceed the 5 to 20 range of typical NPP components. (Indeed, it has been confirmed that for these effects the HI-STORM 100 storage system has very large margins.) As stated above, these casks are not common NPP SSCs but, as will be discussed below, consideration of the Holtec and Sandia analyses of the HI-STORM 100

system with respect to beyond-design-basis earthquake motions and with respect to potential tip-over conditions shows that the effective R_R of the cask system is in excess of 5. Thus, the design of this cask system provides risks reduction factors comparable to those available for typical NPP SSCs.

2. Appropriate Risk Reduction Factors for the PFSF

Q43. Do you have an opinion as to the risk reduction factors applicable to the seismic design of the PFSF?

A43. Yes.

Q44. What in your opinion is an appropriate seismic risk reduction factor to represent the SSCs in the PFSF?

A44. Based on the established and demonstrated margins, I believe that a risk reduction factor of five or more is appropriate for important-to-safety SSCs in the PFSF.

Q45. What is the general basis for your opinion?

A45. The basis for my opinion is (1) my general knowledge and experience regarding risk reduction factors as applied to many different types of structures designed to a wide variety of codes and standards; (2) my general knowledge and experience of risk reduction factors applicable to nuclear power plants designed in accordance with the applicable design codes and standards as specified by the NRC NPP SRP (NUREG-0800); (3) my independent review of the SRPs applicable to ISFSIs and spent fuel storage casks (NUREGs 1567 and 1536) and confirmation that the codes and standards applicable to nuclear power plants are generally applicable to ISFSIs, such as the PFSF; (4) confirmation by those responsible for the design of the structures and components at the PFSF that such structures and components are generally designed to the same codes and standards applicable to nuclear power plants; (5) analytical and qualitative demonstration by those responsible for the design of the PFSF of significant beyond-design-basis margins for structures and components important to safety; (6) the limited fraction of time that certain SSCs are in use; (7) demonstration by Holtec that casks at the PFSF will not tip-over at the

10,000-year earthquake and (8) demonstration by Holtec that a postulated cask tip-over will not result in breach of a cask and release of radioactivity.

Q46. What structures and components have you considered as important to safety in your review?

A46. In my review, I considered the Canister Transfer Building and the cranes and the seismic struts inside the CTB used in transferring the spent fuel canisters from the transportation casks to the storage casks. I also considered the spent fuel storage casks and the storage cask pads on which they are placed.

Q47. On what basis did you decide that these were the appropriate structures and components to consider in your evaluation of risk reduction factors for the PFSF?

A47. I depended on information provided by PFSF personnel, such as the testimony of Mr. Wayne Lewis.

Q48. Please describe the basis of your opinion that the risk reduction factor for the Canister Transfer Building and the cranes and struts inside the building is 5 or more?

A48. The Canister Transfer Building itself and the cranes and seismic struts inside the building are typical of nuclear power plant components for which the risk reduction factor has been shown to be a factor of 5 to 20 or more. That basis alone would be sufficient to conclude that the CTB and the cranes and seismic struts inside the CTB have a risk reduction factor of five or more.

Q49. What else, if anything, do you base your opinion that the risk reduction factor for the Canister Transfer Building and the cranes and struts inside the building have a risk reduction factor of 5 or more?

A49. I rely upon facts described in the testimony of Mr. Bruce Ebbeson, the individual responsible for the design of the CTB, and Mr. Wayne Lewis. First, Mr. Ebbeson's testimony confirms that these components were designed to nuclear power plant standards, where applicable, suggesting that the general conclusion about the R_R values of typical NPP SSCs applies. Second, the beyond-design-basis analyses and margins described in the testimony of Mr. Ebbeson confirm the existence of significant beyond-

design-basis margins in the design of the CTB and the cranes and struts therein, which would enable them to survive earthquake ground motions much greater than those of the 2000-year design basis earthquake. Third, as described in the testimony of Mr. Wayne Lewis, the CTB cranes and seismic struts are in use only a fraction of the time, and thus a canister would be exposed to potential risk of damage due to their failure only a fraction of the time. For such intermittent-use components, the annual likelihood of failure during a safety-important operation is reduced further. For example, even if the fraction of time they are used is 20%, the annual probability of failure causing release due to earthquake ground motions is 5 times smaller. This implies that, even if their R_Rs due to SRP conservatisms were only unity instead of the factors of 5 to 20 or more estimated above, the relevant frequencies of failure of these SSCs would be less than 10^{-4}. With the predicted R_R of 5 to 20 or more, this estimated failure frequency reduces to about 10^{-5}. In short the effect of the 20% use fraction is, in effect, to increase R_R by a factor of 5.

Q50. What about the foundations for the CTB? Have you considered and determined whether a risk reduction factor of 5 or more is applicable to the CTB foundations?

A50. As discussed earlier, based on the NRC's long concern over seismic safety margins there is *a priori* reason to expect that an R_R comparable to those of typical NPP SSCs is available with respect to those modes of PFSF CTB foundation behavior that might lead to loss of containment of hazardous materials. As presented in the testimony of PFS witnesses Mr. Ebbeson and Mr. Trudeau due to differences such as those between calculated and design safety factors, realistic dynamic and the assumed static behavior, mean and the lower bound soil properties, dynamic and static soil properties, etc., that there is significant margin with respect to the ground motions that might cause overturning or bearing failure of these foundations. They conclude that this total expected margin is greater than that needed to meet the 10,000-year ground motions. Local bearing failure would, in any case, likely be tolerated by the building without impairing the performance of hazardous material containments inside it. Therefore these foundation behavior modes can be estimated to have R_R

levels of 5 or more. It has not been demonstrated that the CTB will not slide under ground motions of, say, the 10,000-year level, but, as Mr. Ebbeson states, this sliding would not have negative consequences with respect to loss of containment of hazardous materials.

Q51. Please describe the basis of your opinion that the risk reduction factor for the storage pads is 5 or more?

A51. As discussed in the testimony of PFS witness Paul Trudeau, there are large quantifiable margins of safety against overturning and soil bearing failure at or approaching MRPs 5 times the 2000 DBE level, as well as other significant non-quantified conservatisms. Together these conservatisms safety allow one to reasonably conclude that no overturning or hazardous-to-release bearing failure would be expected under ground motions with MRPs of more than 5 times the 2000-year DBE level. Also, as ~~these~~ PFS witnesses confirm, sliding of the storage pads is not expected, per se, to cause hazardous material release. The effect of any such pad sliding on the behavior of the storage casks has been considered in the assessment of the cask.

Q52. Please describe the basis of your opinion that the risk reduction factor for the spent fuel storage casks at the PFSF is 5 or more?

A52. As described in the testimony of Drs. Singh and Soler of Holtec, the HI-STORM 100 system storage casks are stubby cylindrical weldments of steel and concrete designed to NRC SRPs to tolerate significant earthquake-induced inertial forces as well as those due to drop and tip-over accidents. Therefore, as discussed above, their margins with respect to the 2000-year design basis motions can be expected to be very significant. As testified by Drs. Soler and Singh, in addition to the assessments required by the NRC SRPs, Holtec and Sandia have conducted 10,000-year ground motion analyses predicting that there will be neither cask tip-over nor cask-cask sliding impacts. They testify further that even should there be tip-over the tip-over analysis conducted by Holtec predicts no breach. As testified by PFS witnesses Drs. Singh and Soler, even should one sliding cask impact another the effects are bounded by the tip-over analysis. Further, Drs. Singh and Soler state that these assessments retain elements

of conservatism, e.g., upper and lower bound cask friction coefficients are used, and the cask could suffer even more damage than predicted before breaching. An upper bound on the probability of loss of containment can be estimated easily by use of this information. Given this prediction of no tip-over under a 10,000-year ground motion, the annual probability of tip-over can be judged to be no more than 10^{-4}. Based on the prediction of no breach given tip-over the conditional probability of breach given tip-over can be judged to be significantly less than one. The annual probability of loss of containment of hazardous material due to cask tip-over is simply the product of these two numbers, which is clearly less than 10^{-4}. Based on the information stated above the annual probability of loss of containment due to cask sliding is clearly much smaller than this bound on that due to tip-over. With the $5x10^{-4}$ MAPE of the DBE, the implied R_R for the storage casks is therefore greater than 5.

3. Adequacy of the 2000-year Design Basis Earthquake for the PFSF under a Risk-Graded Approach to Seismic Safety

Q53. Based on your review of the risk reduction factors applicable to the PFSF, do you have an opinion on whether the 2000-year design basis earthquake for the PFSF provides an adequate level of seismic safety?

A53. Yes.

Q54. Please state your opinion and the bases therefore.

A54. I believe that the PFSF 2000-year design basis earthquake (DBE) provides an adequate level of seismic safety because: (1) based on my review of the risk reduction factors (R_R) applicable to the SSCs important to hazardous material containment discussed above I believe that these factors are 5 to 20 or greater; (2) coupled with the 2000-year ($5x10^{-4}$ MAPE) DBE these R_R levels imply that the PFSF SSCs will have achieved a performance goal of $1x10^{-4}$ or better; and (3) I believe, based on the principle of risk-grading discussed above, that $1x10^{-4}$ is an appropriate performance goal for the SSCs of this spent fuel dry storage facility.

Q55. Please state the basis for your opinion that $1x10^{-4}$ is an appropriate performance goal for the PFSF SSCs.

A55. First, applying the risk-graded seismic principle, a performance objective of $1x10^{-4}$ for SSCs ISFSIs such as the PFSF is consistent with the NRC's performance objectives for operating nuclear plants, which ~~THE~~ the NRC ~~HAS~~ has ~~FOUND~~ found pose higher radiological hazard consequences than ISFSIs. While the NRC nuclear power plant seismic performance goals and the quantitative effects of their design criteria are less explicit than those in DOE Standard 1020-94, inferences can be made from existing NRC standards. The NRC's quantitative safety objective with respect to core damage is a mean annual frequency of $1x10^{-4}$ [Ref. 21 (SECY-00-0077 at p. 6)] ("Mean annual frequency" and "mean annual probability" are effectively equivalent). Some undefined fraction of this "budget" is available for seismically induced core damage. Past NRC seismic standards for nuclear power plants have provided a mean annual seismically-induced core damage frequency of about 10^{-5}. [Ref. 22 (NUREG/CR-5501 (1989) at p. 26)] In NUREG/CR-5501, a study prepared for the NRC, the mean annual seismic core damage frequency of seven existing plants was estimated from seismic PRAs to range from about $4x10^{-6}$ to about $1x10^{-4}$, with most lying between 0.6 and $1x10^{-5}$. DOE-STD-1020-2002 [Ref. 18 at p. B-7] quotes NUREG/CR-5042 as finding the same range in 12 more recent NPPs, while 10 of the 12 plants have such frequencies greater than $1x10^{-5}$. ~~[Ref. 22 (NUREG/CR-5501)]~~ As discussed above, DOE-STD-1020-94 also uses, explicitly, a performance goal of $1x10^{-5}$ for nuclear reactor SSCs. The use of a probability of seismic failure or performance goal for the PFSF SSCs, such as $1x10^{-4}$, higher than that for nuclear power plants SSCs (about $1x10^{-5}$) is consistent with the risk-graded approach of the probabilistic approach.

Second, an SSC performance goal of $1x10^{-4}$ is consistent with DOE policy as represented by DOE-STD-1020-94 and DOE-STD-1021-93. As discussed above, the performance goal stated in DOE-STD-1020-94 for category PC3 SSCs is $1x10^{-4}$. The PFSF important-to-safety SSCs would clearly fall into category PC3. DOE-STD-1021-93 [Ref. 40, "Natural Phenomena Hazards Performance Categorization Guidelines for Structures, Systems and Components," July 1993], which defines such catego-

ries, states (at pg. 2-3) "If the adverse offsite consequences of an NPH [Natural Phenomena Hazard] event are significant enough to make them safety-class but are substantially less than those associated with consequences from an unmitigated large Category A reactor severe accident, the SSCs should be placed in PC-3." The State's seismic expert witness, Dr. Arabasz, agreed that ISFSIs, such as the PFSF, would appropriately be classified PC3 facilities under DOE-STD-1020-94 and that the performance objective of $1x10^{-4}$ for the PFSF SSCs would be an appropriate standard on which to determine the acceptability of its seismic design. Arabasz Dep. at 80-81. I conclude that a performance goal of $1x10^{-4}$ for the PFSF would be consistent with a risk-graded approach to seismic safety. The proposed PFSF seismic design basis of a 2,000-year MRP DBE and the SRP design procedures and criteria will meet such a goal and therefore provide an appropriate and consistent level of protection to public health and safety.

IV. DISCUSSION OF SPECIFIC ISSUES RAISED BY THE STATE OF UTAH

Q56. What claims does the State of Utah raise with respect to Section E on the Unified Contention?

A56. The State raised seven issues in the bases supporting what is now Section E of the Unified Contention, some of which relate to issues discussed above. In addition, in the State's Opposition to PFS's Motion for Summary Disposition on this aspect of the contention, the State's experts dispute certain aspects of the analysis that I provided in a declaration dated November 9, 2001 supporting the PFS Motion.

A. Claims of State's Experts Raised in State of Utah's Summary Disposition Opposition

Q57. Focusing first on the claims of the State's experts in the State's Summary Disposition Opposition, what were the main responses of the State's experts regarding the analysis provided in your November 9, 2001 declaration supporting the PFS Motion?

A57. The State's primary expert supporting the State's contention, Dr. Walter Arabasz, agreed with the two basic principles that I set forth in my analyses, which I have also explained above. Dr. Arabasz agreed with the con-

cepts of (i) using a risk graded approach to seismic safety, and (ii) determining acceptable earthquake performance of a facility or structure based on a combination of the mean annual exceedance period of the design basis earthquake for the structure and the conservatisms embodied in the standards and codes governing its design and construction.[17] Further, Dr. Arabasz did not take issue with my application of those principles to the PFSF in my November 9, 2001 declaration, although other experts of the State, Drs. Bartlett and Ostadan, did take issue with certain parts of the declaration.

Q58. What issues did Drs. Bartlett and Ostadan raise with respect to your November 9, 2001 declaration?

A58. Generally, their issues involved the risk reduction factors applicable to the PFSF. They claimed that PFS could not rely upon the risk reduction factors specified by DOE-STD-1020-94 or derived from NUREG/CR-6728 because the PFS design does not meet the intent or requirements of either document.[18] They further claimed that the risk reduction factors applicable to typical SSCs at nuclear power plants are not applicable to SSCs at the PFSF because the NRC Standard Review Plan ("SRP") requirements for nuclear power plants are not applicable to important-to-safety SSCs at the PFSF, and that "the SRPs in NUREG 1536 and 1567" applicable to SSCs at the PFSF "may already incorporate less conservatism than" the SRP for nuclear power plants.[19]

Q59. Let's address Drs. Bartlett and Ostadan's claims in the reverse order that you just mentioned. What about their claim that the SRPs applicable to the PFSF "may incorporate less conservatism" than the SRP for nuclear power plants?

A59. Their claim that the SRPs applicable to the PFSF "may incorporate less conservatism" than the SRP for nuclear power plants is erroneous, at least insofar as the design of the PFSF is concerned. As I discussed above, the

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[17] Declaration of Dr. Walter J. Arabasz ("Arabasz Decl.") (Dec. 7, 2001 ¶¶ 18-19).

[18] Joint Declaration of Dr. Steven F. Bartlett, Dr. Moshin R. Khan and Dr. Farhang Ostadan ("Joint Utah Decl.") (Dec. 7, 2001) ¶ 49.

[19] Id. ¶ 49.

design of important-to-safety SSCs at the PFSF is based on essentially the same nuclear codes and standards specified in NUREG-0800, the SRP for nuclear power plants. Therefore, it is appropriate, to utilize the seismic risk reduction factors of 5 to 20 or more for typical nuclear power plant SSCs to the corresponding SSCs at the PFSF.

Q60. What is your response to the claims raised by the Drs. Bartlett and Ostadan that your reliance on NUREG/CR-6728 is inappropriate?[20]

A60. As set forth in paragraph 25 of my November 9, 2001 declaration and Attachment A thereto (which is the same as Attachment A to this testimony except for minor edits and corrections), I rely upon NUREG/CR-6728 for the basic quantitative input that leads directly to the general proposition that the risk reduction factor, in DOE-STD-1020-94 parlance, for "typical ~~components~~ SSCs" designed to the NRC SRP are in the range of 5 to 20 or greater." See Attachment A at 4. As I describe in Attachment A, this range of risk reduction factors is based on the compilation of the "numerous engineering evaluations of safety margins and 'fragility curves' of SSCs designed to the SRP that have been conducted over the last 20 years in the course of research by the industry and NRC contractors, and on the seismic probabilistic risk assessments and seismic margins studies that have been undertaken at virtually all nuclear power plants in the US (via the NRC IPEEE program). These evaluations have been made by earthquake engineers familiar with nuclear power plant SSC designs prepared to the NRC SRP procedures and criteria, and with the actual behavior of such SSCs in earthquakes as observed in the field and tested in the lab." Attachment A at 3. I have been associated with many of these evaluations as I have described above. As set forth in Attachment A, this experience is summarized in NUREG/CR-6728 as a factor of safety applicable to "typical components SSCs" for nuclear power plants that corresponds in DOE-STD-1020-94 parlance to a risk reduction factor in the range of "5 to 20 or greater." See Attachment A at 4. Therefore, my reliance on NUREG/CR-6728 is appropriate.

[20] Joint Utah Dec. ¶ 56.

Q61. Please define what you mean by a "fragility curve" referred to in your previous answer.

A61. A fragility curve is a quantitative representation of the capacity of a component or structure with respect to seismic ground motion, reflecting both the engineer's best estimate of that capacity and the uncertainty ~~above~~ about the value of that capacity. Graphically, it is an S-shaped curve that plots the probability of failure versus the level of the ground motion. To develop this curve, the engineer must provide, first and by far most importantly, his best estimate (median) of the SSC' capacity. This determines the midpoint of the S-shaped curve. This estimate must be based on removing all conservatisms inherent in customary engineering calculations. The most realistic judgments should be made, even if they are only estimates of what a more detailed analysis might show conclusively. The estimation of the median capacity is unrelated, in principle, to design basis ground motions, codes and standards, etc. It is much more akin to a scientific prediction than to a conventional engineering design assumption. On the other hand, the median capacity, when compared to the capacity as determined instead by codes and standards and standard engineering practices, becomes a quantitative measure of the conservatisms implicit in those standards and practices. Such conservatisms are inevitable because the purpose of customary calculations is to demonstrate compliance to codes and standards, which dictate conservatisms. In addition standard engineering practices introduce additional conservatisms, e.g., selection of a conservative value to represent scattered material property data, and avoidance of making realistic but potentially contentious assumptions simply to avoid delay of acceptance by reviewers.

The second element in a fragility curve is a figure that reflects the uncertainty in the median estimate.[21] This number reflects how narrowly or

[21] In DOE-STD-1020-94 Appendix C ,this is referred to as beta, β, which is formally the standard deviation of the natural logarithm of the capacity, but it is more easily understood as being very roughly the fractional standard deviation of the capacity. A typical nuclear power plant SSC beta is 0.45 [Ref. 21 (NUREG/CR-6728 at pg. 7-15)], implying the standard deviation is about 45% of the median. With typical (e.g., DOE-STD-1020-94) assumptions, this in turn means that there is

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widely the S-shape spreads about the best estimate or median. Its value is based on the scatter in relevant data and the judgment of engineers as to the limitations of the various physical models used to predict the capacity. This number plays a comparatively smaller role in the fragility curve estimation in that conclusions based on the fragility curve are much less sensitive to it than they are to the median (best estimate) that is used. Once the fragility curve is developed for a particular SSC, it can be used together with the site's probabilistic hazard analysis to estimate the annual probability of failure of the SSC in question. With this annual probability of failure and the mean annual probability of exceedance of the design basis earthquake, one can determine the risk reduction factor inherent in the design of the SSC. In nuclear industry practice, there exist guidelines for the preparation of fragility curves, and hundreds of examples of their use. Some of the general conclusions can be distilled from these examples as to the effect of applying particular codes and standards; hence, for example, the ability to make such statements as the risk reduction factor of a typical component designed to nuclear power plant standards and practice are in the range of 5 to 20 or more.

Q62. What about Drs. Bartlett and Ostadan's claim that the risk reduction factors of 5 to 20 derived from nuclear power plant experience do not apply to unanchored dry storage casks that are free standing on concrete pads and which may slide and tip because the fragility curves relied upon in NUREG/CR-6728 did not include fragility curves for unanchored storage casks?

A62. I agree that the fragility curves for sliding and tipping of freestanding casks were not developed as part of the seismic evaluations on which the 5 to 20 factor for typical nuclear power plant components is based. However, as discussed above, given the decades of NRC's concern about seismic safety, and given the codes, standards and criteria they call for, one would expect *a priori* similar levels of conservatism in any SSC designed to their SRPs, such as the HI-STORM 100 casks, and hence a similar

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about a 84% chance that the capacity will be greater than or equal to about 55% of the median (more precisely, under lognormal assumptions, 63%).

range of R_R levels. In such cases, such a factor could be estimated by conducting a fragility analysis, as Drs. Bartlett and Ostadan call for, but it is necessary here only to demonstrate that the R_R factor is larger than 5; this has been affirmatively demonstrated through various analyses conducted by Holtec and the NRC Staff.

Q63. What is your response to the claims raised by the Drs. Bartlett and Ostadan that PFS cannot rely upon DOE-STD-1020-94 because neither the intent nor the requirements of DOE-STD-1020-94 are met?

A63. Contrary to the claims of Drs. Bartlett and Ostadan, it is not necessary to satisfy the requirements of DOE-STD-1020-94 in order to demonstrate acceptable seismic design of the PFSF, and I am not suggesting such a reliance. The purpose of my testimony (both above and in my November 9 declaration) is not to show explicit compliance with the various acceptance criteria embodied in DOE-STD-1020-94. Rather, I use DOE-STD-1020-94 to demonstrate that there is important support in the industry for the use of a risk graded approach to seismic safety, and as a way to demonstrate the general principles involved in applying a risk graded approach.

In this latter respect, DOE-STD-1020-94 clearly demonstrates that in applying a risk-graded approach the level of seismic performance achieved by a facility's design is a function of both the mean annual probability of exceedance ("MAPE"), or mean return period ("MRP"), of the design basis earthquake and the conservatisms embodied in the applicable design codes, standards and acceptance criteria (formally referred to in DOE-STD-1020-94 as the "risk reduction factor incorporated in the design").[22] Thus, as recognized at one point by Drs. Bartlett and Ostadan, I am using DOE-STD-1020-94 as an "analogy."[23] DOE-STD-1020-94 explicit use of a DBE MRP and a "risk reduction factor" shows, by analogy, that it is appropriate to look at the PFSF DBE and the margins inherent in the PFSF

[22] These are principles with which the State's primary expert, Dr. Arabasz, agrees (Arabasz Decl. ¶ 38) and with which Drs. Bartlett and Ostadan also appear to agree. Utah Joint Decl. ¶¶ 22-23.

[23] Id.

seismic design bases as the bases for establishing whether the design of the PFSF SSCs provides an acceptable level of seismic performance.

Q64. You referenced Drs. Bartlett and Ostadan's acknowledgement of your use of DOE-STD-1020-94 as an "analogy." Is there merit in their claim that the conservatisms that PFS believes to exist in its seismic design bases "cannot be analogized to the risk reduction factors in DOE Standard 1020" because PFS has not conducted the "full panoply of analyses required" by DOE-STD-1020-94?[24]

A64. This claim of Drs. Bartlett and Ostadan reflects their mistaken view that I rely upon DOE-STD-1020-94 as the authoritative source for the actual seismic risk reduction factors applicable for the PFSF design. That is not the case. As stated above, I rely upon DOE-STD-1020-94 to demonstrate the interplay between the role of the mean return period for the design basis earthquake of a structure and the level of conservatism in its seismic design. The source of my opinion of the applicable seismic risk reduction factors for the PFSF are, as discussed above, (1) the nuclear codes and standards to which SSCs, important to safety at the PFSF, are designed and the conservatism shown to exist for typical components designed and constructed to those codes and standards, supplemented by the testimony of other PFS witnesses who describe and quantify some of the conservatisms in the PFSF design, and (2) specific analyses undertaken to demonstrate the conservatism inherent in the PFSF design, such as the cask stability analyses performed by Holtec and the NRC Staff for the 10,000-year earthquake. Based on this information, I have concluded with <u>no</u> reliance on DOE-STD-1020-94 that the applicable risk reduction factor for PFSF SSCs, important to safety, is 5 or more, and that, together with the 2000-year DBE, achieves a seismic safety performance goal of 10^{-4}, or lower.

Q65. What about Dr. Bartlett and Dr. Ostadan's specific claim that it is necessary to generate "fragility" curves as described in DOE-STD-1020-94 for each SSC important to safety in order to evaluate its seismic design capacity?

A65. As stated above, fragility curves are quantitative descriptions of the expected conservatisms or margins in the design of components and the un-

[24] Utah Joint Decl. ¶ 23.

certainty in these margins. While a fragility curve can be developed to show quantitatively the value of a component's risk reduction factor, it is not required to generate a fragility curve to confirm that a particular component has a risk reduction factor larger than some specified level or can meet a specified seismic performance level.

First, as discussed above, extensive experience has been developed to show that typical SSCs designed to meet the design codes, standards and acceptance criteria specified in the NRC's standard review plans have seismic risk reduction factors of 5 to 20 or more. It is not necessary to generate fragility curves for such typical SSCs to determine whether that they have a risk reduction factor of at least 5, which, together with the 2000-year PFSF DBE is all that is required here to confirm that they will meet a seismic performance goal of $1x10^{-4}$. (DOE-STD-1020-94 does not itself require the generation of fragility curves for such typical SSCs, to confirm a risk reduction factor of 5 or more; it only requires that the component be designed to DOE-STD-1020-94 PC3 criteria.) Further, one can in other ways demonstrate that a SSC meets at least a specified performance goal without generating a fragility curve for the SSC. For example, if the expected (e.g., mean or median) capacity of the component is somewhat larger than a ground motion with a MAPE equal to a performance goal (e.g., $1x10^{-4}$), then it meets the goal.[25] Again, it is important to keep in mind the difference between the median capacity in the fragility curve and the design basis arrived at by applying relevant codes and standards. In the former case conservatisms (such as lower bound properties, static and linear behavior assumptions in place of realistic dynamic and nonlinear considerations) are removed and replaced by best engineering judgments. Yet another way to confirm that the performance goal is met is to show that the probability of failure (e.g., failure to maintain containment of hazardous material) is less than the specified performance goal. For example, if the performance goal is 10^{-4} and the component is estimated not likely to fail under a ground motion with an annual probability

[25] This approach is referred to as a "median-centered" in DOE-STD-1020-2002 (Ref. 18 at C-4).

of exceedance that is less than the performance goal, then the goal has been met..

Q66. Based on what you just stated, is it necessary to generate a fragility curve for the HI-STORM 100 cask, as claimed by Drs. Bartlett and Ostadan, to show that the HI-STORM 100 cask would meet a seismic performance goal of 1×10^{-4}?

A66. No. It is not necessary to develop fragility curves to make the judgment that the HI-STORM 100 cask system will achieve a performance goal of 1×10^{-4} or better. Rather, following the logic that I described just above, one can determine that the HI-STORM 100 cask meets a seismic performance goal of 1×10^{-4} based on the Holtec and Sandia evaluations of the HI-STORM 100 cask system. Based on Holtec's prediction of no cask tip-over under the 10,000-year ground motion and of no release should a cask tip over,[26] it can be concluded that the loss of containment of hazardous material is unlikely given a 10,000-year ground motion, and that the annual probability of loss of containment will be less than 1×10^{-4}. Further, the evaluation performed by Sandia shows that under the 10,000 year ground motion no sliding impact between casks will occur[27] and, as testified to by Drs. Singh and Soler, even if such impact were to occur the velocities and damage of such impacts would be much less than those associated with cask tip-over for which it has been shown that there is no release of radioactivity. Therefore, one can judge that the probability of unacceptable seismic performance due to cask sliding is less than that associated with cask tip-over, i.e., less than 1×10^{-4}. Thus, no fragility curves are necessary to make an informed determination that the HI-STORM 100 cask system will achieve a performance goal of 1×10^{-4} or better at the PFSF.

Q67. Do you then disagree with the claim made by Drs. Bartlett and Ostadan that the selection of "appropriate risk reduction factors can only adequately be conducted

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[26] See Testimony of Krishna P. Singh and Alan I. Soler on Unified Contention L/QQ (April 1, 2002).

[27] Vincent K. Luk, Jeffrey A. Smith and David A. Aube, "Seismic Analysis Report on HI-STORM 100 Casks at Private Fuel Storage Facility," Sandia National Laboratories, March 2002.

by evaluating a thorough uncertainty analysis of the fragility of each SSC at the PFS site, as outlined in DOE-STD-1020-94 and NUREG/CR-6728?"[28]

A67. Yes, I disagree for the reasons I just stated.

Q68. What about the similar claim by the State's experts, Drs. Bartlett and Ostadan, that PFS has not met DOE-STD-1020-94 requirements for foundation failure through, overturning, or sliding or bearing capacity failure?[29]

A68. As stated above, neither I nor PFS is relying on meeting DOE-STD-1020-94 acceptance criteria, so it is not necessary to satisfy the DOE acceptance criteria discussed by Drs. Bartlett and Ostadan in evaluating whether foundations meet a particular seismic performance goal. *A priori* one would strongly expect foundation designs to have safety levels close to those of other NPP elements. While foundation stability and sliding fragility curve calculation at NPPs have been comparatively limited, foundations under safety-related buildings, tanks, and other structures, etc., are present at every NPP, and their performance is considered in seismic PRA and margins studies. It would seem unlikely that, in the closely monitored NRC process, where margins against seismic failures have been the subject of more than two decades of investigation, foundations would be allowed to have lower levels of safety than these structural/mechanical SSCs. As discussed earlier, the risk reduction factors of structural and mechanical SSCs have been found to be 5 to 20 or more. In any case, using the "median-centered" argument I described above, the expected stability (overturning and bearing failure) margins for the CTB and pad foundations are judged, as discussed previously, to be in excess of that needed to confirm that their risk reduction factors are 5 or greater.

Q69. Dr. Ostadan also claimed that revision of DOE-STD-1020-94 to change the DBE for PC3 SSCs from a 2000 to a 2500 MRP earthquake would invalidate the use of the 2000 MRP as the DBE earthquake for the PFSF.[30] Do you agree?

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[28] Utah Joint Decl. ¶ 59.

[29] Utah Joint Decl. ¶ 41.

[30] Joint Utah Decl. ¶ 31.

A69. No. Dr. Ostadan's claim reflects an apparent fundamental misunderstanding of the risk-graded approach to seismic safety incorporated into DOE-STD-1020-94 and the purpose of my reference to DOE-STD-1020-94. As stated above, under the risk-graded approach satisfactory performance is a function of both the mean return period of the design basis earthquake and the level of conservatism embodied in the design of the SSC. I refer to DOE-STD-1020-94 as an example of how this risk-graded approach is applied. I do not rely upon the DOE Standard for either the appropriate DBE or the risk reduction factor appropriate for the PFSF. Therefore, the recent change in DOE-STD-1020-2002 of the DBE for PC3 SSCs from a 2000 MRP earthquake to a 2500 MRP earthquake does not affect my analysis of the appropriateness of using a 2000-year MRP as the DBE for the PFSF. This is particularly true given that the seismic performance goal for PC3 SSCs remains unchanged in DOE-STD-1020-2002 at 1×10^{-4}. Thus, DOE's conclusion regarding an appropriate performance goal for ISFSI SSCs, which is the final product under a risk-graded approach to seismic safety, has not changed in DOE-STD-1020-2002. Indeed, as discussed above, in raising the DBE for PC3 SSCs to 2500 MRP DOE-STD-1020-2002 simultaneously reduced the level of conservatism required for the design of PC3 SSCs, thereby consciously keeping the performance goal the same.

Q70. Would you please summarize your above responses to the claims raised by Drs. Bartlett and Ostadan?

A70. The general claims made by the State's witnesses are that: (1) PSF and I cannot rely on DOE-STD-1020-94 or NUREG/CR-6728 to confirm that, by selecting a 2000-year DBE and applying NRC SRP design standards, the PFSF SSCs meets a performance goal of 10^{-4} per annum with respect to loss of containment of hazardous materials due to a seismic event, and (2) because of possible differences in the nuclear power plant and ISFSI NRC SRPs, the conclusions based on experience with nuclear power plants may not apply. With respect to the first item, we do not rely on these two documents in the ways alleged by the State's witnesses. We rely on NUREG/CR-6728 only for the range of values it provides for a

particular parameter, which in turn confirms our use of the range 5 to 20 or more for the risk reduction factor for typical nuclear power plant components similar to those in use at PFSF. We do not follow the criteria or specific methods of DOE-STD-1020-94, but rely on it only in support of the proposition that a performance goal of 10^{-4} would be appropriate for the PFSF; we also draw an analogy to DOE-STD-1020-94 in that the DOE Standard treats, as we do, the safety or performance goal as a combination of the level of the DBE and the conservatisms in the design of the PFSF. These conservatisms are a direct product of the codes and standards used in the design, as required by the NRC SRPs. With respect to the second item, at least with respect to the PFSF, the nuclear power plant experience is applicable because the same codes and standards applied at the PFSF are those used in nuclear power plants.

B. SPECIFIC CLAIMS RAISED BY THE STATE OF UTAH IN SECTION E OF UNIFIED CONTENTION UTAH L/QQ

Q71. What claims does the State of Utah raise in Section E of the Unified Contention Utah L/QQ?

A71. In Section E of the Unified Contention Utah L/QQ,[31] the State of Utah asserts that:

Relative to the PFS seismic analysis supporting its application and the PFS April 9, 1999 request for an exemption from the requirements of 10 C.F.R. § 72.102(f) to allow PFS to employ a probabilistic rather than a deterministic seismic hazards analysis, PFS should be required either to use a probabilistic methodology with a 10,000-year return period or comply with the existing deterministic analysis requirement of section 72.102(f), or, alternatively, use a return period significantly greater than 2,000 years, in that:

1. The requested exemption fails to conform to the SECY-98-126 (June 4, 1998) rulemaking plan scheme, i.e., only 1000-year and

[31] Joint Submittal of Unified Geotechnical Contention, Utah L and Utah QQ (Jan. 16, 2002) at 6-7.

10,000-year return periods are specified for design earthquakes for safety-important systems, structures, and components (SSCs) --- SSC Category 1 and SSC Category 2, respectively --- and any failure of an SSC that exceeds the radiological requirements of 10 C.F.R. § 72.104(a) must be designed for SSC Category 2, without any explanation regarding PFS SSC compliance with section 72.104(a).

2. PFS has failed to show that its facility design will provide adequate protection against exceeding the section 72.104(a) dose limits.

3. The Staff's reliance on the reduced radiological hazard of stand-alone ISFSIs as compared to commercial power reactors as justification for granting the PFS exemption is based on incorrect factual and technical assumptions about the PFS facility's mean annual probability of exceeding a safe shutdown earthquake (SSE), and the relationship between the median and mean probabilities for exceeding an SSE for central and eastern United States commercial power reactors and the median and mean probabilities for exceeding an SSE for the PFS facility.

4. In supporting the grant of the exemption based on 2,000-year return period, the NRCs Staff relies upon the United States Department of Energy (DOE) standard, DOE-STD-1020-94, and specifically the category-3 facility SSC performance standard that has such a return period, notwithstanding the fact the NRC Staff categorically did not adopt the four-tiered DOE category scheme as part of the Part 72 rulemaking plan.

5. In supporting the grant of the exemption based on the 2,000-year return period, the NRC Staff relies upon the 1998 exemption granted to DOE for the Idaho National Engineering and Environmental Laboratory (INEEL) ISFSI for the Three Mile Island, Unit 2 (TMI-2) facility fuel, which was discussed in SECY-98-071 (Apr. 8, 1998), even though that grant was based on circumstances

not present with the PFS ISFSI, including (a) existing INEEL design standards for a higher risk facility at the ISFSI host site; and (b) the use of a peak design basis horizontal acceleration of 0.36 g that was higher than the 2,000-year return period value of 0.30 g.

6. Because (a) design levels for new Utah building construction and highway bridges are more stringent; and (b) the PFS return period is based on the twenty-year initial licensing period rather than the proposed thirty- to forty-year operating period, the 2,000-year return period for the PFS facility does not ensure an adequate level of conservatism.

Q72. Which of these bases will you be addressing in your testimony?

A72. I have already discussed the predicate for item 1, in that my testimony shows that the existing design, based on a 2000-year return period earthquake, provides adequate protection against component failure that would risk exceeding regulatory dose limits. I will also address the remaining basis for the State's Contention, except for basis 2.

Q73. Please describe your understanding of the State's Basis 1.

A73. In Basis 1, the State challenges the exemption granted by the NRC Staff to PFS authorizing the use of a 2,000-year return period DBE on the grounds that such an exemption fails to conform to the rulemaking plan set forth in SECY-98-126 (June 4, 1998). That SECY discussed three different rulemaking options for the Commission for incorporating PSHA methods into 10 C.F.R. Part 72 with one of the three being identified as the "preferred" option.

Q74. Please describe the preferred methodological approach set forth in SECY-98-126?

A74. The preferred approach set forth in SECY-98-126 proposed a 1000-year mean return period design basis earthquake for Category 1 SSCs and a 10,000-year mean return period design basis earthquake for Category 2 SCCs, with SCCs whose failure would result in radiological doses ex-

ceeding the requirements of 10 C.F.R. § 72.104(a) being designated Category 2 SCCs.

Q75. Is this two-tiered DBE approach still the Commission's preferred methodology for the rulemaking plan to amend 10 C.F.R., Part 72 to incorporate PSHA methods?

A75. No. In SECY-01-0178, dated September 26, 2001, the NRC Staff recommended to the Commission that the rulemaking plan be modified to add a fourth option. This fourth option eliminated the two-tiered DBE approach for ISFSI SSCs and proposed the use a single 2,000-year mean return period earthquake as the design basis for all ISFSI SSCs. This is the same DBE as that provided for by the proposed exemption for the PFSF. SECY-01-0178, identified this fourth option that would provide for the use of a single 2,000-year mean return period earthquake as the "preferred" option. In a Staff Requirements Memorandum dated November 19, 2001, the Commission approved the modification to the rulemaking plan proposed by SECY-01-0178, further instructing the NRC Staff that the proposed rule should solicit comment on a range of exceedance levels from 5.0E-04 through 1.0E-04.

Q76. Does the PFS proposed exemption conform to this newly identified "preferred" option of the NRC rulemaking plan for amending 10 C.F.R. Part 72 to incorporate PSHA methods?

A76. Yes. It proposes a single DBE for all PFSF SSCs with a mean return period of 2,000 years identical to the preferred option identified in SECY-01-0178.

Q77. Where does that leave the State's Basis 1

A77. I believe that the NRC Staff's action and its approval by the Commission render Basis 1 obsolete.

Q78. Please describe your understanding of the State's Basis 3.

A78. In Basis 3, the State challenges the exemption on the grounds that the NRC Staff's reliance on the lower radiological hazard posed by stand-alone ISFSIs (as compared to commercial power reactors) is based on "incorrect factual and technical assumptions." The alleged incorrect factual

and technical assumptions apparently allude to the State's assertion that, per Regulatory Guide 1.165, nuclear power plant "design ground motions would have to correspond to a median annual probability of exceedance of 10^{-5}", and that for sites in the western U.S. a median of 10^{-5} is not equivalent to a mean of 10^{-4} as generally stated by the NRC Staff in its approval of PFS's exemption request.[32]

Q79. Is the State's assertion that nuclear power plant "design ground motions would have to correspond to a median annual probability of exceedance of 10^{-5}" accurate?

A79. No, the assertion is incorrect. First, Regulatory Guide 1.165, as the title of this series of NRC documents implies, only provides general guidance to applicants as to procedures that the NRC Staff would deem acceptable for satisfying the NRC's new probabilistic seismic criteria in 10 C.F.R. § 100.23. Regulatory Guide 1.165 (Ref. 4 at page 1) specifically provides that "Appendix B describes the procedure used to determine the reference probability for the SSE exceedance level that is acceptable to the Staff." Second, although the Guide does state that the annual probability level of the SSE may be based on a *median estimate of 10^{-5}* [Ref. 4 (Reg. Guide 1.165, Appendix B, p. 1.165-12)], this provision of the Guide is, in my opinion, primarily the result of historical circumstances. There was a significant discrepancy between the two assessments of the mean estimates made by the two major Central & Eastern U.S. ("CEUS") seismic hazard studies available at the time of the Guide's preparation. While the two studies differed with respect to the mean estimates, both studies provided similar median estimates. Therefore, the median estimate was adopted for the purposes of establishing in Regulatory Guide 1.165 an acceptable quantitative basis for satisfying 10 C.F.R. § 100.23. This discrepancy between the two studies has, however, since been largely resolved[33] and it

[32] [Ref. 25 (State's Request for Admission of Late-Filed Modification to Basis 2 of Contention Utah L, pp. 8- 11)].

[33] This history is recounted in Ref. 33 (T.C. Hanks, Imperfect Science: Uncertainty, Diversity, and Experts, EOS, Transactions, American Geophysical Union, Vol. 78, No. 35, Sept. 2, 1997, 369, 373, 377). The author concludes: "When LLNL used elicitation techniques more in line

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has been clearly established that the typical SSE at existing plants across the country has a mean annual probability of exceedance of approximately 10^{-4}.

Q80. Where is it documented that the mean annual probability of exceeding the SSE at existing nuclear power plants is approximately 10^{-4}?

A80. That the mean annual probability of exceeding the SSE at existing nuclear power plants in the CEUS sites is on the average about 10^{-4} is demonstrated in DOE-STD –1020-94 at p. C-17 [Ref. 11], in NUREG/CR-6728 at p. 7-14 [Ref. 20], and in DOE Topical Report for Yucca Mountain TR-003 at App. C [Ref. 26]. A set of the relatively recent CEUS sites were those used in the preparation of Regulatory Guide 1.165. See Ref. 26 at pg 12. It has also been demonstrated more recently in the DOE Topical Report II TR-003 at App. C [Ref. 26, also identified as PFS Exhibit FFF.] that this same number is also approximately representative of Western US nuclear power plant sites for which the average mean annual probability of exceeding the SSE is about 2×10^{-4}, or 5,000 years.

Q81. Please explain the significance of the fact that it has been clearly established, since the issuance of Regulatory Guide 1.165, that the typical SSE at existing plants across the country has a mean annual probability of exceedance of approximately 10^{-4}?

A81. The significance is that it can now be shown that, for nuclear power plants, there is a uniform DBE MAPE throughout the United States, as measured by the consistent use of the 10^{-4} number. The mean estimate is commonly preferred to the median estimate when making decisions based on uncertain annual probabilities or frequencies. It is preferred (1) because it is sensitive to that uncertainty, usually leading to more conservative estimates when the uncertainty is greater, and (2) because the mean estimate is consistent with formal decision theory which concludes that mean risks should be the basis for decisions in the face of uncertainty; the

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with the EPRI approach, the resulting answers were, within the likely uncertainties of either study, the same." Ref. 33 at 373.

mean accident risk of a facility is in turn proportional to the mean (not the median) estimate of the (uncertain) probability of that accident.

Q82. What estimate, the mean or the median, does the Commission typically use when estimating probabilities?

A82. When faced with uncertain probability estimates, the Commission has generally chosen to use the mean probability estimate. For example, the Commission's "Safety Goals for Operations of Nuclear Power Plants; Safety Policy Statement" states: "The Commission has adopted the use of the mean estimates for purposes of implementing the quantitative objectives of this safety goal policy (i.e., the mortality risk objectives)." [Ref. 30 (51 Fed. Reg. 28,044, 28,046 (1996)]. The NRC's choice of the mean estimate for all such risk objectives, including the subsidiary core melt damage frequency, is discussed in Regulatory Guide 1.174, "An Approach for Using PRA in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", at p. 14 [Ref. 5] and in SECY-00-0077, "Modifications to the Reactor Safety Goal Policy Statement" at p. 6 [Ref. 22]. Thus, in accordance with common practice, the Commission has clearly stated its general preference for the use of mean estimates as opposed to median estimates.

Q83. Based on the above, what conclusion do you draw regarding Basis 3?

A83. I conclude that 1×10^{-4} per annum, which has been found to be the mean estimate of the annual probability of exceedance of the design basis earthquake (DBE) of the typical nuclear power plant in all regions of the United States, is the appropriate basis from which to establish, via the principles of the risk-graded philosophy adopted by the Commission, the mean annual probability of exceedance of the DBE of an ISFSI anywhere in the country, including specifically at the PFSF site. This conclusion is independent of how or why the NRC Staff established the acceptable procedure highlighted in Regulatory Guide 1.165. It should be noted that both the original 10 C.F.R. part 72 rulemaking plan (SECY-98-126) and the modified plan (SECY-01-0178) approved by the Commission call for the use of mean probability estimates. Finally, as the State correctly points out, ratio between mean and median estimates of the probabilities is

not the same at typical CEUS sites as it is at most WUS sites. If risk-graded DBE decisions were based on median estimates, the result would be non-uniform mean probabilities of seismic accidents at nuclear power plants across the country.

Q84. Please describe your understanding of Basis 4.

A84. In Basis 4, the State challenges the exemption granted to PFS on the grounds that the NRC Staff inappropriately relied on DOE-STD-1020-94 (or DOE-STD-1020), which also provided for a 2000 MRP earthquake for ISFSIs, because the NRC Staff did not adopt this Standard in SECY-98-126.

Q85. What is the significance of DOE-STD-1020-94?

A85. As discussed above, DOE-STD-1020-94 is illustrative of the risk-graded approach toward seismic analyses. DOE-STD-1020-94 has been carefully prepared, with the support of recognized experts in the field, by a major federal agency that has experience with a broad spectrum of nuclear facilities, has authority to set standards, and has responsibility for public safety. The document is considered a model of explicit, graded, risk-consistent seismic criteria. Further, DOE-STD-1020-94 also clearly illustrates the general principle, embodied in using a risk-graded approach, that the probability of failure depends on both the DBE MRP and the level of conservatism in design procedures and criteria. It does so by establishing performance goals for acceptable seismic performance that are expressly the product of the DBE MRPs and the level of conservatism in design procedures and criteria, formally referred to in DOE-STD-1020-94 parlance as the risk reduction factor, R_R. It was for these reason that I used DOE-STD-1020-94 above to illustrate the application of a risk graded approach.

Q86. Has DOE-STD-1020-94 been updated?

A86. Yes, the DBE for category PC3 structures (the category in which ISFSIs would fall were they DOE facilities) has recently been changed from 2,000 years to 2,500 years.

Q87. Does this affect your opinion of whether DOE-STD-1020-94 is relevant to and supports the NRC Staff's approval of the PFS exemption?

A87. No. As just stated, under DOE-STD-1020-94 acceptable seismic performance is the product of the DBE MRPs and the level of conservatism in design procedures and criteria. While the DBE MRP for PC3 structures was increased to 2,500 years, the level of conservatism in the applicable design procedures and criteria was reduced such that the performance goal for PC3 structures remains unchanged at $1x10^{-4}$. The State's expert witness, Dr. Arabasz, has stated that he supports the use of the DOE PC3 performance goal of 10^{-4} for the PFSF. Arabasz Dep. at 80-81.

Q88. Please describe your understanding of Basis 5.

A88. In Basis 5, the State challenges the grant of the PFSF exemption claiming that the NRC Staff's reliance on the 1998 exemption granted to DOE for the Idaho National Engineering and Environmental Laboratory ("INEEL") ISFSI for the Three Mile Island, Unit 2 ("TMI-2") facility fuel is misplaced because the grant of the exemption there was based on circumstances not present with the PFS ISFSI, including (a) existing INEEL design standards for a higher risk facility at the ISFSI host site; and (b) the use of a peak design basis horizontal acceleration of 0.36 g that was higher than the 2,000-year return period value of 0.30 g.

Q89. Assuming for the sake of the argument that these differences in circumstances between the PFSF and the INEEL ISFSI exist, would they affect the appropriateness of using the 2,000-year MRP earthquake as the DBE for the PFSF?

A89. No. As discussed above, application of well established risk-graded principles to the specific circumstances of the PFSF show that use of a 2,000 MRP DBE for the PFSF provides sufficient protection to the public health and safety in accordance with established Commission use of risk-informed principles in its regulatory functions. The fact that a similar conclusion was reached for the INEEL ISFSI corroborates the appropriateness of this conclusion.

Q90. Please describe your understanding of Basis 6.

A90. In Basis 6, the State claims that the 2,000-year mean return period for the PFS facility does not ensure an adequate level of conservatism because design ground motion levels for certain new Utah building construction

and highway bridges are more stringent. As set forth in the State's September 28, 2001 discovery response [Ref. 27] to Interrogatory No. 8, this conclusion was based on the observation that, for example, the International Building Code 2000 (or "IBC-2000") will, when in effect, require a MRP of approximately 2500 years for the DBE, which is greater than the 2,000-year MRP DBE proposed for PFS.

Q91. Does this difference in definition of the DBE imply a lower probability of failure if an SSC is designed to IBC-2000 codes?

A91. No. One should not draw the erroneous conclusion that the difference in the definition of the DBE implies a lower probability of failure for SSCs designed to IBC-2000 versus those, such as the PFSF, designed to the 2,000-year MRP and the NRC's SRP design procedures and criteria. As I stated previously, the level of safety achieved depends on *both* the DBE MRP and on the design procedures and criteria utilized. The State's witness, Dr. Arabasz, expressly agrees that one needs to consider both the level of DBE MRP and the level of conservatism in the design in determining unacceptable seismic response of a structure. Utah Joint Decl. ¶ 38. The design procedures and criteria of the IBC-2000 are much less conservative than those of the SRP. For example, as described by the State's witness,[34] a first step of the IBC-2000 design procedures and criteria is to multiply the DBE by two-thirds, which at the PFSF site would reduce the effective IBC-2000 DBE MRP from 2500 years to about 800 years. Only in the case of those "essential structures" that merit the IBC-2000 "importance factor" of 1.5 is this two-thirds reduction, in effect, recovered.

Moreover, even for those "essential structures" for which this reduction is in effect recovered, the model building codes' design procedures and acceptance criteria are significantly less conservative than those in the SRP. The IBC-2000 and UBC model building codes permit much more liberal

[34] State of Utah's Objections and Response to Applicant's Seventh Set of Formal Discovery Requests to Intervenor State of Utah (Sept. 28, 2001) at 18.

allowances for the benefits of post-elastic behavior than either DOE-STD-1020-94 PC-3 and PC-4 criteria, or the NRC SRPs. As shown in Table 1, the net effect of the UBC design and acceptance criteria, which are in net effect quite similar to those in IBC-2000 and to DOE-STD-1020-94 PC1 and PC2, is a risk reduction ratio R_R of only 2, versus a value of 10 for DOE-STD-1020-94 (PC-4) and typically 5 to 20 or more for the facilities designed to the NRC SRPs These differences represent a factor of 2.5 to 10 or more in increased conservatism (as measured by R_R) in the design procedures for nuclear facilities versus those in model building codes, even if the multiplier of two-thirds in the IBC-2000 is ignored.

Q92. What conclusion do you draw with respect to the State's claim in Basis 6 as it relates to the IBC-2000?

A92. Even though the use of IBC-2000 for essential or hazardous buildings will imply a DBE with a 25% larger MRP than that for the PFSF, the more conservative design procedures and criteria of the ISFSIs SRP will provide that the typical PFSF SSCs have a mean annual probability of failure several times (2 to 8 or more) lower than buildings designed to IBC-2000 standards. Moreover, all PFSF important-to-safety SSCs have a risk reduction factors sufficient to provide a probability of failure of 10^{-4} or lower, i.e., at least two times lower than essential facilities designed to the IBC-2000. In addition, as discussed above, a number of key important-to-safety SSCs in the PFSF have great robustness and/or fractional operating periods, which reduce their probabilities of failure even further

Q93. How does the PFSF design compare to the bridge codes cited by the State?

A93. With bridge codes, like Dr. Bartlett, the State's witness[35], it is my understanding that, the AASHTO (American Association of State Highway Transportation Officials) model bridge code is used almost universally in the U.S. and that the currently governing version requires only a 500-year return period DBE. Further, it is my understanding that they have structural design procedures and criteria similar in conservatism to those of

[35]Deposition of Steven F. Barlett (Nov. 2, 2001) at 75-76.

model building codes such as UBC and IBC-2000. Therefore, assuming that a 2,500-MRP DBE is used in place of the 500-year value for the design of certain essential bridges in Utah, my discussion of IBC-2000 standards is equally applicable to bridges. The design of the PFSF under a 2,000-year return period earthquake and NRC seismic SRP design criteria provides higher safety levels than those available in the design of these special Utah bridges.

Q94. The State also claims in Basis 6 that the 2,000-year mean return period for the PFS facility does not ensure an adequate level of conservatism because the return period was chosen based on the twenty-year initial licensing period rather than a potential thirty to forty-year operating period. Does the fact that the PFSF license may be extended for twenty years have any affect on the appropriate choice of a design basis earthquake?

A94. No. In virtually all areas of public safety hazards are measured in terms of frequency of occurrence (e.g., as measured in annual probabilities, in probabilities per 50-year period, or in per human lifetime units), and the same safety criteria are specified regardless of the length of the activity in question, the exposure time, the estimated facility life, or the licensing duration [Ref. 12 (Paté-Cornell paper)]. This is also the case with respect to the risk acceptance guidelines promulgated by the NRC where the subsidiary performance objectives are the risk metrics Core Damage Frequency (CDF) and Large Early Release Frequency (LERF). [Ref. 5 (Reg. Guide 1.174 at p. 10)] and [Ref. 22 (SECY-00-0077 at p. 6)], usually measured in per annum terms. The reasons for focusing on frequencies such as annual risks in making facility safety decisions include the fact that any facility providing a needed service will, at the end of its operating life, most likely be replaced by some other facility used for the same purposes with its own, similar risks. The spent fuel to be stored at the proposed PFSF is currently being stored in or near nuclear power plants, and after leaving the PFSF it will likely be stored at the proposed Yucca Mountain facility.

Q95. Does this conclude your testimony?

A95. Yes, it does.

ATTACHMENT A

DETERMINATION OF RISK REDUCTION FACTORS FOR SSCs AT FACILITIES DESIGNED USING NRC SEISMIC SRP STANDARDS

The objective of this Attachment is to show the analytical process used to determine quantitatively the degree of conservatism inherent in the design procedures and acceptance criteria found in both DOE Standard 1020 [Ref. 11 and Ref. 18] and the NRC SRPs [e.g., Ref. 2 (NUREG 0800)]. This level of conservatism is captured in the risk reduction factor or ratio R_R. By calculating the values of R_R resulting from DOE Standard 1020 and the NRC SRPs, the risk reduction factors implicit in the SRP design procedures and criteria can be compared to risk reduction factors expressly provided for in DOE-STD-1020. The precise calculated value of R_R depends on several technical parameters (defined below) whose values may vary from site to site and from SSC to SSC. Accordingly, one can produce only a representative range of R_R values for both the SRP and DOE-STD-1020. (As an example, Figure C-4 on page C-11 of DOE-STD-1020-94 [Ref. 11] shows the range of R_R values for SSCs designed to the criteria specified for category PC4 SSCs in DOE-STD-1020.)

The risk reduction ratio, R_R, is defined in NUREG/CR-6728 [Ref. 21 pp. 7-9] by the equation:

$$R_R = F_R^{K_H} \left(e^{x_p \beta}\right)^{K_H} e^{-\frac{1}{2}(K_H \beta)^2}$$

A different formulation of this same equation appears also in DOE-STD-1020-94 at page C-9. In this equation, the variables are as follows:

- K_H, a measure of the slope of the PSHA seismic hazard curve;
- β, a measure of the degree of uncertainty in the response and capacity of SSCs;

- F_R, a measure of the margin (achieved by the procedures and criteria) between the level of the DBE and a reference SSC capacity; and
- x_p, a measure of the margin between this reference capacity and the median value of the SSC capacity.

These variables are defined in more detail in both of the references cited above (DOE-STD-1020-94 at Appendix C.2 and NUREG/CR-6728 at Section 7.2).

For the purposes of this comparison, I will use for both the SRP and the DOE-STD-1020 R_R determinations a range of values for the hazard curve slope $K_H = 2.1$ to 3.3 [Ref. 21 (NUREG/CR-6728 at pg. 7-6)]. These values are representative of the relevant hazard interval (10^{-4} to 10^{-5}) for nuclear power plants at CEUS sites (DOE-STD-1020 at pg. C-8-9, and C-12)[36], and also of the relevant hazard interval (10^{-3} to 10^{-4}) for DOE PC3 (i.e., ISFSI) SSCs at the PFSF site (e.g., the K_H at the PFSF site for peak ground acceleration is 2.8, as I determined from [Ref. 28 (Revised Geomatrix Appendix F at Fig. 6-11)]. For simplicity, I use here a typical value[37] of $\beta = 0.4$. (The conclusions are quite insensitive to β as shown in DOE-STD-1020-94 [Ref. 11] at Figure C-4 on page C-11.) These values for K_H of 2.1 to 3.3 and for β of 0.4 are common to the calculations below of the R_R for both DOE-STD-1020 and the NRC SRP.

First, I consider the DOE-STD-1020 R_R standards. For these standards, the appropriate value of x_p is 1.28 and the appropriate value of F_R is 1.5 SF, both of which appear in DOE-STD-1020-94 at Eq. C-6, pg. C-9 [Ref. 11]. For PC4 the value of the "scale factor" SF is set at 1.25 (and for PC3 it is set[38] at 1.0) in order to achieve the desired risk reduction ratio R_R [Ref. 11 (DOE-STD-1020-94 at pg. 2-13)]. Substitution of the above values

[36] For clarity, if one uses this reference, it needs to be pointed out that the K_H range above corresponds precisely to the A_R range of 2 to 3 that will be found at this citation; A_R is an alternative hazard curve slope measure, DOE-STD-102-94 at pg. C-8 [Ref. 11].

[37] Ref. 21 (NUREG 6728) at pg. 7-15) cites an average value of 0.45.

for K_H, β, x_p, and F_R into the equation for R_R leads to a range of values of R_R from 8 to 17 for DOE-STD-1020 category PC4, as can be seen on Figure C-4 on page C-11 of DOE-STD-1020-94. The results of these and similar calculations were used in DOE-STD-1020 to confirm the conclusion that the DOE-STD-1020 design procedures and acceptance criteria set forth in Chapter 2 would achieve a value of R_R of about 10, as required to meet the PC4 performance goal. DOE-STD-1020-94 at p. C-12 [Ref. 11].

Unlike DOE-STD-1020, the NRC SRPs have not been "tuned" to give a particular R_R (or more precisely a representative value, such as 10 above, applicable to a range of sites). Accordingly, it has been necessary to depend on the numerous engineering evaluations of safety margins and "fragility curves" of SSCs designed to the SRP that have been conducted over the last 20 years in the course of research by the industry and NRC contractors, and on the seismic probabilistic risk assessments and seismic margins studies that have been undertaken at virtually all nuclear power plants in the US (via the NRC IPEEE program). These evaluations have been made by earthquake engineers familiar with nuclear power plant SSC designs prepared to the NRC SRP procedures and criteria, and with the actual behavior of such SSCs in earthquakes as observed in the field and tested in the lab. This experience is summarized in NUREG/CR-6728 [Ref. 21] at pg. 7-3 by the conclusion: "For nuclear power plant design the factor of safety has typically been 1.25 to 1.5." NUREG/CR-6728 at pg. 7-4 [Ref. 21]. This "factor of safety" is the variable F_R in the above equation. This factor is, however, coupled with a value of x_p of

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[38] As described in the body of my testimony, this number has been changed to 0.9 in DOE-STD-1020-2002.

2.33. NUREG/CR-6728 (at Ch. 7), which determines the definition of the reference capacity (referred to as a "HCLPF" or C_1) used in engineering evaluations of SRP conservatisms. This value of x_p is much more conservative than that used in DOE-STD-1020.

Using this value of x_p and this range of F_R values one finds (for the same β value and range of K_H values used for the DOE-STD-1020 calculations above) that the R_R for the NRC NPP SRP is in the range 8 to 32. Compared to the range of 8 to 17 calculated for DOE-1020, this result confirms that the DOE-STD-1020 PC4 standard does indeed only "approach" those of the NRC NPP SRP, as stated in DOE-STD-1020-94 at page C-5 [Ref. 11].

If one looks, not at the range of hazard curve slope values of 2.1 to 3.3 used for K_H in the above calculations, but rather at the specific value $K_H = 2.8$ associated with peak horizontal ground acceleration at the PFSF site, the range of NRC NPP SRP R_R values is 12 to 21. For the subset of SSCs sensitive to 1-second spectral accelerations, the ratios range from 8 to 12 based on the reduced slope of the hazard curve for this period. Revised Geomatrix Appendix F at Fig. 6-11 [Ref. 28].

For simplicity in the body of my testimony I have summarized such detailed results in the statement that "the R_R's for typical ~~components~~ SSCs designed to the NRC SRP are in the range 5 to 20 or greater".

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Reference 38: U.S. Nuclear Regulatory Commission, NUREG-1536, *Standard Review Plan for Dry Cask Storage Facilities*, January 1997.

Reference 39: U.S. Nuclear Regulatory Commission, Regulatory Guide1.61 *"Damping Values for Seismic Analysis for Nuclear Power Plants,"*1974

Reference 40: Department of Energy, DOE-STD-1021-93., *"Natural Phenomena Hazards Performance Categorization Guidelines for Structures, Systems and Components"*, July1993.

C. ALLIN CORNELL

EDUCATION:

Stanford University, Architecture	AB	1960
Stanford University, Civil Engineering (Structures)	MS	1961
Stanford University, Civil Engineering (Structures)	PhD	1964

Doctoral Thesis: "Stochastic Process Models in Structural Engineering"

PROFESSIONAL EMPLOYMENT:

Stanford University :	Acting Assistant Professor	1963-1964
Universidad Nacional Autonoma de Mexico :	Visiting Professor	Summer 1966
University of California, Berkeley :	Visiting Associate Professor	1970-1971
Basler and Hofmann, Zurich:	Research Engineer	Summer 1972
Laboratorio Nacional de Engenharia Civil, Lisbon:	Visiting Research Investigator	1974-1975
Massachusetts Institute of Technology:	Assistant Professor and Ford Post-Doctoral Fellow	1964-1966
	Assistant Professor	1966-1968
	Associate Professor	1968-1974
	Holder of Gilbert Winslow Career Development Chair	1971-1974
	Professor	1974-1983
Stanford University :	Visiting Professor	1981-1983
	Professor (Research) - Half-Time	1983-present
	Co-Director, Reliability of Marine Structures Program	1988-present
	Fellow, SU-USGS Institute on Earthquake Engineering and Seismology	1986-1996
Consulting Practice:	Part-Time	1965-1981
	Half-Time	1981-present
Cygna, Inc., San Francisco	Senior Vice President	1984-1985
C. Allin Cornell, Co.	President	1981-present

PROFESSIONAL ORGANIZATIONS AND COMMITTEES (Current and Former):

American Iron and Steel Institute:
- Advisory Committee on Load-Factor Building Design

American National Standards Institute:
- Building Loads Code Committee A58

American Society of Civil Engineers:
- Committee on Structural Safety
- Committee on Nuclear Power Plant Safety
- Committee on Bridge Safety

Committee on Offshore Structure Safety

Earthquake Engineering Research Institute:
Editorial Board: Earthquake Spectra, 1991-1993
Seismic Risk Committee
Planning Committee, 50th Anniversary Annual Meeting, 1998-99

Joint European Committee on Structural Safety

National Academy of Engineering (Elected 1981)

Phi Beta Kappa

Seismological Society of America:	Board of Directors,	1984-1987
	Vice-president	1985-1986
	President	1986-1987

Sigma Xi

Society of Risk Analysis:
Senior Advisory Board, 1991 P.S.A.M. Conference

JOURNAL EDITORIAL BOARDS:

Structural Safety; Risk Abstracts; Probabilistic Engineering Mechanics; Engineering Structures; Earthquake Spectra, Uncertainties in Structural Mechanics

GOVERNMENT COMMITTEES AND SERVICE:

NBS, Consultant	1967-1975
USGS, Advisory Committee to Seismicity and Risk Analysis Branch	1974
UNESCO, Working Group on Definition of Seismicity and Ground Motion	1974
USGS, Workshop on Earthquake Prediction and Engineering Hazards	1977
NAE/NRC-Marine Board	
Committee on Offshore Technology	1979-1981
Committee on Marine Structures,	
Loads Advisory Group	1986-1987
Parent Committee	1987-1989
NAS Committee on Seismology	1981-1984
Panel on Science of Earthquakes	1996-2001
NAS-Water Board	
Committee on Techniques for Estimating Probabilities of Extreme Floods	1986-1988
NAE/NRC-Geotechnical Board - Comm. for Workshop on Reliability Methods for Risk Mitigation in Geotechnical Engineering	1992-1994

NRC Seismic PRA Seminar Technical Coordinator	1982
OECD-CSNI Specialist Meetings: Probabilistic Methods in SRA for NPP's	
Chairman	1980
Technical Organizing Committee	1983
NATO, Advanced Study Institute, Reliability of Structures and Soils, Lecturer, (Seismic Safety of NPP's)	1982

AWARDS RECEIVED:

Huber Research Prize, American Society of Civil Engineers	1971
Guggenheim Fellowship	1974-1975
Fulbright-Hayes Advanced Research Grant	1974-1975
Moisseiff Award, American Society of Civil Engineers	1977
Norman Medal, American Society of Civil Engineers	1983
(First) ICASP Award, Committee of Inter. Conference on Applications of Statistics and Probability in Soils and Structures	1987
Fruedenthal Medal, American Society of Civil Engineers	1988
Offshore Technology Research Center Honors Lecture, OTC	1995
EERI Distinguished Lecturer	1999
EERI Outstanding Paper of 1998 (Earthquake Spectra) (Co-authors: Shome, Bazzurro, and Carballo)	2000

SOME REPRESENTATIVE RECENT SPONSORED UNIVERSITY RESEARCH CONTRACTS:

SPONSOR:

NSF Stochastic Models of Structural Loads.
Spatial and Temporal Memory in Earthquake Recurrence and Hazard.
Nonlinear Seismic Assessment Procedures for Buildings
Probabilistic Prediction of Near-Source Strong Ground Motion and Nonlinear Structural Response

PEER (NSF Earthquake Engineering Center): Technical Foundation for Performance-Based Design

SAC Nonlinear Seismic Demands in Fracturing Steel Moment-Resisting Frames

ONR Reliability Analysis of Moored Marine Structures.

EPRI Multi-site Wind Record Analysis for Transmission Lines Structural Loads.

Effectiveness of Strong Ground Motions.

MMS Probability-Based Design Procedures for Offshore Structures

NRC Hazard-Consistent Nonlinear Analysis of Structures and Soils

JOINT INDUSTRY PROJECT
(36 company consortium, managed by Amoco Production Company)
Structural Systems Reliability Analysis for Offshore Structures.

INDUSTRIAL AFFILIATES PROGRAM 1986-present
Reliability of Marine Structures.

[resumes\largeparts\log.vitae\04\00]

REPRESENTATIVE CONSULTING PROJECTS

1999

Seismic Studies (Seismic Hazard Analysis;
Seismic Probability Risk Assessment;
Seismic Margins; Criteria Development;
Policy Advising, etc.):
DOE/Woodward-Clyde (Yucca Mountain PSHA Peer Review)
NRC/REI (Ground Motions Procedures Peer Review Panel)
SAC (Reliability-based Building Assessment Guidelines)
BP Amoco/EQE (ISO Offshore Seismic Guidelines)
Westinghouse (Savannah River Seismic Review)
NRC/ICF (Advisory Committee: New Dry Storage Cask Guidelines)

Offshore Platform Structures/Marine Reliability
REI/JIP (Riser Reliability)
E&P Forum JIP/REI (Low Probability Storm Assessment)
ABS (M.O.B.: Probability-based Design Procedures)
BP-Amoco (Prob. Asses. Of Extreme Ice Effects)

Other
DOE/Geomatrix (Design Decision Process: Yucca Mtn.)
BC Hydro (Dam Safety Guidelines; review)
WES/Ben Gerwick (Dam PRA Methodology)

1998

Seismic Studies (Seismic Hazard Analysis;
Seismic Probability Risk Assessment;
Seismic Margins; Criteria Development;
Policy Advising, etc.):
DOE/Woodward-Clyde (Yucca Mountain PSHA Peer Review)
DOE/Geomatrix (Yucca Mountain Volcano Hazard Analysis)
NRC/REI (Ground Motions Procedures Peer Review Panel)
B.C. Hydro (Keenleyside Dam Seismic Risk, Peer Review Panel)
Bechtel (Hanford Vitrification Plant PSHA)

Offshore Platform Structures/Marine Reliabiity
REI/JIP (Riser Reliability)
ABS (Risk-Based Ship Criteria)
Mobil (Seismic Design Frequency)
E&P Forum JIP/REI (Low Probability Storm Assessment)
EPR (Reliability Tutorial)
ABS (M.O.B.: Probability-based Design Procedures)

Other
DOE/Geomatrix (Design Decision Process: Yucca Mtn.)
BC Hydro (Dam Safety Guidelines; review)

1997

Seismic Studies (Seismic Hazard Analysis;
Seismic Probability Risk Assessment;
Seismic Margins; Criteria Development;
Policy Advising, etc.):
DOE/Woodward-Clyde (Yucca Mountain PSHA Peer Review)
DOE/Geomatrix (Yucca Mountain Volcano Hazard Analysis)
NRC/REI (Ground Motions Procedures Peer Review Panel)
NRC/Waterways Exper.Sta. (Probabilistic Liquefaction Analysis)
B.C. Hydro (Keenleyside Dam Seismic Risk, Peer Review Panel)

DOE/Pacific Engineering (Probabilistic Soil Amplification; Savannah River Site)
Cal. Earthquake Authority (Expert testimony)

Offshore Platform Structures/Marine Reliabiity

Amoco (Offshore Reliability)
REI/JIP (Riser Reliability)
ABS (Risk-Based Ship Criteria)
Bechtel (M.O.B.: Extreme Environment Characterization; Reliability)
ABS (M.O.B.: Probability-based Design Procedures)
Exxon Production Research (Seismic Criteria)

Other

EPRI/Sargent and Lundy (Temporary Loads Reliablity)
BC Hydro (Dam Safety Guidelines; review)

1996

Seismic Studies (Seismic Hazard Analysis;
Seismic Probability Risk Assessment;
Seismic Margins; Criteria Development;
Policy Advising, etc.):

USGS/DOE (Review of U.S. Hazard Maps)
DOE/Woodward-Clyde (Yucca Mountain PSHA Peer Review)
DOE/Geomatrix (Yucca Mountain Volcano Hazard Analysis)
NRC/REI (Ground Motions Procedures Peer Review Panel)
NRC/Waterways Exper.Sta. (Probabilistic Liquefaction Analysis)
Warburg Pincus (Seismic Insurance Risk Methods)
Aon Insurance Services (Seismic Insurance Risk Analysis)
Seattle Seahawks (King Dome Seismic Review)
B.C. Hydro (Keenleyside Dam Seismic Risk, Peer Review Panel)

Offshore Platform Structures/Marine Reliabiity

Chevron (Hurricanes)
Amoco (Offshore Reliability)
REI/JIP (Riser Reliability)
Shell/PMB (Maui A and B Seismic Reliability)
ABS (Risk-Based Ship Criteria)

1995

Seismic Studies (Seismic Hazard Analysis;
Seismic Probability Risk Assessment;
Seismic Margins; Criteria Development;
Policy Advising, etc.):

DOE/Woodward-Clyde (Yucca Mountain PSHA Peer Review)
DOE/Geomatrix (Yucca Mountain Volcano Hazard Analysis)
DOE/BNL (Short-term Seismic Exposure)
MMS/LLNL (Santa Barbara Channel PSHA)
B.C. Hydro (Seismic Risk Methods)
NRC/LLNL (Seismic Source Characterization)
EQE (Review of Cal. Eq. Auth. Analysis)
USGS/ATC (Paper/Workshops on PSHA)

Offshore Structures Reliability

Exxon Pro.Res. (Seismic Hazard and Response: Caspian Sea/Sakalin Island)
Mobil (Seismic Hazard and Response: Holly Platform)
PMB/JIP (Hurricane Andrew Bayesian Update of
Structural Loads and Capacities II)

1994

Seismic Studies (Seismic Hazard Analysis;
Seismic Probability Risk Assessment;
Seismic Margins; Criteria Development;

Policy Advising, etc.):
DOE/LLNL (Senior Hazard Advisory Committee; site hazard revisions)
DOE/BNL (Tanks Seismic Expert Panel; site reviews)
NRC/LLNL (Appendix B Revision; expert committee)
DOE/High-Level Waste Review Board
Commonwealth Edison Co. (Short-Term Criteria)
Woodward-Clyde (Hazard Methodology Update)
SRI/EDF (France) (SPRA Methodology)
Westinghouse Hanford (Safety Class Definition)
REI/DOE (SHA review)
Guy Carpenter Inc. (Loss estimation review)
ISEC/Golden Gate Bridge Retrofit
Offshore Structures Reliability:
PMB/JIP (Hurricane Andrew Bayesian Update of Foundation Capabilities)
PMB/JIP (Hurricane Andrew Bayesian Update of Structural Loads and Capacities II)
REI/JIP (Reliability Software Development Advice)
Chevron (Hurricane Statistics)
Exxon Production Research (Response Analysis)
Statoil (Failure Probability Bases}

1993 *Seismic Studies (Seismic Hazard Analysis;*
Seismic Probability Risk Assessment;
Seismic Margins; Criteria Development;
Policy Advising, etc.):
DOE/LLNL (Senior Seismic Hazard Advisory Committee; site reviews)
DOE/BNL (Tanks Seismic Expert Panel)
Woodward-Clyde (SHA)
San Diego Gas & Electric (SHA Review)
EPRI (Max. Magnitude Project)
NRC/CNWRA (HLW Seismic Criteria)
ISEC/Golden Gate
REI/NRC(Seismic Motions/PRA)
EPRI (Max. Magnitude Project)
Ofshore Structures Reliability:
PMB/JIP (Hurricane Andrew Bayesian Update of Structural Loads and Capacities)
Unocal (Seismic safety review; SHA reviews)
Chevron (Extreme Wave Reliability-Methodology)
Statoil (Norway) (North Sea SHA review)
PMB/JIP (Dynamic Capacity)

1992 *Seismic Studies (Seismic Hazard Analysis;*
Seismic Probability Risk Assessment;
Seismic Margins; Criteria Development;
Policy Advising, etc.):
DOE/LLNL (NPR Senior Advisory Committee; Interim Criteria, site reviews)
DOE/BNL (TSEP; site SHA reviews)
NRC/LLNL (Appendix B Revision, expert panel)
EPRI (Maximum Magnitude Project)
Geomatrix (CalTrans SHA reviews)
Woodward-Clyde (CalTrans SHA reviews)

Portland General Electric (Senior Seismic Panel)
ISEC/Golden Gate
REI/NRC (Seismic Motions/PRA)
ESA (Aqueduct Analysis)
REI/NSF (Loma Prieta Motions Analysis)

Offshore Structures Reliability:
Unocal (SHA review; SHA and criteria)
REI (TLP-LRFD JIP)
PMB/USN
PMB/JIP (Dynamic Capacity)
PMB/JIP (Andrew Bayesian Update)
Chevron (Reliability Methodology)
API (Seismic Requalification Criteria)

1991

Seismic Studies (Seismic Hazard Analysis; Seismic Probability Risk Assessment; Seismic Margins; Criteria Development; Policy Advising, etc.):
DOE/LLNL (Natural Hazards; NPR Senior Advisory Committee; Interim Criteria, site reviews)
BC Hydro (Seismic Hazard Committee)
Portland General Electric (Senior Seismic Panel)
EPRI (Maximum Magnitude Project)
NRC
REI/CGMG (Seismic Motion Analysis)
REI/NRC (Seismic Motions/PRA)

Offshore Structures Reliability:
PMB/USN (Underwater Array Reliability)
EPR (Seismic Review)
API (Seismic Requalification Criteria)

Other:
Paul, Hastings, Janofsky and Wal (Fiber Pipe Reliability)

1990

Seismic Studies (Seismic Hazard Analysis; Seismic Probability Risk Assessment; Seismic Margins; Criteria Development; Policy Advising, etc.):
DOE/LLNL/BNL (NPR Senior Advisory Committee; Interim Criteria; Site Reviews; High-Level Waste Tanks)
EPRI/NUMARC/IPEEE
Exxon Production Research (Reliability)
USGS/NEPEC (Bay Area Seismic Hazard)
NRC/ACNW
Portland General Electric
Woodward-Clyde Consultants

Offshore Structures Reliability:
Exxon Production Research (EPR) (reliability software)
PMB/NCEL
ELF Aquitaine (France)/LRFD Development

Other:
NASA/Veritas Research (Structural Reliability)

1989

Seismic Studies (Seismic Hazard Analysis;
Seismic Probability Risk Assessment;
Seismic Margins; Criteria Development;
Policy Advising, etc.):
DOE/LLNL (Senior Review Group: External Events Assessment and Criteria; NPR Criteria)
Pacific Gas and Electric
Portland General Electric
Electric Power Research Institute (Severe Accident Policy, Seismic Hazard, High Frequency Ground Motion Effects)
Nuclear Regulatory Commission/ANL
Woodward Clyde Consultants
Risk Engineering, Inc.
Geomatrix
Offshore Structures Reliability:
Joint Industry Project (12 sponsors); Full-scope Reliability ("MCAPS"); Amoco Production Co., Manager.
ELF Aquitaine (France)
Exxon Production Research
Statoil (Norway)

1988

Seismic Studies (Seismic Hazard Analysis;
Seismic Probability Risk Assessment;
Seismic Margins; Criteria Development;
Policy Advising, etc.):
Pacific Gas and Electric Co. (Diablo Canyon Long-Term Seismic Program, Advisory Board and Consultant)
Electric Power Research Institute (Senior Advisory Group: Eastern U.S. Seismic Hazards Project)
Risk Engineering, Inc.
U.S. Nuclear Regulatory Commission/ANL
Portland General Electric (Senior Seismic Panel)
Bechtel Corporation
Canada Oil and Gas Administration
Statoil (Norway)
Offshore Structures Reliability:
Joint Industry Project (36 sponsors); Structural Systems Reliability; Amoco Production Co., Manager
Joint Industry Project (12 sponsors): Full-Scope Systems Reliability ("MCAPS"); Amoco Production Co., Manager
ELF Aquitaine (France)
Amoco Production Co.
Exxon Production Research
Bridge Loadings:

NCHRP (Jointly with Imbsen and Associates, Inc.)

1987

Seismic Studies (Seismic Hazard Analysis;
Seismic Probability Risk Assessment;
Seismic Margins; Criteria Development;
Policy Advising, etc.):
Pacific Gas and Electric Co. (Diablo Canyon Long-Term Seismic Program, Advisory Board and Consultant)
Electric Power Research Institute (Senior Advisory Group: Eastern U.S. Seismic Hazards Project)
(Non-Poissonian Earthquake Recurrence Analysis Project)
U.S. Nuclear Regulatory Commission
Geomatrix
Offshore Structural Reliability:
Joint Industry Project (36 sponsors); Systems Reliability; Amoco Production Co., Manager
Joint Industry Project (12 sponsors): Full-Scope Systems Reliability ("MCAPS"); Amoco Production Co, Manager
ELF Aquitaine (France)
Site-Specific Bridge Loads:
NCHRP (Jointly with Imbsen and Associates, Inc.)

1986

Seismic Studies (Seismic Hazard Analysis;
Seismic Probability Risk Assessment;
Seismic Margins; Criteria Development;
Policy Advising, etc.):
Pacific Gas and Electric Co. (Diablo Canyon Long-Term Seismic Program, Advisory Board and Consultant)
Electric Power Research Institute (Senior Advisory Group: Eastern U.S. Seismic Hazards Project)
Woodward-Clyde
Impell
Bechtel Corp.
Yankee Atomic Electric Co.
U.S. Nuclear Regulatory Commission
Offshore Structures Reliability:
Joint Industry Project (36 sponsors); Systems Reliability; Amoco Production Co., Manager
Joint Industry Project (12 sponsors): Full-Scope Systems Reliability ("MCAPS"); Amoco Production Co., Manager
ELF Aquitaine (France)
Amoco Production Co.

1985

Seismic Studies (Seismic Hazard Analysis;
Seismic Probability Risk Assessment;
Seismic Margins; Criteria Development;
Policy Advising, etc.):

Pacific Gas and Electric Co. (Diablo Canyon Long-Term Seismic Program, Advisory Board and Consultant)
Electric Power Research Institute (Senior Advisory Group: Eastern U.S. Seismic Hazards Project)
(Non-Poissonian Earthquake Recurrence Analysis Project)
Maine Yankee Power Co.
Yankee Atomic Electric Co.
U.S. Nuclear Regulatory Commission (Design Margins and SPRA Validation Senior Advisory Committees)
Bechtel Corp.
Sandia (Long-Term Nuclear Waste Disposal)
Electricite de France

Structural Systems Reliability:
G.A. Technologies (through DOE) (HTGR Probability-Based Design Criteria Advisory Board)

Offshore Structures Reliability:
ELF Aquitaine (France)
Joint Industry Project (36 sponsors); Structural Systems Reliability; Amoco Production Co., Manager
Joint Industry Project (12 sponsors): Full-Scope Systems Reliability ("MCAPS"); Amoco Production Co., Manager

Statistical Analysis of Construction Quality Sampling:
Anolik et al (Shelter Ridge Condominiums)
Fairfield et al (Hunters Point Housing Project)

1984 *Seismic Studies (Seismic Hazard Analysis; Seismic Probability Risk Assessment; Seismic Margins; Criteria Development; Policy Advising, etc.):*
Maine Yankee Power Co. (Maine Yankee)
Lawrence Livermore National Laboratory
Pacific Gas and Electric Co. (Diablo Canyon)
Yankee Atomic Electric Co. (Yankee Rowe, et al)
Niagara Power (through Dames and Moore)
NRC (Design Margins and SPRA Validation Senior Advisory Committees)
Dames and Moore (Millstone)
Electric Power Research Institute (Senior Advisory Group: Eastern U.S. Seismic Hazards Project)

Probabilistic Extreme Precipitation and Flood Analysis:
Yankee Atomic Electric Co.

Risk Analysis Tutorials, Short Courses, etc.:
Woodward-Clyde Consultants
ACTA, Inc.

Offshore Structures Design Criteria:
PMB Systems (SOHIO, Shell)

1983 *Seismic Studies (Seismic Hazard Analysis: Seismic Probability Risk Assessment; Seismic Margins; Criteria Development; Policy Advising, etc.):*
Maine Yankee Power Co. (Maine Yankee)

Lawrence Livermore National Laboratory
NRC, (ACRS)
Yankee Atomic Electric Company
Cygna, Inc.
Boston Edison (through Yankee Atomic Electric Co.)
Pickard, Lowe & Garrick, Inc. (Seabrooke)
Niagara Power (through MPR, and Dames and Moore)
Electric Power Research Institute (Research through Yankee Atomic Electric Co.)
Electric Power Research Institute (Eastern Seismic
Hazard Project Senior Advisory Committee)
Law Engineering and Testing Co. (Duke Power Co.)
Office of Naval Research
A. Anolik (Westborough Housing Study)

Structural Code Development:
Electric Power Research Institute/Col. State Univ. (Transmission Lines)
ACTA, Inc.

Probabilistic Extreme Precipitation and Flood Analysis:
Yankee Atomic Electric Co.

Risk Analysis Tutorials, Short Courses, etc.:
Woodward-Clyde Consultants (Probabilistic Methods)
ACTA, Inc. (Extreme Events)

Offshore Structures Design Criteria (Waves, Ice, System Reliability, etc.):
PMB Systems (SOHIO, Shell)

1982

Seismic Studies (NPP Sites):
Pickard, Lowe and Garrick (Zion, Indian Point, Seabrooke)
Yankee Atomic Electric Co. (Yankee Rowe)
Maine Yankee Power Co. (Maine Yankee)
Woodward-Clyde Consultants
Stone and Webster Corp. (Millstone)
Dames and Moore (Millstone)
Electric Power Research Institute (through Yankee
Atomic Electric Co.: Development of Historic SHA)
NRC, Lawrence Livermore National Laboratory

Pile Foundation System Reliability:
NUCLEN, (Brazil)

Structural Code Development:
Electric Power Research Institute/Colorado State Univ.
ACTA, Inc.

Load Combination Analysis:
Lawrence Livermore National Laboratory

Risk Analysis Tutorials, Short Courses, etc.:
NRC (through Sandia National Laboratory)
Woodward-Clyde Consultants

1980-81

Seismic Studies:
Pickard, Lowe and Garrick
Yankee Atomic Electric Power Co.
Lawrence Livermore National Laboratory

Pile Foundation System Safety:
NUCLEN, (Brazil)

Load Combination Analysis:
Lawrence Livermore National Labortory

1979-80 *Seismic Studies*:
Pickard, Lowe and Garrick
Weston Geophysical Research
Woodward-Clyde Consultants
Lawrence Livermore National Laboratory/NRC
Yankee Atomic Electric Co.
Air Pollution Hazard Study:
Pickard, Lowe and Garrick
Structural Safety Short Course:
Raytheon Co.
Load Combination Analysis:
G.E. Mark II Reactor Owners Group (through N.M. Newmark)

1978-79 *Seismic Studies*:
T.V.A.
Weston Geophysical Research
Southern California Edison Co.
Woodward-Clyde Consultants
Lawrence Livermore National Labortory/NRC
Load Combination Studies:
G.E. Mark II Reactor Owners Group (through N.M. Newmark)

1977-78 *Seismic Risk Analysis and Ground Motion Predictions*:
T.V.A.
Pacific Gas and Electric Co.
Woodward-Clyde Consultants
Seismic Reliability Studies of Nuclear Power Plant Systems:
Southern California Edison Co. Pacific Gas and Electric Co.
Pickard, Lowe and Garrick
Technical Chairman; one-week seminar for German Government (BAM)
Nuclear Regulatory Commission and Lawrence Livermore National Laboratory;
Senior Advisory Group: Seismic Safety Margins Research Project

1976-77 *Seismic Risk Analysis and Ground Motion Consultation*
Bell Laboratories
Pacific Gas and Electric Co.
Law Engineering
U.S. Army Corps of Engineers
Boston Edison Co.
Weston Geophysical Research, Inc.
Statistical Analysis of Fires:
NFPA

1975-76 *Probabilistic Systems Analysis; Dutch Oosterschelde Closure Project*:
T. W. Lambe and Associates
Seismic Risk Analysis and Ground Motion Consultation:
Nuclear Fuel Services
Dames and Moore

Weston Geophysical Research, Inc.
Boston Edison Co.
Basler and Hofmann
Advisory Committee on NFPA Project on Probabilistic Fire Safety Analysis

1974-75 *Seismic Risk Analysis Consultation*:
Dames and Moore
Weston Geophysical Research, Inc.
Aircraft Crash Risk Consultation:
Pickard and Lowe

1973-74 *Aircraft Crash Risk Studies for Nuclear Power Plants*
for PEPCO and Stone and Webster through Weston Geopysical Research, Inc. and others
Seismic Risk Analyses and Artificial Design Motions
for Several Engineering Projects
Assorted Hazard Study Reviews
for Pickard and Lowe
Refinement and Documentation of Seismic Risk Analysis Programs
for J. A. Blume and Associates
Wind-Loading Studies on Boston's John Hancock Building
for Hansen, Holley and Biggs
National Bureau of Standards Building Live Loads Survey
Report Preparation; and (through J. H. Wiggins and Company)
Survey Implementation Review

1972-73 *Through Weston Geophysical Research, Inc., American Electric Power; Stone and Webster; et al.*:
Design Response Spectra and Probabilistic Artificial Motions for Several
Nuclear Power Plant Projects
For Pickard and Lowe:
Wind-Induced Wave Risks on Great Lakes
Review of Seismic Risk Analysis for Dames and Moore
Consultation to NBS on Live Load Survey Implementation
Aircraft Crash Risk Analysis for Nuclear Power Plants
for Oregon Nuclear and Thermal Energy Council

1971-72 *Design of a Building Live Loads Survey*
for National Bureau of Standards
Through Weston Geophysical Research, Inc.:
a) Response Spectra and Seismic Design Criteria for Several Nuclear Power Plants
b) Development of Seismic Risk Map for American Electric Power
Retained as Seismic Consultant to Environmental Research, Inc., Las Vegas, Nevada
Through Hansen, Holley and Biggs:
Seismic Design Levels and Response Spectra for Drydock Sites on West Coast
for Crandall Drydocks, Inc.
Wind Dispersion Analysis
for Pickard and Lowe
Advisor to University of Mexico Earthquake Engineering Project
for UNESCO

1970-71 *Review of Fire Loads Survey Analysis for CEACM, Paris*

Through Weston Geophysical Research, Inc.: Seismic Design Criteria for several Nuclear Power Plants
Aircraft Crash Risk Analysis for Pickard and Lowe

PUBLICATIONS

Book:

Benjamin, J. R. and Cornell, C. A., *Probability, Statistics, and Decision for Civil Engineers*, McGraw-Hill Book Company, 1970.

Papers in Referred Journals:

Torres, G. G. B., Brotchie, J. R., and Cornell, C. A., "A Program for the Optimum Design of Prestressed Concrete Highway Bridges", *Journal of the Prestressed Concrete Institute*, Vol. 11, No. 3, June, 1966.

Reinschmidt, K. F., Cornell, C. A., and Brotchie, J. R., "Iterative Design and Structural Optimization", *Journal of the Structural Division*, ASCE, Vol. 92, No. ST6, December, 1966, pp. 281-318.

Sturman, G. M., Albertson, L. C., Cornell, C. A., and Roesset, J. M., "A Computer-Aided Bridge Design System", *Journal of the Structural Division*, ASCE, Vol. 92, No. ST6, December, 1966, pp. 141-165.

Cornell, C. A., "Bounds on the Reliability of Structural Systems", *Journal of the Structural Division*, ASCE, Vol. 93, No. ST1, February, 1967, pp. 171-200.

Ayer, F. and Cornell, C. A., "Grid Moment Maximization by Mathematical Programming", *Journal of the Structural Division*, ASCE, Vol. 94, No. ST2, February, 1968, pp. 529-549.

Cornell, C. A. and Vanmarcke, E. H., "Some Practical Implications of Elementary Safety Analysis", *Journal of the Boston Society of Civil Engineers*, Vol. 55, No. 3, July, 1968.

Cornell, C. A., "Engineering Seismic Risk Analysis", *Bulletin of the Seismological Society of America*, Vol. 58, No. 5, October, 1968, pp. 1583-1606.

Cornell, C. A., "A Probability-Based Structural Code", *Journal of the American Concrete Institute*, No. 12, Proc. Vol 66 December, 1969, pp. 974-985.

Corotis, R. B., Vanmarcke, E. H., and Cornell, C. A., "First Passage of Non-Stationary Random Processes", *Journal of the Engineering Mechanics Division*, ASCE, No. EM2, April, 1972, pp. 401-414.

Peir, J. C. and Cornell, C. A., "Spatial and Temporal Variability of Live Loads", *Journal of the Structural Division*, ASCE, Vol. 99, No. ST5, May, 1973, pp. 923-943.

McGuire, R. K. and Cornell, C. A., "Creep of Concrete Under Stochastic Live Load", *Journal of the Structural Division*, ASCE, Vol. 99, No. ST5, May, 1973, pp. 923-943.

Merz, H. A. and Cornell, C. A., "Seismic Risk Analysis Based on a Quadradic Magnitude-Frequency Law", *Bulletin of Seismological Society of America*, Vol. 63, No. 6, December, 1973, pp. 1999-2006.

McGuire, R. K. and Cornell, C. A., "Live Load Effects in Office Buildings", *Journal of the Structural Division*, ASCE, Vol. 100, No. ST7, July, 1974, pp. 1351-1366.

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[resumes\Basic large pcs.\all publications] (updated 02/13/01)

1 MR. GAUKLER: We have three exhibits
2 that go with Dr. Cornell's testimony. The first is
3 Exhibit PFS Exhibit DDD, which are excerpts from
4 DOE Standard 1020-94. The next one is Exhibit EEE,
5 which are excerpts from the deposition of
6 Dr. Walter Arabasz of October 31, 2001. And the
7 last exhibit is FFF, which are excerpts from
8 Topical Report YMP/TR-003-NP, Preclosure Seismic
9 Design Methodology for a Geologic Repository at
10 Yucca Mountain Division 2, August 1997.
11 I would move for the admission of these
12 exhibits into evidence.
13 JUDGE FARRAR: These are the same as
14 were with your prefiled testimony?
15 MR. GAUKLER: Yes.
16 JUDGE FARRAR: And were they previously
17 given to the reporter or today?
18 MR. GAUKLER: They were in the book that
19 we handed out at the beginning of the testimony.
20 JUDGE FARRAR: All right, then we'll
21 assume those have been marked by the court reporter
22 for identification. Is there any objection?
23 MS. NAKAHARA: No, objection, Your
24 Honor.
25 JUDGE FARRAR: Mr. Turk?

1 MR. TURK: No, Your Honor.

2 JUDGE FARRAR: All right. Three

3 exhibits, DDD, EEE and FFF will be admitted. We

4 will never again use that number.

5 MR. GAUKLER: When you get to three of

6 them, it does get hard.

7 (EXHIBITS-DDD, EEE & FFF MARKED AND ADMITTED.)

8 MR. GAUKLER: I now present Dr. Cornell

9 for cross-examination.

10 JUDGE FARRAR: Okay. Let me just ask

11 one question. You had on I think answers 13 and

12 14, a definition of the difference between the

13 deterministic and probabilistic approaches. Does

14 the -- when you're doing probabilistic, does the

15 capability of a fault under the old deterministic

16 rules come into play, or do you look at fault --

17 potential activity of a fault differently?

18 DR. CORNELL: Yes, the capability of the

19 fault is still important.

20 JUDGE FARRAR: The capability within --

21 as defined by the NRC regulations or as defined by

22 you all who specialize in this?

23 DR. CORNELL: I think probably more by

24 those who specialize in the field.

25 JUDGE FARRAR: Okay. At some point

1 during the hearing, we had raised a question about
2 that before. That would be helpful if sooner or
3 later some of the witnesses talk about that
4 difference.
5 Let me ask counsel a question. We have
6 to rule eventually on the legitimacy of the
7 exemption. Is that a factual question, a question
8 of expert opinion or a legal opinion -- legal
9 standard or a mixture of all three? What does the
10 State think?
11 MS. CHANCELLOR: It's basically a policy
12 decision as well as a risk-based analysis. And to
13 a certain extent, it is legal because you have to
14 meet the standard in 72.7 for exemptions. It's
15 not -- so I think it's a question of fact and law.
16 JUDGE FARRAR: Which is to be informed
17 by expert opinion?
18 MS. CHANCELLOR: That's correct.
19 MR. GAUKLER: We would agree with that.
20 We believe it's a mixture of fact involved.
21 Experts can certainly testify to the facts, also
22 can testify to the policy in terms of the
23 relationship between safety and margin and hazard.
24 Those are all things that an expert can testify to.
25 JUDGE FARRAR: All right, Mr. Turk?

1 MR. TURK: I think it's pretty well
2 covered. It sounds like it is fact law and policy.
3 The legal standard is 72.7, I believe, the
4 exemption standard, and ultimately you have to
5 reach a public health and safety finding.
6 JUDGE FARRAR: Just one last question of
7 the witness. When you do a probabilistic analysis
8 and come up with whatever conclusion you come up
9 with, do you design the facility any differently?
10 I don't mean whether it's a more stringent design
11 versus a less stringent, but do you put in
12 different kinds of things? I'm not talking about
13 the level of safety you're designing for, but do
14 you design the plant with using different
15 techniques because you used a probabilistic rather
16 than a deterministic?
17 DR. CORNELL: Generally, no. The
18 probabilistic assessment comes in to analyze a
19 given design, however it's been established.
20 JUDGE FARRAR: Right, but then you would
21 then rely on regular or trusted techniques to match
22 the level of design you had decided on?
23 DR. CORNELL: Yes, I think I understand
24 your question, and I believe the answer is
25 absolutely yes.

1 JUDGE FARRAR: And if at any point, you
2 don't understand our questions, you will not insult
3 us by reframing our questions and answering the
4 question we should have asked.
5 JUDGE LAM: Telling us how much we don't
6 know. Oh, I have a question if the Chairman is
7 finished. I had a question for Mr. Turk.
8 Mr. Turk, isn't it true ordinarily the
9 Staff has full authority to grant an exemption?
10 MR. TURK: If there was no adjudicatory
11 proceeding, yes, that's been delegated to the
12 Staff.
13 JUDGE LAM: Thank you.
14 MS. CHANCELLOR: I would just ask if
15 that's the case in this proceeding? Because the
16 Commission has to issue the license, and my
17 understanding was that the Staff was making a
18 recommendation to the Commission rather than
19 granting the exemption.
20 JUDGE FARRAR: In this case, it's
21 entirely different because of the Commission's
22 decision saying this was part of the licensing
23 process. In other words, as I read their decision,
24 because you all had a contention pending and this
25 went to the merits of the contention, that it's a

1 question for us to resolve in this hearing. And,
2 of course, our decision is also subject to
3 Commission review.
4 MS. CHANCELLOR: That wasn't quite what
5 I was getting at. Mr. Turk had scolded me in
6 filing documents when we had stated that the Staff
7 has approved the exemption, and Mr. Turk had always
8 made the distinction, no, we are making a
9 recommendation to the Commission. Just talking
10 about the Staff review side of it as opposed to the
11 adjudicatory side. So I was just surprised that
12 the Staff had actual authority under Part 72 for an
13 ISFSI to grant an exemption. Maybe I'm confused.
14 MR. TURK: Could I clarify?
15 JUDGE FARRAR: Yes. Let's take the case
16 where there was no litigation, that an Applicant
17 came into the Staff, there was no litigation
18 pending, no intervenors involved, how does the
19 system work?
20 MR. TURK: I'd like to answer that, but
21 I need to state that Ms. Chancellor is correct in
22 part and incorrect in part. There is a provision
23 in the regulations that states that no license
24 shall be issued for an ISFSI except by the
25 Commission. In other words, even if there was no

1 adjudicatory hearing, the Staff would issue its
2 SER, would reach a determination on whether an
3 exemption should be granted. The final review of
4 that would go to the Commission, whether or not
5 there was a Licensing Board proceeding.
6 JUDGE FARRAR: Which is not the case in
7 a reactor license staff. At least in the old days,
8 the Staff could issue a license, if I remember
9 correctly.
10 MR. TURK: I'd have to double-check to
11 be sure. But there is a specific requirement,
12 unlike other types of nuclear material facilities.
13 For an ISFSI, the Commission must make the final
14 issuance of the license. Now, so to that extent,
15 Ms. Chancellor is absolutely right. And I hope she
16 doesn't think I scolded her. If anything, I was
17 trying to make a distinction between a grant of the
18 exemption and the Staff's final position on that.
19 JUDGE FARRAR: No, she was, I think
20 using the term facetiously.
21 JUDGE LAM: But isn't it true Mr. Turk
22 in reactor proceedings, the Staff's authority to
23 grant an exemption is immune to challenges? Isn't
24 that true?
25 MR. TURK: It's been about four or five

1 years since I was involved with Part 50, but I
2 think you're correct. There's nothing that I can
3 recall in Part 50 that would prevent the Staff from
4 making its own determination on the exemption.
5 JUDGE LAM: Right.
6 JUDGE FARRAR: I think, of course, the
7 most recent word on all of this was, in fact, the
8 Commission's decision on this case -- in this case
9 tying exemption requests into pending proceedings
10 and so forth, which covered the area. Mr. Gaukler,
11 you were going to volunteer something.
12 MR. GAUKLER: All I was going to
13 volunteer is that for reactor operating licenses
14 for full power, the Commission has to approve the
15 issuance of the decision, as well. The Staff
16 doesn't have authority to issue the decision on
17 full power licenses for reactors above five
18 percent.
19 JUDGE FARRAR: But in any event, in this
20 case, the Staff has indicated that it would grant
21 the exemption. It's wide open for litigation in
22 front of us, and I assume that this part of this
23 case as well as every other part will be reviewed
24 by the Commission in due course.
25 All right, then, with that helpful

1 background and these preliminary questions,
2 Ms. Nakahara, you may proceed.
3 MS. NAKAHARA: I believe Mr. Turk goes
4 first.
5 JUDGE FARRAR: You're correct again. I
6 guess it's because I still think of in the posture
7 of this case, the cross-examination means the State
8 since the -- but we do call what the Staff does
9 cross, so you're right, and under our procedures,
10 they go first. Thank you. Go ahead, Mr. Turk.
11 MR. TURK: Thank you, Your Honor.
12
13 CROSS EXAMINATION
14 BY MR. TURK:
15 Q. Good morning, Dr. Cornell.
16 A. Good morning.
17 Q. My name is Sherwin Turk. I'm an
18 attorney with the NRC Staff. Seated with me
19 is Mr. Martin O'Neill who's also with the General
20 Counsel's office. We have other members of the
21 Staff and consultants with us today, as well. Some
22 of them I think you know.
23 I don't have very much in the way of
24 questioning for you today. Probably something on
25 the order of a half an hour. And I'm probably not

1 going to the key points of your testimony, but
2 instead I'm looking to clarify certain areas that I
3 either didn't quite understand or that I'd like a
4 little more explanation of.
5 And the first of those is with respect
6 to your Answer 9 on Page 6. You mention a DOE
7 standard, DOE Standard 1021-93. And I believe
8 that's also Reference 40 attached to your
9 testimony.
10 A. Yes.
11 Q. And that document is entitled Natural
12 Phenomena Hazards Performance Categorization
13 Guidelines for Structure, Systems and Components
14 dated July 1993?
15 A. Yes.
16 Q. Could you explain what that document is
17 and what its significance is here?
18 A. The purpose of that document is to help
19 the engineers decide which particular category
20 referred to as PC category one, two, three or four
21 in the DOE standards that a particular SSC, system,
22 structure or component, should be categorized as,
23 as a function of the implications of its failure,
24 the hazardous implications of its failure.
25 Q. Is that document utilized in any way in

1 the DOE Standard 1020-94 or 1020-2002?

2 A. Yes, I believe 1020-94 or 202 asks the

3 user to go to that document to give him guidance in

4 establishing which category to place a particular

5 SSC.

6 Q. And these are Department of Energy

7 documents that we've been describing, correct?

8 A. Correct.

9 Q. Those documents have not been adopted as

10 part of NRC regulatory practice, have they, to your

11 knowledge?

12 A. No, they have not.

13 Q. Not in a general sense?

14 A. No.

15 Q. Also if you would please go to Answer 31

16 on Page 60. I'm sorry, Answer 31 to start with,

17 and I believe that's on Page 19. Could you explain

18 what NUREG/CR-6728 is? And I believe this is again

19 listed in your references, this is reference No. 21

20 entitled Technical Basis for Revision of Regulatory

21 Guidance on Design Ground Motions, Hazard and Risk

22 Consistent Ground Motion Spectra Guidelines, dated

23 October 2001.

24 A. NUREG/CR-6728 is a first volume of a

25 pair of volumes that are being prepared by a

1 contractor. I believe the contracting company is
2 Risk Engineering. The purpose of that document was
3 to prepare technical background for the Staff in
4 anticipation of their preparing a reg guide on
5 ground motions in the future.
6 Q. Has that document, to your knowledge,
7 yet been adopted or referenced in any NRC
8 regulations or other regulatory guidance?
9 A. I believe it has probably not been, but
10 I do not know for sure.
11 Q. Okay. If you would, I'd like to go to
12 your Answer No. 14, turning back for a moment. In
13 this question, you were asked to describe the
14 probabilistic seismic hazard analysis or PSHA
15 methodology. And in your answer, there's a
16 sentence that reads as follows: You're describing
17 a curve here and you say, quote, "The result is a
18 curve of estimated annual probability of exceedance
19 versus level of ground motion."
20 Is that something that's normally or
21 commonly referred to as a seismic hazard curve?
22 A. Yes.
23 Q. And then you say that, "This curve can
24 be used to select the design ground motion at a
25 level corresponding to a pre-specified mean annual

1 probability of exceedance."

2 Could you explain how that is done?

3 A. That means if someone has specified the

4 mean annual probability of exceedance to be used,

5 for example, in a set of specifications or design,

6 one can enter that curve on the vertical axis,

7 which would be the probability level, and read down

8 to the horizontal axis and find the corresponding

9 level of ground motion.

10 Q. So that the seismic hazard curve would

11 contain within it all different levels of mean

12 annual probability of exceedance?

13 A. Correct.

14 Q. And one really turns to the curve and if

15 they say I want a 3,000-year or 1,000-year or

16 2,000-year MAPE, then they would simply be able to

17 look at that chart and see where the intersection

18 curves?

19 A. Exactly right.

20 Q. Turning to your Answer No. 28, and this

21 is on pages 17 to 18. You have a discussion in

22 this answer of a recent revision to DOE Standard

23 1020-94. And starting at the bottom of Page 17,

24 you state as follows: Quote, "A minor change has

25 also been made to the PC3 category to permit the

1 use for PC3 category structures and components of
2 USGS national probabilistic seismic hazard maps",
3 closed quote.
4 Are those maps what are commonly
5 referred to as the national earthquake hazard
6 reduction program maps?
7 A. Yes, they are.
8 Q. You go on to say, "To meet building code
9 needs, these maps are printed for this 2500-year
10 level."
11 As I understand your answer in this
12 question, you're stating that DOE revised its
13 Standard 1020 from a 2,000-year to a 2500-year
14 return period in order to be consistent with the
15 NP -- I'm sorry, NEHRP maps?
16 A. To the best of my knowledge, that is why
17 that change was made, yes.
18 Q. Are you aware of any document issued by
19 DOE which contains that explanation?
20 A. No, I am not.
21 MR. TURK: Your Honors, I'd like to
22 distribute at this time, a document. And while the
23 document is being distributed, I will identify it
24 for the record. This is the cover page and pages
25 iii to v of a document entitled DOE Standard

1 1020-2002 dated January 2002 and the document is
2 entitled DOE Standard Natural Phenomena Hazards
3 Design and Evaluation Criteria for the Department
4 of Energy Facilities.
5 JUDGE FARRAR: Did you want this marked
6 as a potential exhibit?
7 MR. TURK: Yes, I would, Your Honor.
8 JUDGE FARRAR: That would be Staff --
9 MR. TURK: I believe this one becomes
10 II.
11 JUDGE FARRAR: All right, we'll have the
12 reporter mark that for identification as Staff
13 Exhibit II.
14 MR. TURK: Thank you, Your Honor.
15 JUDGE FARRAR: Give her a moment to do
16 that.
17 (EXHIBIT-II MARKED.)
18 JUDGE FARRAR: All right, go ahead,
19 please.
20 MR. TURK: Thank you, Your Honor.
21 Q. (By Mr. Turk) Dr. Cornell, you're
22 familiar with this document, are you not?
23 A. Yes, I am.
24 Q. And if you would, take a look at the top
25 of page iv, and could you read to yourself the

1 statement beginning with, "For PC3 SSCs there is no
2 change to the performance goal when compared to the
3 previous version of the standard."
4 Your Honor, I won't read the whole
5 paragraph into the record because it is in the
6 proposed exhibit. But read the balance of this
7 paragraph to yourself and I'll ask you if this
8 refreshes your recollection whether there was a DOE
9 document that discusses making the change for PC3
10 category facilities.
11 A. Yes, it seems to make it very clear that
12 it's doing this for exactly the reasons I believed,
13 and I appreciate your reminding me of where I had
14 read that.
15 Q. In fact, it states that it is not the
16 intent of this revision to alter methodology for
17 evaluating those types of facilities, PC3
18 facilities or to increase the performance goal by
19 increasing the return period for a 2,000-year
20 earthquake to a 2500-year earthquake; correct?
21 A. Correct.
22 Q. And then it explicitly states, "The
23 intention is more for convenience to provide a
24 linkage from the NEHRP maps and DOE Standards";
25 correct?

1 A. Correct.

2 MR. TURK: Your Honor, I would ask that

3 Staff Exhibit identified as II be admitted into

4 evidence.

5 JUDGE FARRAR: Any objection?

6 MR. GAUKLER: No objection.

7 MS. NAKAHARA: No objection.

8 JUDGE FARRAR: All right, then the

9 document will be admitted.

10 (EXHIBIT-II ADMITTED.)

11 Q. (By Mr. Turk) I'd like to turn to your

12 Answer No. 69. And this appears starting -- the

13 question appears on Page 40 of your testimony, the

14 answer appears on Page 41. You use a term here

15 that I'm not familiar with. You mention something

16 called an SF factor. And excuse me one second, may

17 we go off the record? It may be that I've cited

18 the wrong answer.

19 JUDGE FARRAR: Off the record.

20 (A discussion was held off the record.)

21 Q. (By Mr. Turk) I'm sorry, I misspoke.

22 This is Answer 29, not 69. Show you how a

23 typographical error can throw a lawyer off track.

24 In the Answer 29, you refer to a factor

25 SF contained in DOE Standard 1020-2002. What is

1 the SF factor?

2 A. In DOE Standard 1020-94 and dash 2002,

3 the standard asks that you, having obtained the

4 ground motion associated with the particular annual

5 probability which you've entered the curve we just

6 discussed, once you have that ground motion level,

7 that you then increase it -- or multiply it rather

8 generally by 1.5 and then by another factor called

9 SF. That factor depends upon whether you value --

10 the value of that factor depends on whether you're

11 in category PC4, the more hazardous conditions such

12 as a nuclear power plant when you multiply it by

13 1.25 for the value of SF. Or whether you're in

14 PC3, in this case, under the old standard, 1020-94,

15 you multiplied it by an SF value of simply one.

16 Under DOE 1020-2002 as the footnote indicates, that

17 factor has been reduced from 1.0 to 0.9.

18 Q. Does the acronym SF stand for a

19 particular phrase, to your recollection?

20 A. Not to my recollection.

21 Q. And what is the purpose, then, for

22 including this SF factor?

23 A. The purpose is to adjust the level of

24 conservatism in the standards and criteria of this

25 DOE document.

1 Q. In order to reflect the appropriate risk
2 posed by the facility?
3 A. In order to maintain the -- what they
4 call the target performance goal, which is, in
5 turn, associated, yes, with the level of hazard
6 associated with failure of the facility.
7 Q. Now, in your footnote No. 7, you
8 indicate that the SF factor has been reduced from
9 1.0 to 0.9 to accomplish this change, and if I read
10 back into the text, I see that this is associated
11 with changing the PC3 return period earthquake from
12 a 2,000-year to a 2500-year. Could you explain the
13 interface of these concepts?
14 A. Yes. As I have discussed in my
15 testimony, the nature or the degree of safety that
16 one obtains in a seismic design is that it has two
17 intimately related elements. One is the level of
18 the so-called design-basis ground motion or DBE in
19 NRC plan, and the other is the level of
20 conservatism built into the design standards and
21 acceptance criteria, such as implied by the
22 Standard Review Plan.
23 In the DOE framework, those levels of
24 conservatism criteria are established in this
25 document 1020-94, 1020 now, 1020-2002. So both of

1 those factors determine the resulting level of
2 safety or the resulting likelihood of failure,
3 which is the inverse of that.
4 And one can obtain a particular level of
5 safety by adjusting the ground motion design-basis
6 earthquake up and down or by -- and/or by designing
7 the level of conservatism in the acceptance
8 criteria up and down. And this is an example of
9 where the objective of the DOE was to maintain a
10 common level of probability of failure, namely
11 about 10 to the minus four for the PC3 category,
12 and when adjusting for convenience, the ground
13 motion -- design-basis ground motion from one
14 associated with 2,000-year return period up to
15 2500-year return period, which implies that the
16 design-basis ground motion is getting increased
17 somewhat, if they had left the standards or
18 criteria as they had been, they would be obtaining
19 a higher level of safety, a lower level of
20 probability failure.
21 So because they did not want to change
22 that level of safety, they thought it was already
23 adequate, they reduced the conservatism in the
24 standard somewhat by taking this factor from 1.0
25 down to 0.9 maintaining the same safety level.

1 Q. I'd like to turn now to your answer No.
2 36, and this appears at Page 22. Actually the
3 answer begins on Page 21, and continuing onto Page
4 22, you describe some studies conducted by EPRI.
5 That's the Electric Power Research Institute, is
6 that the correct acronym?
7 A. That's correct, yes.
8 Q. And LLNL, is that Lawrence Livermore
9 National Laboratory?
10 A. Yes, it is.
11 Q. And you mention that they had been
12 engaged in studies in the Central and Eastern U.S.
13 sites with respect to seismic hazards?
14 A. Yes.
15 Q. And then you also mention something
16 entitled the Senior Seismic Hazard Analysis
17 Committee or SSHAC report. That's what's commonly
18 referred to as the SSHAC report?
19 A. Correct.
20 Q. Could you explain what is the SSHAC
21 report?
22 A. The SSHAC report is quite thick, several
23 volumes, but its purpose is to deal with
24 probabilistic seismic hazard analysis and to
25 provide guidance to those people who use such

1 analysis methods, both in helping them step through
2 the characterization of the seismicity of the site
3 or the faults surrounding the site, to deal with
4 the characterization of ground motion prediction.
5 And probably most importantly, to give guidance as
6 to how to deal with the significant uncertainties
7 that are associated with characterization of
8 seismic hazard in the United States, especially in
9 the Central and Eastern United States.
10 To deal with the fact that different
11 experts in this field, while having equal levels of
12 expertise, may have somewhat different opinions
13 about elements of that -- of those models. And
14 that what the SSHAC report tries to do is to give
15 guidance as to how to deal with the aggregation of
16 these so-called multiple expert opinions and how to
17 put that into a quantitative form, and finally to
18 pass that through so that the uncertainties in the
19 characterization of the probabilistic seismic
20 hazard curve that we discussed earlier are properly
21 captured and consistently captured from one study
22 to the next.
23 Q. Do you recall the approximate date of
24 publication of the SSHAC report?
25 A. Approximately 1995. I don't have the

1 reference in front of me, I'm sorry.

2 Q. Is it possible that the official date

3 would have been 1997?

4 A. Absolutely.

5 Q. Does that refresh your recollection? If

6 not, that's fine. If you're not sure, that's fine.

7 A. It's quite fine, I'm sure.

8 Q. Do you recall who are the authors of the

9 SSHAC report?

10 A. The head of the committee was a

11 Dr. Robert Budnitz. I was on that committee.

12 Dr. Peter Morris, Dr. Kevin Coppersmith, Professor

13 George Apostolakis, and then Mr. Lloyd Cluff. That

14 may be all, perhaps I'm missing one.

15 Q. Are these gentlemen considered to be

16 experts in the field?

17 A. I believe they are, yes. I don't like

18 to characterize myself, but the rest of them surely

19 are.

20 Q. Do you recall if any of them were

21 associated with the Nuclear Regulatory Commission

22 at that time, at the time the report was issued or

23 was this a non-NRC publication?

24 A. This was a joint publication. The

25 project was jointly funded by the Nuclear

1 Regulatory Commission, EPRI, Electric Power
2 Research Institute, representing owners of power
3 plants, and the Department of Energy, because they
4 all had interest in this issue. The members of
5 that committee were not -- were independent
6 consultants of one kind or another.
7 Q. Now, later on in your testimony, I
8 believe this is in Answer 79 on Page 46, again you
9 refer to the studies of Central and Eastern U.S.
10 seismic hazards. This is about in the middle of
11 the paragraph.
12 A. Yes.
13 Q. On Page 46. And you say there, "There
14 was a significant discrepancy between the two
15 assessments of the mean estimates made by the two
16 major Central and Eastern U.S. seismic hazard
17 studies available at the time of the Guide's --
18 and you're referring here to Reg Guide 1.165 --
19 preparation."
20 When you refer to the two assessments or
21 to the two major CEUS studies, are you referring
22 here again to the EPRI and LLNL studies?
23 A. Yes, I am.
24 Q. Could you take a look at your answer
25 there and explain your statement that the two

1 studies differed with respect to their calculation
2 of mean estimates, and that a decision was then
3 made to utilize a median value.
4 A. Yes, would you like me to take a moment
5 to read that?
6 Q. Yes, please.
7 A. Yes. Now, you would like me to explain
8 what I've said there?
9 Q. Yes, please, if you could amplify that a
10 little bit.
11 A. Yes. As indicated, it's deemed
12 important in probabilistic hazard analysis, to try
13 to capture the uncertainties or varying opinions
14 among experts with respect to various parameters in
15 the characterization of the faults and the ground
16 motion prediction. And therefore, one obtains not
17 a single estimate of the ground motion at any -- or
18 the probability of exceeding the ground motion, but
19 a range of estimates, and those ranges are
20 quantified and captured as an uncertainty band
21 about in a central value.
22 The median estimate corresponds to that
23 of which 50 percent of the estimates are above and
24 50 percent below. The mean estimate is associated
25 with the average or center of gravity. Now, this

1 probability distribution that results, given that
2 it's on very small probabilities, 10 to the minus
3 threes, 10 to the minus fours, is typically what we
4 call a very skewed distribution, it has a short
5 left-hand tail and a very long right-hand tail.
6 Q. I'm sorry to interrupt. Could you
7 explain what those tails represent?
8 A. The tails represent -- the lower tails
9 would represent estimates of the probability of
10 exceeding, say, .7 g. Estimates of that
11 probability, annual probability that are below the
12 median, and the upper tails would represent
13 estimates of that probability that are above the
14 median estimate. And the right-hand tail stretches
15 much further, that is the upper tail stretches much
16 further than the lower tail. The implication of
17 that is that the mean estimate is considerably
18 larger than the median estimate. And hence -- and
19 further, that the more uncertainty that exists in
20 characterizing that estimate, the further apart the
21 mean and median estimates are. If that's clear.
22 And what was -- what typically happens is that the
23 median estimate represents something like a central
24 estimate, roughly comparable to what you would
25 obtain if you failed to go through the

1 quantification of all of these uncertainties. If
2 you just took a central value point.
3 If you conclude as a result of your
4 characterization or quantification of these
5 uncertainties, that the uncertainties are large,
6 the mean will be considerably larger than the
7 median. If you conclude that the uncertainties are
8 comparatively small, the mean will be quite close
9 to the median. So the mean becomes indirectly a
10 measure of the degree of uncertainty.
11 And what happened in these two studies
12 was that the assessment of the uncertainty -- in
13 the initial studies, the assessment of the
14 uncertainty was quite different even though the
15 assessment of the medians were quite similar.
16 JUDGE FARRAR: Are you saying that in
17 your -- in the judgment of the experts, the median
18 is more properly reflective of the truth, whatever
19 that means, but it would be in these long right
20 tail situations, it would be less conservative than
21 the mean would be?
22 DR. CORNELL: Yes.
23 JUDGE FARRAR: Okay.
24 Q. (By Mr. Turk) Now, you mention that
25 there were two different studies, one by Lawrence

1 Livermore and one by EPRI. What did their studies
2 show? What curves were they showing for median and
3 mean as compared to each other's studies?
4 A. Correct. Well, in very general terms,
5 what -- the result of the first round of those
6 studies was that the median hazard curves at a
7 particular site in the Eastern United States might
8 be quite similar, but the Lawrence Livermore curve
9 which would have a significantly higher mean
10 estimate than the EPRI curve. Indirectly, that
11 implies that the Livermore assessment procedure was
12 ascribing considerably more uncertainty to the
13 estimates than the EPRI curve was.
14 The Livermore group subsequently went
15 back and redid some of their assessment of the
16 individual experts and their treatment of this
17 quantification of the uncertainty, and in their
18 later release of that report, came up with a
19 significantly reduced uncertainty in their
20 estimates, in their characterization of the
21 estimates, drawing the two mean curves much closer
22 together after the revision of the Livermore
23 assessments.
24 Q. And when did Lawrence Livermore issue
25 its revised study or analysis?

1 A. I believe that was dated about 1993.
2 Q. You go on to say in this answer that
3 although the median estimate was adopted for
4 purposes of establishing the regulatory guide 1.165
5 guidance, you state that the discrepancy between
6 the two studies -- and here you're referring to the
7 EPRI and the LLNL studies?
8 A. That's right.
9 Q. Has since been largely resolved. And
10 then you conclude, "It has been clearly established
11 that the typical SSE at existing plants across the
12 country has a mean annual probability of exceedance
13 of 10 to the minus four."
14 So is it correct, then, that your
15 testimony indicates that the mean annual
16 probability of exceedance for nuclear power plants
17 across the country, without respect to particular
18 regions but nationwide, is approximately 10 to the
19 minus four?
20 A. The result of my -- of the discussion
21 that we were just having with respect to Livermore
22 and EPRI, related only to the Central and Eastern
23 United States.
24 Q. I'm sorry.
25 A. Their studies were restricted to those

areas, and what we just discussed led to the conclusion in the Central and Eastern United States, those sort of typical values were in the order of 10 to the minus four, both studies concluding that.

Q. All right. In the last sentence of this answer, appearing on Page 47, you say -- you don't refer to these as CEUS sites, Central and Eastern U.S. sites, you say the typical SSE at existing plants across the country has a MAPE of approximately 10 to the minus four.

A. Yes, I stand by that conclusion, and you're right, I perhaps should have, in the context of this question, made it specific to the Central and Eastern United States. But elsewhere in the document, I discuss other elements for the fact that plants in the Western United States also have an average or typical mean annual probability of exceedance of their SSEs of also about 10 to the minus four. I believe that's the following question.

Q. Now, you're speaking about nuclear power plants?

A. Correct.

Q. Is it not correct that nuclear power

1 plants in the Western United States have an average
2 mean annual probability of exceedance of
3 approximately, I believe it's 5,000-year return
4 period. What would that equate to in the 10 to the
5 minus four scale?
6 A. Two times 10 to the minus four.
7 Q. And that's correct, then, is it your
8 understanding that nuclear power plants in the
9 Western United States have an average between them,
10 mean annual probability of exceedance of two times
11 10 to the minus four?
12 A. Yes, that's my understanding.
13 Q. There's a lot of discussion in your
14 testimony of the different DOE standards, and I'd
15 like to ask you a question that may be
16 self-evident, but would you agree that the
17 Commission and the Licensing Board here are not
18 obliged to follow DOE standards in reaching their
19 decisions with respect to this facility?
20 A. No, to my knowledge, there's no reason
21 why you are obliged to follow those standards.
22 Q. Okay. I'd like to turn now to your
23 Answer No. 91.
24 JUDGE FARRAR: Mr. Turk, before you go
25 onto that, let me back up a question. I would have

1 expected that the answer on the Western U.S.
2 nuclear power plants would be a 10,000-year return
3 earthquake, just from the way the regulations are
4 framed. Why is it less?
5 DR. CORNELL: Well, recall what is
6 happening here is we're talking about plants which
7 were licensed typically in the 1970s when they were
8 following Appendix A which was a deterministic
9 procedure. What we're discussing now is what has
10 been found in retrospect having done probabilistic
11 hazard studies for these sites. We find, and the
12 Staff has found, that that process has led to a
13 mean return period typically of about 10 to the
14 minus -- let's put it in mean return periods. I'm
15 sure you're accustomed to flipping back and forth.
16 The return periods are about 10,000 years in the
17 east, and they appear to be about 5,000 years in
18 the west.
19 JUDGE FARRAR: So if you were designing
20 the western plants now, they would have to have a
21 more stringent design?
22 DR. CORNELL: That would not be my
23 decision generally. However, it would suggest that
24 if the standard were set at 10 to the minus four
25 for all plants, they would be -- result in somewhat

1 more stringent criteria than we have currently in
2 the west.
3 JUDGE FARRAR: All right, I would have
4 expected from the course this case has taken, that
5 a probabilistic approach, which I know you are, is
6 better than deterministic, would yield a less
7 conservative design. Yet what you're saying is for
8 the western nuclear power plants, it would have
9 yielded a more conservative design?
10 DR. CORNELL: Let's see. I do not
11 believe that I have suggested that it would lead to
12 a more conservative design for a nuclear power
13 plant.
14 MR. TURK: Your Honor, I may be able to
15 help with one question.
16 JUDGE FARRAR: Yeah, this may be one of
17 these instances where your version of my question
18 is better than my version of the question. So feel
19 free to help out if you sense either now or at any
20 time that you see where we're going but we're not
21 getting there, you and any witness, as well, to try
22 to help us out.
23 DR. CORNELL: Thank you. It seems to me
24 that if the criteria were set that the design-basis
25 earthquake should be a 10,000-year earthquake based

1 on a mean annual probability of 10 to the minus
2 four, both in the east and in the west, and then
3 when applied, that would set the ground motion
4 level at the same probability level elsewhere,
5 everywhere. The next question is, what is the role
6 of the Standard Review Plan and its criteria? What
7 would the impact of them be on the net level of
8 safety? That is, let's say, the probability of
9 core damage. There curiously, if you apply the
10 same criteria, you would end up with somewhat more
11 conservative designs in the Western United States,
12 because those criteria are designed to give you a
13 margin in a multiplicative sense on the ground
14 motion level.
15 To take this one step further, the
16 hazard curves, these curves we've been discussing,
17 are much steeper in the Western United States, and
18 so a given margin, let's say of 1.5 in ground
19 motion, will provide you with a much more -- a much
20 larger reduction in failure probability when you go
21 that much further beyond the ground motion level.
22 So the combination of the 10,000-year
23 ground motion and the SRP would lead to somewhat
24 more conservative designs in the Western United
25 States. To the degree that it's been suggested,

1 for example, by DOE, then you can back off a little
2 bit for western -- some western plants.
3 So if I understand your question, I
4 believe your reasoning was correct.
5 MR. TURK: Your Honor, may I interject a
6 question. Not to interrupt your questioning, but
7 maybe this will help.
8 JUDGE FARRAR: Yeah.
9 Q. (By Mr. Turk) Dr. Cornell, there's no
10 bright line that I know of on any map that says
11 this is the Eastern, this is the Central or this is
12 the Western United States. But obviously, there's
13 something different about earthquakes in the west
14 and earthquakes in the east, at least our
15 understanding of them. Could you explain whether
16 the uncertainty levels with respect to western
17 earthquakes are different than the uncertainty
18 levels associated with eastern earthquakes.
19 A. The discussions we're having here, the
20 distinction, is usually made between what we call
21 the Central and Eastern United States and the
22 Western United States, and that distinction is
23 usually the other side of the Rocky Mountains from
24 where we are now. And the Western United States --
25 the Eastern United States is what we call -- refer

1 to as an intra- plate condition, and the geology is
2 very old and it's very -- much of the area is
3 covered by large depths of sediment, and it's very
4 difficult to -- and also the seismicity level is
5 low. All of this leads to making it more difficult
6 to get scientific understanding as to the cause and
7 nature of earthquakes in the Central and Eastern
8 United States.
9 In contrast, in the Western United
10 States, the geology is younger. As we go to the
11 far west, along the Pacific Coast, we're really at
12 the plate margins where the understanding of how
13 earthquakes happen and occur is much better. And
14 the data that we receive is much greater because we
15 have more earthquakes. So the positive side of
16 having more earthquakes is a better understanding
17 of the earthquakes. The net result that will show
18 up in the seismic hazard curves that we discussed,
19 and those uncertainty bands that exist around our
20 estimates of the probability of exceeding a
21 particular ground motion level, those uncertainty
22 bands will be narrower for a hazard assessment done
23 in the Western United States than they are in the
24 Eastern United States.
25 Q. The uncertainty bands would be narrower?

1 A. Narrower in the Western United States.
2 Q. In other words, the uncertainties are
3 reduced in the Western United States?
4 A. The uncertainties are reduced in the
5 Western United States, yes.
6 Q. Now, is that possibly a factor that goes
7 into your conclusion that in the Western United
8 States, the seismic hazard curve interaction for
9 western nuclear power plants, that the intersection
10 on that curve would be at about 5,000 MAPE average
11 between the western plants? Is that too confusing?
12 Let me try that again. Strike the question.
13 Is the reduced level of uncertainty
14 associated with western nuclear power plant sites,
15 a factor that enters into your conclusion that
16 western nuclear power plants have been designed for
17 us essentially an average MAPE of 5,000 years?
18 A. The conclusions of the differences in
19 the average return periods is 10,000 in the east
20 and 5,000 in the west, with respect to mean
21 estimates, are obtained as I alluded to earlier by
22 a back calculation in probabilistic terms of what
23 was -- what the SSEs were that were obtained by the
24 former Appendix A approach, so-called deterministic
25 approach.

1 In my mind, it's a little difficult to
2 connect the fact that these differences, 10,000 and
3 5,000, are -- because they actually arose from
4 licensing practice in the east and west, that
5 licensing practice being deterministic, did not
6 calculate and recognize these uncertainties in an
7 explicit way, at least. It's a little hard for me
8 to make the stretch to say that's exactly why we
9 come up with these differences. .
10 Q. Are the tectonics different in the
11 Central and Eastern United States versus the
12 Western United States?
13 A. Yes, that's what I was alluding to, it's
14 a plate margin versus intraplate.
15 Q. Okay. If we wanted to retain
16 approximately the same level of risk at all
17 facilities, both CEUS and Western, would we set the
18 probability of exceedance of the SSE at a slightly
19 higher level for plants in the west using the
20 Standard Review Plan guidance, the design?
21 A. Would we do so in the Standard Review
22 Plan guidance? The current Standard Review Plan
23 guidance, which is a procedure considered to be
24 acceptable to the Staff, is, in fact, based on the
25 median estimate, not on the mean estimate. And it

1 recommends, finds acceptable in the Central and
2 Eastern United States, the use of a 10 to the minus
3 five median estimate. And this is reflecting that
4 difference between mean and medians again, and
5 going back to the history we discussed earlier. If
6 that same standard of 10 to the minus five median
7 were used in the west, it would, yes, lead to --
8 because of the uncertainty differences, lead to a
9 subsequent difference in the mean failure -- mean
10 probabilities of exceedance of the DBE. Because of
11 the differences on the uncertainties as just
12 discussed.
13 JUDGE FARRAR: Let me follow up on --
14 getting back to tectonic plates. How far are we
15 from -- in Skull Valley, from the plate boundary,
16 the west coast plate boundary to the extent that
17 you know, you know, precisely where it is?
18 DR. CORNELL: I would defer to the
19 residents of Salt Lake City. You probably know
20 better how far from the west coast they are than I
21 do, but let's say a thousand miles.
22 JUDGE FARRAR: Now, the Rockies
23 represent intraplate activity, do they not?
24 DR. CORNELL: Well, yes, they do.
25 Distinctions are a little difficult here, but it's

1 certainly clear that in California, particularly
2 the coastal part of California, you're next to the
3 plate boundaries which are slipping noticeably, and
4 by the time you go to the eastern United States --
5 Eastern Central United States, you're clearly in a
6 intraplate condition. In between the situation is
7 complex.
8 JUDGE FARRAR: So the Rockies are not,
9 clearly not a plate boundary, but a lot went on?
10 DR. CORNELL: But a lot went on and is
11 still going on, and this can be measured by GPS
12 measurements and by the -- as indicated by the
13 higher level of seismic activity between the
14 Rockies and the west coast.
15 JUDGE FARRAR: You mentioned the
16 difficulty in the central and eastern because of
17 the sediments and so forth. Do I recall correctly
18 that New Madrin is the least understood earthquake
19 and is it because of that reason?
20 DR. CORNELL: I'm not sure it's the
21 least understood. We have lots of earthquakes we
22 don't understand well. But it's certainly not well
23 understood because of these reasons.
24 JUDGE FARRAR: Yeah, not well understood
25 given how large it was?

1 DR. CORNELL: Particularly given how
2 large it was. It's also, of course, some 200 years
3 ago where we had relatively little recording
4 capability. No recording capability and only
5 historical inference.
6 JUDGE FARRAR: Okay, thank you.
7 Q. (By Mr. Turk) I'd like to turn to a
8 different subject. This is in Answer 91. And in
9 this answer, you discussed the International
10 Building Code, 2000 procedure in which the
11 procedures would multiply the design basis
12 earthquake by two thirds. Do you see that?
13 A. Yes, uh-huh.
14 Q. Could you explain how the IBC procedures
15 work and why they do that?
16 A. Yes, I can. It's a little torturous. I
17 have to step back one step in history. Until just
18 a few years ago, all building code standards were
19 based on a 500-year return period. And then they
20 applied certain levels of conservatism and criteria
21 of the type that appear in all seismic structural
22 standards. The net result of those conservatisms
23 was to introduce some margin and that the
24 failure -- implied failure probabilities were
25 certainly less than one on 500, perhaps one on a

1 thousand. That is what the DOE 1020-24 concludes.
2 What was, let's say, unfortunate about that
3 procedure of setting the design-basis at 500 years
4 and then providing a margin, is that it leads to a
5 nonuniform safety across the United States for
6 buildings.
7 JUDGE FARRAR: Dr. Cornell, hold on one
8 second.
9 DR. CORNELL: So my conclusion -- my
10 statement at that -- prior to changing the tape was
11 that this prior policy of using a 500-year return
12 period and some comparatively large margin, was
13 that it leads clearly to nonuniform safety and the
14 reason is because the hazard curves we've been
15 discussing have different slopes. They're much
16 steeper in the west and much shallower in the east.
17 Again, as we discussed earlier, the safety criteria
18 give you a margin on the horizontal axis of a given
19 fraction or multiplier of 1.5 instead of 1.2. and
20 therefore, it led to nonuniformity in the implied
21 failure probabilities. And the implication was
22 that the failure probabilities were lower in the
23 Western United States where the curves were
24 steeper, and the failure probabilities were higher
25 in the Central United States where the curves are

1 shallower.

2 So the decision was made through the

3 committees involved in making these changes in

4 building codes, to try to homogenize that safety

5 level, and they did so by the very rational

6 approach of changing the design ground motion from

7 500 years to another number, about 2,500 years,

8 where such that the probability of exceeding the

9 ground motion for the design-basis is much closer

10 to the failure probability. So that you reduce

11 this sensitivity to the varying slopes.

12 So that changed the design-basis to a

13 more -- a larger ground motion by increasing the

14 return period of the design-basis. However, no

15 one, particularly no one in California where

16 building costs are very much driven by seismic

17 criteria, no one there believed that the safety

18 levels were wrong, and so it was concluded that the

19 safety levels should be basically calibrated to

20 those in the Western United States. To achieve

21 that calibration -- this is very convoluted. You

22 may wonder why engineers go through these steps.

23 But to achieve this calibration, the first step

24 in -- the next step following the establishment of

25 a 2500-year ground motion in this code, is to

1 immediately multiply it by two thirds to make it
2 smaller again.
3 Q. (By Mr. Turk) Excuse me, where is that
4 done? Is that location specific or --
5 A. No.
6 Q. Region specific?
7 A. No, that's the point. It is done
8 uniformly across the country.
9 Q. And I'm sorry, what was the reduction
10 you mentioned?
11 A. Two thirds. They're the same as
12 indicated in your question.
13 So this returned the design ground
14 motion, the effective design ground motion in
15 California about back to where it was before.
16 Which is another way of saying the difference
17 between a ground motion of 500 years and 2500 years
18 in California is about a factor of one and a half
19 or two thirds. So California didn't change, that
20 left the people in California happy. But when
21 applied to other regions, for example, Salt Lake
22 City, that same factor of two thirds reduces the
23 ground motion by about the same amount, which is,
24 let's say, recovered by the margins in criteria.
25 But it means that the mean return period of this

1 reduced ground motion is somewhat higher, like 800
2 years, again reflecting the shallower slope of the
3 hazard curves.
4 Q. Shallower as compared to California?
5 A. As compared to California, exactly. So
6 that is basically the idea, which was to increase
7 the design-basis ground motion return period, but
8 to, in turn, reduce it again such that things would
9 be more or less -- the net failure probabilities
10 would be maintained in California where it had the
11 greatest impact and where it was felt then the
12 greatest experience and suggested we had about the
13 right level of safety in buildings. But to
14 effectively increase the conservatism of designs in
15 the Central and Eastern United States or other
16 parts of the country where the hazard curves have
17 this shallower slope.
18 Q. All right. Taking the Salt Lake City
19 area as an example, if you reduced the 2500-year
20 return period by this factor of two thirds, you get
21 approximately, I calculate about 850?
22 A. Yes, I used the number 800 on Page 51,
23 Answer 91.
24 Q. 850, is that a more correct number or
25 are there some hidden --

1 A. Nothing hidden. Just reading curves,
2 estimating, interpolating.
3 Q. And then for Salt Lake City, they would
4 multiply back up in order to reflect greater
5 uncertainties?
6 A. No.
7 Q. Okay. Could you explain what happens
8 with respect to Salt Lake City?
9 A. With respect to Salt Lake City, they
10 would start with a 2500-year return period. They
11 would -- that would give them a particular ground
12 motion. Usually we've been discussing in these
13 proceedings, the ground motion in terms of the peak
14 ground acceleration. They would then into code
15 multiply that number -- in applying the code,
16 multiply that by two thirds to reduce the amplitude
17 of that ground motion, which has the net effect of
18 giving a ground motion with a return period of
19 about 800 years.
20 Q. About 800?
21 A. 800, 850, yes.
22 Q. Okay. And then you say there is a 1.5
23 importance factor that would be applied. Now, is
24 that applied only in areas other than California
25 or --

1 A. No, that's applied only to certain
2 structures, to certain facilities that -- for
3 example, facilities that contain hazardous
4 materials, where one would like higher safety
5 levels or lower failure probabilities. So
6 uniformly everywhere in the country, one would use
7 that same importance factor for those, a subset of
8 structures that falls into this condition, into
9 this categorization.
10 Q. By way of background, in California, if
11 they had a hazardous material site, would they also
12 be applying an importance factor?
13 A. That's my understanding, yes.
14 Q. Now, in Utah -- I don't know if you're
15 familiar specifically with how Utah applies the
16 building code.
17 A. It's my understanding that Utah has
18 recently adopted the IC -- IBC 2000 code. Just as
19 of January of this year if I'm correct.
20 Q. So they have adopted this procedure of
21 taking the peak ground acceleration, 2500-year
22 return period, scaling it down by two thirds?
23 A. Correct.
24 Q. And then they would apply the importance
25 factor based upon the importance -- the safety

1 importance of the facility or structure under
2 consideration?
3 A. Correct. Generally, you would -- for
4 most buildings, you would leave it simply at that
5 reduced level and only for some subset of hazardous
6 buildings or perhaps buildings essential to post
7 earthquake recovery, like police, fire departments,
8 hospitals, perhaps one might apply this increase.
9 Q. In your testimony at Answer 91, you
10 refer to this importance factor as the factor of
11 1.5.
12 A. Yes.
13 Q. How does the multiplication work? What
14 do you multiply that factor against, and what
15 number do you come up with?
16 A. You multiply that against the ground
17 motion. So the net effect of the two thirds
18 reduction, the 1.5 increase is to put you back at a
19 2500-year ground motion for these essential or
20 hazardous facilities.
21 Q. Then obviously I'm multiplying the wrong
22 thing, because if I take, let's say, 800 or 850 and
23 multiply it by 1.5 -- I'll start with 850 and
24 multiply that times 1.5, I get 1275.
25 A. Yes. I'm sorry, but you are multiplying

1 wrong, thanks.

2 Q. Okay. So what should I do -- that's why

3 I'm asking you the question. What should I do to

4 get it right?

5 A. What you should do to get it right is

6 you should enter the hazard curve at 2500 years,

7 let's say, for argument sake you come up with

8 .5 g. No, let's say .6 g, I can do that in my

9 head. You would then multiply that by two thirds

10 and get .4 g. We're saying that that -- for sake

11 of this argument, that would give you a return

12 period of 800 years in Salt Lake City, or 850. If

13 you had an essential building, you would then take

14 that .4 g and multiply it by 1.5 g, back up to .6

15 g, giving you again a return period of about

16 2500-year.

17 Q. So I was using the wrong multiplier.

18 It's not the return period that you're multiplying,

19 it's the --

20 A. The ground motion.

21 Q. The ground motion.

22 All right, thank you for that. And let

23 me turn back -- Your Honor.

24 JUDGE FARRAR: Let me -- while you're

25 thinking up the makeup work for Mr. Turk,

1 Professor, let me get back to tectonics for a
2 moment before it gets too distant.
3 Under the old deterministic rules, there
4 was a tectonic province theory. Does that play any
5 role in coming up with the concepts you need to
6 deal with in probabilistic approaches?
7 DR. CORNELL: Under the old approach, as
8 you call it, Appendix A, the deterministic
9 approach, as applied in the Central and Eastern
10 United States, there was something called a
11 seismotectonic province. And that was a concept
12 where it was suggested that one could block out
13 parts of the map based on the two issues of the
14 tectonics, might they be similar in terms of what
15 old faults might have been there, and also in terms
16 of the seismicity. That is, how many earthquakes
17 were counted per square kilometer.
18 JUDGE FARRAR: Right.
19 DR. CORNELL: So those arrived at those
20 seismic tectonic provinces. Those were then -- one
21 associated a particular maximum earthquake with
22 each of those provinces, one then predicted for
23 each of those provinces, what might be the ground
24 motion at my site. That was the Appendix A
25 procedure.

1 We don't have exactly the same thing in
2 the Central and Eastern United States today. You
3 will see, however, because of the failure to be
4 able to identify nice clean faults as we might in
5 the Western United States, one will see zones which
6 are called areals, areal sources. That is zones on
7 the maps which are considered to be basically
8 homogeneous with respect to the rate at they
9 produce earthquakes, with respect to the largest
10 magnitude they can produce.
11 JUDGE FARRAR: Areal is spelled how?
12 DR. CORNELL: A-R-E-A-L, as to do with
13 area as opposed to doing with line, which might be
14 the fault. And those areal sources play a very
15 analogous role to the old seismotectonic zones,
16 yes.
17 JUDGE FARRAR: Okay. And refresh me,
18 getting back to tectonic plates, where's the
19 eastern edge of the North American plate?
20 DR. CORNELL: I would be somewhere in
21 the middle of the Atlantic ocean.
22 JUDGE FARRAR: And is earthquake
23 activity there as vigorous as the western edge?
24 DR. CORNELL: No, it's not. It's a
25 different -- I would defer to perhaps Dr. Arabasz.

1 I believe that one plate is subducting under the
2 other perhaps. No, pardon me, pardon me. I
3 believe that's a place where material is coming
4 from the bottom of the Atlantic ocean.
5 MR. TURK: There's an uplift process
6 going on there.
7 JUDGE FARRAR: Right.
8 DR. CORNELL: Well, a welling up of
9 material from underneath the plate to create new
10 plates.
11 JUDGE FARRAR: All right, go ahead,
12 Mr. Turk.
13 Q. (By Mr. Turk) Thank you. I need to
14 return to a prior line of questioning. I think
15 maybe I can understand something a little better if
16 I understand this. You were talking about
17 establishing a comparable level of risk in the
18 Eastern and Central U.S. versus the Western U.S..
19 A. In our discussion of the IBC building
20 code?
21 Q. No, before that, when we were talking
22 about the difference between the ten to the minus
23 four mean annual probability exceedance on the east
24 coast versus the two times ten to the minus four
25 figure on the west coast.

1 A. Yes.
2 Q. If in the Western United States, we
3 wanted to maintain the same level of risk as
4 existing nuclear power plants located there --
5 A. Yes.
6 Q. -- would we use a 5,000-year return
7 period average?
8 A. It seems to me that would be legitimate,
9 yes.
10 Q. So that would give you comparability to
11 existing nuclear power plants in the Western United
12 States?
13 A. Correct. And as I alluded to earlier,
14 probably more nearly a similar failure
15 probabilities or, say, core level damage level.
16 Q. I'm sorry. It will be essentially a
17 comparable level of core damage?
18 A. Probabilities.
19 Q. Probabilities. And they would be
20 comparable to existing nuclear power plants in the
21 Western United States?
22 A. Correct.
23 Q. I have one more question I wanted to ask
24 you, turning back to this Staff Exhibit II, the
25 introduction to DEO Standard 1020-2002.

1 A. Yes.

2 Q. There's another sentence that we didn't

3 read aloud on Page iv at the top -- the top

4 paragraph on the page. The last sentence of the

5 paragraph states, all PC3 SSCs -- that's system,

6 structures and components; correct?

7 A. Yes.

8 Q. "All PC3 SSCs which have been evaluated

9 for compliance with the previous version of this

10 standard do not require any reevaluation

11 considering that the PC3 level of performance has

12 not changed."

13 Do you see that statement?

14 A. Yes, I see that.

15 Q. Could you provide your understanding of

16 that sentence and what it means?

17 A. What that means is that although the

18 design-basis ground motion mean return period has

19 been changed from 2000 to 2500 years,

20 simultaneously, the conservatism in the acceptance

21 criteria, the other hand of our two-handed

22 approach, has been made less conservative, that was

23 the reduction of one of these factors from 1.0 to

24 0.9, and the net result is that the probability of

25 failure of this SSC has been retained at the target

1 value that they prefer in 1020, which is 10 to the
2 minus four. So they have not -- they have
3 intentionally not changed the target performance
4 probability by these two, these simultaneous
5 adjustments.
6 Q. If given your understanding of this
7 document, which I have no reason to dispute, if DOE
8 was to build a PC3 category facility in the future,
9 for instance, if they were to build something akin
10 to a dry cask storage ISFSI using a 2500-year
11 return period, would that facility be built to a
12 higher safety standard than existing PC3
13 facilities?
14 A. No.
15 Q. And that's because of the reduction from
16 1.0 to .9?
17 A. Correct. There would be the same amount
18 of steel in the wall.
19 Q. So the fact that they're building to
20 what is nominally called a 2500-year return period
21 earthquake, doesn't mean they're building a
22 stronger or safer facility than previously, it's
23 the same design to be utilized?
24 A. Precisely.
25 Q. I'd like to turn to one last area of

1 inquiry. In Attachment A to your testimony, this
2 looks to be a description of a calculation that you
3 performed or would be performed, you entitle it to
4 Determination of Risk Reduction Factors for SSCs at
5 Facilities Designed using NRC Seismic SRP
6 Standards. And on the first page of attachment A,
7 you have an equation and I see that you've
8 explained virtually all of the different components
9 of that equation. I don't see an explanation of E.
10 Could you explain what E is?
11 A. E is a number which is I believe 2.7,
12 approximately. It's what's called the basis of the
13 national logarithms.
14 Q. It's a constant volume?
15 A. Constant volume?
16 MR. TURK: Your Honor, that concludes my
17 examination. I thank the witness for his time and
18 I thank you all.
19 JUDGE FARRAR: Judge Lam has some
20 questions. I believe you and he -- he spent a
21 little time at your university.
22 JUDGE LAM: Yeah, four long years.
23 MR. TURK: That's about what we spent in
24 Salt Lake City.
25 JUDGE LAM: Good morning,

1 Professor Cornell, how are things back in the farm?
2 DR. CORNELL: Very nice. The weather is
3 beautiful.
4 JUDGE LAM: Great, great.
5 Professor Cornell, by reading your
6 prefiled testimony, I came across with the distinct
7 impression that you are a great proponent of
8 probabilistic seismic hazard study and have made
9 many contributions in the field. Now, one of the
10 statements that I read, you said was that one of
11 the advantages of this approach is it would permit
12 a more equitable and rational allocation of
13 resources; is that correct?
14 DR. CORNELL: Yes, I did.
15 JUDGE LAM: Now, with that background,
16 let me share with you my understanding and ask you
17 a couple of generic questions. I see for this
18 particular facility, if one were to use Appendix A,
19 Part 100, one would obtain a design-basis
20 earthquake with a set of ground motion values and
21 anyone were to go to a 2,000-year return interval
22 probabilistically, there will be another set of
23 ground motions, which would be significantly less
24 than the deterministic approach; is that correct?
25 DR. CORNELL: That is correct.

1 JUDGE LAM: Therefore, using a
2 probabilistic approach would result in a reduction
3 in safety margin without labeling which approach is
4 appropriate. Would you agree there is a reduction
5 in safety margin?
6 DR. CORNELL: Let me be quite precise.
7 The safety margin we would usually refer to as the
8 difference between the design-basis earthquake and
9 the safety achieved, but I would agree with you,
10 that there would be a reduction in the safety level
11 achieved or the probability of failure would
12 increase.
13 JUDGE LAM: Right, right, without
14 getting into the definition of what safety margin
15 if. If I do a deterministic approach, I'm at a
16 much higher level of ground motion, if I do it
17 probabilistically, I would have a lesser level of
18 ground motion, so there's a difference in those two
19 sets of ground motions.
20 Now, then if one were to adopt the
21 2,000-year return interval for this particular
22 facility, then the issue remaining for us to deal
23 with is is this, indeed, a design-basis earthquake
24 that would represent an adequate level of safety?
25 So there are two questions here, Professor Cornell.

1 One is, does Part 100 Appendix A represent an
2 unnecessarily conservative approach? That's
3 question No. 1. Question No. 2 is, does this
4 2,000-year return interval design-basis earthquake
5 represent an adequate level of safety for us to
6 consider?
7 DR. CORNELL: You're pausing for my
8 answer?
9 JUDGE LAM: Right. There are two
10 questions, Dr. Cornell. Would you share with us
11 your opinion on both.
12 DR. CORNELL: In brief, my opinion on
13 both is the answer is yes, that for this facility,
14 a design-basis associated with Appendix A would be
15 too conservative. And two, this facility designed
16 with an earthquake associated with a design-basis
17 of 2000 years would be safe enough, would be
18 adequately safe for the public.
19 JUDGE LAM: Would you elaborate --
20 first, let's go to question No. 1. Unnecessarily
21 conservative. How much more conservative than we
22 need here in Part 100 Appendix A? For example, you
23 know, if we design a large structure, like a
24 containment to withstand internal pressure,
25 typically, the design pressure, the failure

1 pressure is two to three times the design pressure.
2 So I would say we had a safety factor of maybe two
3 to three; right?
4 DR. CORNELL: Yes.
5 JUDGE LAM: And with Part 100 Appendix
6 A, when you label and also when I label it as
7 unnecessarily conservative, how many extra
8 conservatism is there?
9 DR. CORNELL: The conservatism that we
10 obtain -- when we refer to that, it's usually
11 associated with the margin beyond the design-basis
12 for which we have capacity. I believe that's
13 consistent with your example of the pressure in the
14 containment structure. Your actual capacity is
15 perhaps two to three times your design-basis
16 capacity.
17 JUDGE LAM: Right.
18 DR. CORNELL: We have that same
19 phenomena in the seismic area, and it's what I
20 alluded to, that we have a design-basis ground
21 motion, for example, proposed here, 2000 years and
22 we have a margin to be able to take earthquakes
23 beyond that because of the conservatisms in the
24 Standard Review Plan acceptance criteria.
25 Now, the question is how large is that

1 margin?

2 JUDGE LAM: Yes.

3 DR. CORNELL: That margin -- first, let

4 me observe that that margin will be the same

5 whether it's a factor of two or three, would be the

6 same whether we designed the -- chose the ground

7 motion on the basis of Appendix A or whether we

8 chose the ground motion on the basis of a

9 2,000-year return period. The margin would be the

10 same in both cases, it would be a multiplier of two

11 to three. The difference between Appendix A, the

12 deterministic approach and the use of a 2,000-year

13 return period to establish the design-basis

14 earthquake, would lead to a major difference in the

15 starting point or the design-basis level, but not

16 in the degree of margin one has above that level.

17 Is that clear?

18 JUDGE LAM: Yes, yes.

19 Well, the way I understand the State of

20 Utah contention, one of the major themes in the

21 contention is there is a deterministically

22 determined criteria and perhaps one should just

23 stick with that regardless of what the level of

24 conservatism represented in that approach. How

25 would you deal with that issue?

1 DR. CORNELL: Well, my approach to this,
2 which is consistent with a probabilistic
3 characterization of safety, which is consistent
4 with that adopted by the Staff and the Commission
5 in other areas, is that we should be thinking in
6 terms of establishing -- attempting to arrive at a
7 probability of significant hazardous condition.
8 For example, in a nuclear power plant, it might be
9 the probability of a core damage event, or here it
10 would be the probability of an event involving
11 release of hazardous material. We should be
12 thinking about setting standards, about what is
13 safe enough in terms of that probability level,
14 what is the likelihood of an incident or an
15 accident that it would be damaging to public
16 safety.
17 And that, what I have suggested, if we
18 take, for example, a 2,000-year return period as
19 our design-basis earthquake, because of the
20 margins, that probability of the hazardous release
21 event, would be a probability of 10 to the minus
22 four or less, say 10,000 years or greater.
23 So the effect of those margins that
24 you're -- that we know exist in the process of
25 design, is to make the failure probability

1 substantially lower than the probability of the
2 design-basis earthquake.
3 JUDGE LAM: Right. But Professor
4 Cornell, given that there are substantial
5 uncertainties associated with any probabilistic
6 assessment, isn't it a prudent approach to say,
7 well, sure, that's all well and good, you know,
8 because when you testified to societal benefits by
9 more equitable and more rational allocation of
10 safety resources which is consistent with
11 Commission goal of risk form regulations, that's
12 all well articulated. But given the number of
13 uncertainties involved in any probabilistic safety
14 assessment, isn't it prudent to add a little bit
15 more margin here and there, which I would refer to,
16 well, should we then use 3,000 years, 5,000 years,
17 10,000 years? I mean where does one draw the line?
18 DR. CORNELL: First, because -- the
19 uncertainties that you allude to, uncertainties
20 we've talked about in trying to scientifically
21 assess what the earthquake conditions are, what the
22 rates of slip on faults are, all of those are
23 subject to uncertainty and all of those are inputs,
24 to, let's say, the seismic hazard analysis --
25 probabilistic or deterministic seismic hazard

1 analysis. Those uncertainties are there. In a
2 proper probabilistic approach, we try to capture
3 and quantify those uncertainties. Some of those
4 uncertainties are reflected in the fact that two
5 experts with the same information have somewhat
6 different interpretations about what the slip rate
7 on a fault might be.
8 Similarly, when we go to the side of the
9 margins, what are the capacities of the structures
10 to withstand earthquakes of different degrees, we
11 not only do our best in trying to -- in making a
12 probabilistic approach. In trying to estimate what
13 that margin is, for example, by removing, trying to
14 remove the conservatisms that exist in standard
15 practice and evaluate them, let's say, in a
16 predictive or scientific way, we try to not only
17 remove those conservatisms, but we also try to
18 characterize that degree of uncertainty that you
19 allude to.
20 So in the computation of this failure
21 probability or its evaluation or estimate, one
22 would not only consider your best estimate of what
23 those margins are, but your uncertainties in
24 evaluating them. We attempt to do our best to
25 reflect those uncertainties in the process, as

1 well.

2 JUDGE LAM: But isn't it true, Professor

3 Cornell, one of the biggest uncertainty of

4 everything is the error of omission, the issue of

5 completeness, what we don't know, we don't know?

6 That is beyond quantification. You and I can't

7 quantify what we don't know?

8 DR. CORNELL: That is a potential

9 concern in doing probabilistic assessments.

10 JUDGE LAM: Right. The issue of

11 completeness is always the biggest thing?

12 DR. CORNELL: The issue of completeness

13 is always there. And in the area of earthquake

14 engineering, we -- again one does his best to

15 understand based on tests of structures into the

16 damaged and failure area, to understand how they

17 behave so that those surprises are not there.

18 JUDGE LAM: Okay. So is it fair to

19 conclude, based on your prefile testimony, that it

20 is your expert assessment that this 2,000-year

21 return interval earthquake is safe enough?

22 DR. CORNELL: That is my testimony.

23 JUDGE LAM: Thank you, Professor.

24 JUDGE KLINE: Okay. Professor Cornell,

25 we've just spent a week here discussing matters in

1 testimony related to the PFS proposal and
2 particularly addressing such matters as likelihood
3 of cask tipover and seismic stability of pads and
4 that sort of thing. But we also heard some
5 testimony indicating that even if the cask does tip
6 over, there are likely to be small consequences
7 because of the integrity of the multipurpose cask
8 inside and likely not to be a release.
9 Are you telling us now that in your
10 probabilistic seismic hazard analysis, that you
11 somehow take account of the total system
12 performance and come to an end point that says you
13 have somehow considered the likelihood of
14 radiological release? Is it incorporated in your
15 seismic hazard analysis or is it there implicitly?
16 How do we know that it's accounted for?
17 DR. CORNELL: Let me take the elements
18 of your question one at a time. First just a
19 clarification. When we use the term probabilistic
20 seismic hazard analysis, we're referring to that
21 component of the analysis that leads up to the
22 ground motion characterization.
23 JUDGE KLINE: Yeah, I understand that
24 limited definition, but I thought I heard you say
25 that you were somehow also taking account of the

1 end point risk?

2 DR. CORNELL: Yes.

3 JUDGE KLINE: Of the system, the overall

4 system risk?

5 DR. CORNELL: Correct, that was just a

6 clarification.

7 JUDGE KLINE: Okay.

8 DR. CORNELL: Those probabilistic

9 studies that go further beyond the input ground

10 motion through to the representation of the margins

11 in the structures into the effects of failure, for

12 example, as you allude to, what are the

13 implications of the casks tipping over, and the

14 margins that exist, because of the robustness of

15 that cask, that kind of analysis would usually be

16 referred to as a probabilistic risk assessment or

17 PRA. These kind of systems, full systems analyses

18 are conventionally conducted in nuclear power

19 plants, for example. So those studies, yes,

20 absolutely, try to go through the full chain from

21 the initiating event, in this case, the earthquake,

22 through the kinds of effects that earthquake will

23 have on the various elements of the problem

24 through, in their full case, to what are the

25 release implications and the off-site dose

1 implications.

2 JUDGE KLINE: What I'm trying to see is

3 where the boundaries are between these various

4 analyses, and I'm just trying to ascertain whether

5 the seismic hazard analysis, the probabilistic

6 analysis itself captured any of these or considered

7 explicitly any of these subsequent events?

8 DR. CORNELL: Yes. Well, you refer to

9 the probabilistic seismic hazard analysis which has

10 been done by Geomatrix that we've heard much about,

11 that analysis did capture the issues of the faults

12 and their uncertainties, et cetera. There has not

13 been done, to my knowledge, for this facility, a

14 full probabilistic risk assessment. That is not

15 necessary, in my opinion, to lead to the conclusion

16 that I have drawn, that the 2,000-year design-basis

17 earthquake will provide failure probabilities which

18 are low enough for public safety.

19 JUDGE KLINE: When the Board is seeking

20 under NRC regulations, a reasonable assurance of

21 overall safety, should we be looking to the end

22 point, then, of the failure sequence, or is there

23 anything we can learn from, say, just the seismic

24 hazard analysis and how it's related to cask

25 tipover in isolation?

1 DR. CORNELL: I believe, yes, the
2 ultimate question is public safety, and that's the
3 end point issue of hazardous release. So one
4 should be considering the full chain of the
5 likelihoods of initiating earthquake levels, that's
6 the seismic hazard analysis, and the potential
7 implications as realistically defined.
8 JUDGE KLINE: Okay. I think that's what
9 we're really trying to understand here, maybe you
10 can help us out, is how we get this sort of
11 integrated view of safety, even though we've spent
12 a week focusing on the details of it. And it isn't
13 clear how we assemble it all. So we need to
14 explore that. But I think I understand your
15 testimony, at least this far.
16 DR. CORNELL: My written testimony does
17 go to exactly how we tried to do that in the
18 structural area, as, for example, described by DOE
19 1020. And what it tries to do is to couple those
20 two parts of the problem, the level of the
21 earthquake and the degree of margin and its
22 implications through to a definition of failure
23 which would be associated with hazardous release.
24 And those two parts of the problem then become what
25 is the probability level of the design-basis

1 earthquake, 2000 years, and a factor which we refer
2 to as R sub R, at least in DOE 1020 terminology.
3 That R sub R is a number that I talk about in my
4 testimony which might be 10. For example, R sub R
5 is 10, the implication is that the probability of
6 the failure of that component, which might lead to
7 hazardous release, would be a factor of 10 below
8 the probability level of the earthquake. So in our
9 case, if that probability level of the earthquake
10 is five times 10 to the minus four, the margins
11 would provide a factor of 10, we would be at five
12 times 10 to the minus five as the likelihood of
13 this hazardous release. That's the attempt to
14 integrate these two effects.
15 JUDGE KLINE: Okay, thank you.
16 JUDGE LAM: If Judge Kline would permit
17 me to interrupt.
18 JUDGE KLINE: Please.
19 JUDGE LAM: Dr. Cornell, since you are
20 an expert in earthquakes, let me ask you your
21 opinion on one of the statements the State of
22 Utah's expert had said. They basically put forward
23 this observation. We have a facility with shallow
24 pad, which is three feet thick, an anchor cask at
25 relatively high center gravity, the height to the

1 diamond ratio is about 1.8, with poor foundation --
2 poor, I mean relatively poor -- and then with
3 untested soil cement, so this is relatively risky
4 proposition. I mean this is the impression that
5 they are putting forward. Now, as an expert in
6 this field, do you have an opinion, based on first,
7 principle -- now, we have heard a great deal of
8 testimony about the soil foundation, about how the
9 casks may behave, but I would ask you to go rely on
10 first, principles. Do you have an opinion about,
11 is this, indeed, a risky proposition? By risky, I
12 mean is this something one should not even think
13 about building?
14 DR. CORNELL: Not in my opinion. In my
15 opinion, this is not a risky situation.
16 JUDGE LAM: Would you elaborate?
17 DR. CORNELL: I believe that we -- that
18 I can infer and judge from the -- first, the
19 characterization of the ground motion hazard and
20 secondly, the levels of margins which are implicit
21 in the Standard Review Plan criteria, which have
22 been evaluated by other means. By information
23 provided by the various engineering firms
24 associated with the project, with respect to some
25 quantitative assessments of how large the first

1 principle margins as opposed to let's say the
2 regulated margins are, how large those first
3 principle margins might be, or at least lower
4 bounds on how large those first principle margins
5 might be. That one can infer that the likelihood
6 of hazardous release of this facility is very low.
7 For example, a number in the order of 10 to the
8 minus four. And for me, that is an adequate level
9 of public safety for a facility of this type.
10 JUDGE FARRAR: Let me see if I can just
11 clarify one thing. In the last few minutes, you've
12 talked about the likelihood of radiological
13 releases, but I thought in terms of the
14 design-basis accident and the probabilistic seismic
15 analysis, you were attempting with the Staff and
16 the Applicant, to guard against cask tipover,
17 whether or not that had radiological consequences.
18 And I understand why there may be an additional
19 factor of safety before you get to radiological
20 consequences, but your precise work here deals with
21 protecting against cask tipover, does it not?
22 DR. CORNELL: To go back to Judge Lam's
23 position, from the first principle's point of view,
24 it goes beyond tipover. This goes to Judge Kline's
25 comment, too. In my mind, the first principles go

1 to likelihood of release. And so in my mind, in
2 thinking about that, it's very important that it's
3 been demonstrated that even if the cask tips over,
4 the damage will unlikely lead to radioactive
5 release. That gives me a margin and redundancy
6 beyond the simple issue -- not so simple issue of
7 cask tipover.
8 JUDGE FARRAR: Maybe after the break, I
9 can ask Staff counsel about that, and after the
10 break, Ms. Nakahara, I'll ask you to introduce the
11 new faces that I should have asked for before, but
12 I think we're past our break time. So it's 10 of,
13 let's come back at five after.
14 (A recess was taken.)
15 JUDGE FARRAR: We're back in session.
16 Mr. Turk, in light of the -- some of the questions
17 I asked, let me ask you first. I want to make sure
18 I understand exactly what's at issue here. The
19 design-basis accident is cask tipover or protecting
20 against cask tipover as opposed to protecting
21 against consequences from tipover?
22 MR. TURK: The design-basis accident is
23 not cask tipover. The standard that the Commission
24 applies is defense in-depth approach, and what we
25 do is, as part of our analysis for any application,

1 whether there's an exemption request filed or not,
2 we require cask vendors to demonstrate the
3 consequences of a hypothetical cask tipover event.
4 So that we understand if a tipover was to occur,
5 what would be the consequence? That was done here
6 in a hypothetical way by Holtec. As all vendors
7 do, they assume regardless of the cause that a cask
8 tips over, they look to see what's the consequence.
9 And you heard testimony from the Applicant's
10 witnesses that the deceleration force is 43.2 g,
11 which is bounded by the 45 g design analysis limit
12 for the cask. We've also heard testimony that the
13 MPC itself has a 60 g deceleration force design
14 limit. So we're satisfied that in the event that a
15 cask was to tip over, there would be no release of
16 radiation. That's part of the defense in-depth
17 approach.
18 The standard is, under the regulations,
19 that systems, structures and components important
20 to safety must be shown to be able to perform their
21 intended safety function. Now, that's a general
22 statement, but that includes all aspects of a
23 facility. That all aspects of a facility are able
24 to perform their intended safety function in the
25 event of a design level event.

1 Now, in this case, the design level
2 event at issue is the design earthquake. The
3 Applicant has asked to do the probabilistic seismic
4 hazard analysis approach using a specific
5 2,000-year return period. The Staff has determined
6 that that's an appropriate level at which to set
7 the size of the earthquake. And we've determined
8 that with respect to that earthquake, they'd
9 meet -- the systems, structures and components will
10 meet their intended safety functions.
11 JUDGE FARRAR: Meaning --
12 MR. TURK: Now, tipover will not occur
13 in that event. We've determined that even for a
14 10,000-year event, through Dr. Luk's analysis and
15 through our review of the Applicant's calculations
16 in the SER, we've determined that even a 10,000 --
17 well --
18 JUDGE FARRAR: Stay with the 2000 for a
19 moment. For the 2000, the Staff's position is this
20 exemption was warranted and the application should
21 be approved, because the 2,000-year earthquake is
22 the right one, and if it occurs, the cask will not
23 tip over? That's the essence -- or that's the
24 fundamental Staff position. You then go beyond
25 that and say even if it does, there won't be

1 consequences.

2 MR. TURK: That's a slight variation of

3 how I would describe it. In the event of a design

4 level earthquake, the structures, systems and

5 components, including the casks will perform their

6 intended safety functions, which means there will

7 be no release of radiation, there will be no harm

8 to the public. In determining that, one of the

9 factors is they won't tip over. Now, we've done

10 the sensitivity study to say, let's go far beyond

11 the 2,000-year. Let's do the 10,000-year

12 earthquake. Our review of -- I'm sorry, Dr. Luk's

13 performance of his independent analysis, confirms

14 that the cask will not tip over even in that

15 extreme earthquake and therefore, there will be no

16 release of radiation even under those conditions.

17 JUDGE LAM: So the acceptance criteria

18 is not the lack of tipover, the acceptance criteria

19 is there's no releases?

20 MR. TURK: That's my understanding. The

21 intended safety function is maintaining the

22 confinement boundaries so that radioactive material

23 is not released.

24 JUDGE FARRAR: Mr. Gaukler, does this

25 square with your understanding?

1 MR. GAUKLER: I believe it does, in the
2 sense that there's a design-basis earthquake and we
3 must meet all the codes and standards for the
4 design-basis earthquake. And by doing so, there
5 will be no radioactive release, the systems will
6 perform their function. The point that Dr. Cornell
7 makes in his testimony, is that when you meet other
8 codes and standards for a design-basis earthquake
9 for a nuclear facility, built into those are the
10 extra capacity, and that's what he's talking about
11 when he goes to the first principle systems.
12 There's extra capacity on many different fronts
13 that shows that you're going to have a probability
14 of release much less, much smaller than the
15 design-basis earthquake.
16 JUDGE FARRAR: Ms. Nakahara,
17 Ms. Chancellor, does this square with your
18 understanding?
19 MS. CHANCELLOR: No, Your Honor.
20 JUDGE FARRAR: How did I know that?
21 MS. CHANCELLOR: Because we're on the
22 other side, Your Honor.
23 We're wondering what NRC NUREGs and
24 standard review plans have to do with this if the
25 whole issue is no radiation release? NUREG-1536

1 states that there can be no cask tipover.

2 JUDGE FARRAR: NUREG which?

3 MS. CHANCELLOR: 1536. The title is

4 Standard Review Plan for Dry Cask Storage Systems.

5 Not for nuclear power plants, by for ISFSIs. What

6 you are being asked to do is to relax the standard

7 for the ground motions, and you are also being

8 asked to squeeze down the conservatism in the

9 design. You really need to either increase the

10 level of ground motion so that you can have a

11 relaxed margin in the conservatism of the design.

12 PFS and the Staff want it both ways. They want to

13 relax the design-basis ground motions that PFS has

14 to design to, and they want to relax, squeeze the

15 standard down so that the only thing that you look

16 at is if a cask tips over, what's going to happen?

17 They don't seem to be concerned if the foundations

18 fail, if the CTB slides, if the pad slides. All

19 they seem to be concerned about, based on what

20 we've heard so far in this hearing, is there won't

21 be a release of radiation. Well, that's not the

22 standard. That's certainly not the standard for an

23 exemption, from existing regulations that require a

24 DSHA, which give you ground levels of about 1.5 g.

25 JUDGE FARRAR: This may not be the time

1 to resolve this issue, as long as we know what the
2 differences are, but can you all respond to what
3 Ms. Chancellor said, and particularly the reference
4 to NUREG-1536?
5 MR. GAUKLER: The NUREG-1536 is NRC
6 guidance and it's also the thing that you have to
7 meet with respect to design-basis earthquake, okay.
8 And so you got to meet the codes and standards with
9 respect to design-basis earthquake, and that goes
10 far beyond ensuring there's no radioactive release
11 because the defense in-depth principle that the
12 Commission has in all its standards.
13 Your Honor, we're not relaxing that.
14 Those are things that have to be met for the
15 design-basis earthquake. Then there's the second
16 question whether the design-basis earthquake
17 provides a adequate level of public health and
18 safety? Whether the 2,000-year design-basis
19 earthquake proposed by PFS, provides adequate
20 public health and safety? And in that context, you
21 look at the inherent capacity, the inherent
22 conservatisms in the designs by virtue of the
23 defense in-depth that the NRC employs in this
24 design-basis licensing process. Because that
25 defense in-depth, you get down to a much lower

1 probability for tipover, tipover to be shown as
2 less than one times ten to the minus four, and
3 assuming tipover, no radioactive releases. So you
4 get down to a much lower level of probability in
5 terms of any harmful release to the public health
6 and safety, which shows the inherent capacity of
7 the system and which is part of the process of
8 determining whether or not the exemption adequately
9 protects public health and safety. As Dr. Cornell
10 has talked to in terms of this, there's two things
11 that go into type of consideration. One is the
12 design level ground motion, and the second is in
13 inherent capacity by virtue of the conservatisms
14 built into the codes and standards.
15 JUDGE FARRAR: Mr. Turk, before you add
16 something more, an observer over the last couple of
17 weeks might have come away with the conclusion that
18 what was an issue here was, among other things,
19 cask tipover, and that if the Applicant's and the
20 Staff's witnesses failed to demonstrate that the
21 cask would not tip over under the relevant ground
22 motion from the relevant earthquake, that the State
23 would win. Now, I know there have been people,
24 Dr. Cornell has said it and I think Mr. Guttmann
25 said it, you know, don't worry about it a whole

1 lot, because even if the cask tips over, there
2 won't be a -- you know, a release. But someone --
3 people might have thought the issue was, we've got
4 to prove the casks won't tip over. Am I wrong?
5 MR. TURK: That's wrong. Let me give
6 you an example. If a cask vendor came to us with
7 an application and said I have a single pellet of
8 uranium fuel, spent fuel, I'm encasing it in a
9 20-foot wide, 50-foot high cask, that one single
10 pellet and we did our review and we found the cask
11 would tip over -- well, let me first start by
12 something else. The standard is not will that cask
13 tip over? The standard is will that cask release
14 radiation if it tips over? Because the intended
15 safety function is protection of public health and
16 safety.
17 Now, there are different steps you get
18 to. You might first do the analysis and as the NRC
19 guidance imposes upon applicants to demonstrate
20 will it tip over. Because if it doesn't tip over,
21 you don't have to get to the second question, which
22 is, what's the consequence of a tipover? So the
23 first question is, will it tip over? You don't
24 stop there. Even if the cask tips over, then what?
25 Then you look to see, will there be a release in

1 excess of 10 CFR 72.106(b), accident limits. If
2 there is, you can't license unless they change that
3 design. But if as a result of cask tipover,
4 there's no release in excess -- or no radiation
5 resulting in excess of the dose limits in the
6 regulations, then public health and safety is
7 protected, and the facility would be licensing.
8 JUDGE FARRAR: And where is all this in
9 the prefiled -- this latter point in the prefiled
10 testimony?
11 MR. TURK: We address in the testimony
12 of Mike Waters under dose limits, what is the
13 consequence of a cask tipover? What is the
14 appropriate dose limit that has to be met? But I
15 would also note that we've said many times in this
16 proceeding and the Board is well aware, what has to
17 be complied with is the regulation. The guidance
18 in whether it's NUREG-1536 or NUREG-1567 or the
19 Standard Review Plan, those are ways that we use --
20 those are methods that we use to get to the
21 ultimate determination of whether public health and
22 safety is protected in accordance with the
23 regulations. I'll give you a hypothetical case --
24 JUDGE FARRAR: Let me interrupt there.
25 As far as these Staff guidance documents go, to

1 restate again what I think is obvious, these are
2 how you go about your day-to-day business, but an
3 applicant is free to say, I have a better idea?
4 MR. TURK: Yes.
5 JUDGE FARRAR: Shall we defer the more
6 expensive review by you all, because it's not the
7 norm?
8 MR. TURK: Yes.
9 JUDGE FARRAR: And an intervenor is free
10 to say, that doesn't cut it, it's not good enough?
11 MR. TURK: I'll give you another
12 hypothetical case. Let's come back to the massive
13 cask with a single pellet of fuel in it. If that
14 cask is shown to tip over, even though the
15 regulatory guidance, which says, demonstrate the
16 consequence of tipover and show that it won't tip
17 over, that vendor or applicant can come to us and
18 say, well, it doesn't matter that it tips over, we
19 still meet the regulation. And the Commission
20 could license them, fine, if the regulation has
21 been satisfied. Even though the particular
22 approach of the guidance has not been followed or
23 has not been satisfied.
24 JUDGE FARRAR: Ms. Chancellor, before I
25 ask you to respond. Dr. Cornell, how many months

1 can you spend with us? Are you available after
2 today?
3 DR. CORNELL: Yes.
4 JUDGE FARRAR: Okay. I don't want to
5 extend this argument too long because our business
6 of the day is Dr. Cornell and we can wrap this up
7 later. But Ms. Chancellor, do you have something
8 to add before we return to the witness?
9 MS. CHANCELLOR: Yes, I would, Your
10 Honor. I agree that PFS must meet the regulations.
11 If that is the case, they should be complying with
12 Part 100.
13 Second, there has been no risk
14 assessment for the performance of SSEs at the PFS
15 facility. The way PFS gets there -- and the Staff
16 doesn't do this, the Staff doesn't rely on this at
17 all. The way PFS gets there is to point to NUREGs
18 and Standard Review plans for nuclear power plants.
19 When NUREG-0800 requires a factor of safety of 1.1.
20 In many instances, what we have been arguing about
21 in the proceedings is that PFS doesn't have that 10
22 percent design margin, let alone it's more than 10
23 percent. So I think to say that the bottom line
24 is, is there going to be a release of radiation,
25 that is not -- that does not give us any level of

1 conflict as to health and safety. If you are going
2 to allow a facility to be designed, that you can --
3 that potentially can -- the foundations can slide,
4 the casks may tip over, and all you're left with,
5 whoops, no problem, there's no radiation release.
6 That to me is pretty sloppy in terms of designing
7 and approving a nuclear facility. And there's a
8 chicken and egg problem here. Well, maybe not
9 chicken and egg, but we're looking at the
10 design-basis earthquake, and we're trying to
11 establish what those ground motions should be for
12 the design-basis earthquake. So maybe you should
13 be doing a radiation dose analysis on a 10,000-year
14 DBE to see just how safe this thing is.
15 JUDGE FARRAR: All right. We have
16 everyone's differing views. Obviously, it's not
17 appropriate for a ruling at this time, and so it
18 will be something you'll want to argue at the end
19 of the case. So I think with that understanding,
20 we can proceed with the State's cross-examination
21 and, Ms. Nakahara, we usually try not to ask too
22 many questions at the outset, but this was one of
23 these instances where we felt we had to understand
24 the witness's testimony in full early on so that we
25 could -- it would give you a better idea of the

1 extent of his views, and allow us to understand all
2 of the answers he'll give you on cross-examination
3 better. So with that, you may go ahead, but please
4 introduce your new faces to us.
5 MS. CHANCELLOR: I can do that, Your
6 Honor. To Ms. Nakahara's right is Dr. Walter
7 Arabasz. He's a research professor and director of
8 the University of Utah seismographic stations, and
9 these are earthquake -- the seismograph stations
10 record earthquakes in the eastern basin and range.
11 And Dr. Arabasz has worked with Dr. Cornell. We
12 found in this proceeding that all these experts
13 seem to know each other.
14 To my left is Dr. Jim Pechmann. He's a
15 research associate professor at the University of
16 Utah, also with the seismographic station, and his
17 specialty is earthquake seismology and earthquake
18 hazard analysis.
19 JUDGE FARRAR: But he doesn't want to be
20 a witness?
21 MS. CHANCELLOR: Oh, no, he pointed to
22 Dr. Arabasz. And the other State people are faces
23 that I think you've seen before.
24 JUDGE FARRAR: Right. We're delighted
25 to have both of you here in whatever capacity.

1 Ms. Nakahara, go ahead.

2 MS. NAKAHARA: Thank you, Your Honor.

3

4 CROSS EXAMINATION

5 BY MS. NAKAHARA:

6 Q. Good morning, Dr. Cornell. For the

7 record, my name is Connie Nakahara, and I represent

8 the State of Utah.

9 Isn't it true an NRC contractor ICF was

10 responsible for the technical basis of modifying

11 the rules in Part 72 to allow probabilistic seismic

12 hazard analysis for Osifies?

13 A. No, I don't believe that is true. I

14 believe the contractor ICF was hired to provide

15 technical guidance to the Staff after the

16 rule-making was put forward, which proposed

17 probabilistic seismic hazard analysis.

18 Q. For Osifies; is that correct?

19 A. For Osifies, yes.

20 Q. Thank you.

21 And you were hired as a committee member

22 or panel member to advise ICF in this matter; is

23 that correct?

24 A. That is correct.

25 Q. What were the time periods that you were

1 retained by ICF, if you recall?

2 A. Yes. My memory for dates is not the

3 best, but I was hired roughly the late summer of

4 1999, and my contract -- well, I was approached at

5 that time to become a member of this panel. The

6 contract extended, to the best of my knowledge,

7 from October of 1999 to March of 2000. About five

8 months, six months.

9 Q. And during this period that you were

10 retained by ICF, isn't it correct you were also

11 retained by PFS as a witness in this matter?

12 A. I was approached by PFS at about exactly

13 the same time, August that summer. I did not

14 commit to a contract at that time with PFS, but I

15 agreed to serve as a consultant if they needed me.

16 Q. Thank you.

17 Judge Lam asked you a few questions

18 about the design proposal at PFS, and so I'll

19 exclude those, but this follows up a little bit on

20 your understanding of what PFS has proposed.

21 Do you agree PFS's design proposes to

22 use cement-treated soil underneath these storage

23 pads?

24 A. Yes, I understand that is what they

25 intend to do.

1 Q. And are you aware that PFS proposes to
2 place soil cement adjacent to the storage pads?
3 A. Yes.
4 Q. And are you aware that soil cement will
5 be used as a structural support element in the
6 storage pad design?
7 MR. TURK: For clarification, is this
8 the cement-treated soil or the soil cement?
9 MS. NAKAHARA: Soil cement.
10 DR. CORNELL: I'm not sure I would
11 characterize it as a structural element, but it
12 certainly surrounds the pad, yes.
13 Q. (By Ms. Nakahara) Are you aware of the
14 purpose of the soil cement with respect to the
15 storage pads?
16 A. Of the soil cement?
17 Q. Yes.
18 A. Not totally, no. I understand one
19 reason is to permit the transporters to approach
20 the pad more easily.
21 Q. Do you agree PFS's design also proposes
22 to use soil cement around the canister transfer
23 building to provide resistance to sliding?
24 A. Yes, I'm aware of that.
25 Q. Are you aware of any facility in the

1 U.S. that has a similar design to PFS's design
2 using freestanding single dry cask -- dry storage
3 casks at a site where the ground accelerations, the
4 free field ground accelerations equal or exceed
5 those for a 2,000-year design-basis earthquake at
6 PFS?
7 A. No, I'm not.
8 Q. Are you aware of any facility in the
9 U.S. where spent nuclear fuel will be stored in
10 freestanding casks supported by cement-treated
11 soil?
12 A. No, I am not.
13 Q. Are you familiar with the term seismic
14 category one?
15 A. Yes.
16 Q. Are you aware of any structure at a
17 nuclear power plant where seismic category -- where
18 a seismic category one structure is not anchored
19 and is allowed to freely slide, uplift and rotate
20 when subject to strong ground motion?
21 A. I can't think of one at the moment, no.
22 Q. And are you aware that the storage pads
23 at the PFS site are on a shallow foundation?
24 A. They're on a foundation. I believe the
25 geotechnical engineer would categorize that as a

1 shallow foundation.

2 Q. Are you aware of any structure at a

3 nuclear power plant facility on a soil site with a

4 shallowly embedded foundation with design ground

5 motions with similar intensities of those as the

6 PFS design-basis ground motion?

7 A. No, I'm not, but I'm not a specialist in

8 that area. There may be some.

9 Q. Are you aware of any nuclear structure

10 where settlement of 1.7 inches or more were

11 anticipated during the design life of the facility?

12 A. No, I'm not, but again, I do not have an

13 extensive background in that area. There may be

14 some.

15 Q. Are you aware of any nuclear structure

16 where soil cement has been used to resist seismic

17 loading of the structure?

18 A. No, I'm not.

19 Q. PFS requests an exemption from the

20 regulations for a lower design-basis earthquake

21 equivalent to a 2,000-year mean return period;

22 correct?

23 A. Correct.

24 Q. You, yourself, did not recommend the use

25 of this 2,000-year mean return period, did you?

1 A. No, I did not.

2 Q. In Answer 23 of your testimony, you

3 state the Commission has rejected the notion that

4 licensing standards should be as high for ISFSI --

5 JUDGE FARRAR: Ms. Nakahara, if that's a

6 slightly new subject, let me follow up that last

7 answer.

8 When you say you did didn't recommend a

9 2000, did you not make any recommendation or did

10 you recommend something else?

11 DR. CORNELL: I was not involved with

12 PFS at the time they made that decision. I was

13 retained at that point as a possible advisor on

14 their PSHA, their probabilistic seismic hazard

15 analysis.

16 JUDGE FARRAR: Okay, thanks.

17 Q. (By Ms. Nakahara) In Answer 23 of your

18 testimony, you state, "the Commission has rejected

19 the notion that licensing standards should be as

20 high for ISFSI as for nuclear power plants";

21 correct?

22 A. Yes, that's what I say there.

23 Q. Other than in Part 72, the Commission

24 has not, in fact, quantified a lesser level of

25 licensing standards for ISFSIs in comparison to

1 nuclear power plants; isn't that right?
2 A. They have not quantified that
3 difference, is that what you said?
4 Q. Yes.
5 A. Only, for example, through the --
6 perhaps the recommendations as implied in the rule
7 making.
8 Q. Not the record, I apologize, I can't
9 enunciate today.
10 The Commission itself has not approved a
11 specific design-basis earthquake or mean return
12 period for an ISFSI under a probabilistic seismic
13 hazard analysis, have they?
14 MR. TURK: Objection, is this a generic
15 question she's asking, or is she asking whether any
16 other facility has been approved?
17 MS. NAKAHARA: It's a generic question.
18 DR. CORNELL: The Commission itself, no.
19 I believe the Staff has recommended --
20 JUDGE FARRAR: Wait, wait.
21 MR. TURK: The clarification goes to the
22 fact that there is an existing ISFSI that has been
23 allowed to operate with a PSHA with a certain
24 return period.
25 JUDGE FARRAR: Given the clarification,

1 do you still object to the question?

2 MR. TURK: No, as long as it's clear

3 that it's a generic question rather than has there

4 been a case where this has happened before.

5 JUDGE FARRAR: Okay. Go ahead,

6 Dr. Cornell.

7 MS. NAKAHARA: Would you reread the

8 question.

9 JUDGE FARRAR: Yeah, you may answer it,

10 if you remember it.

11 DR. CORNELL: I'd like to have it

12 repeated, please.

13 JUDGE FARRAR: Okay.

14 (Record was read as follows: "The

15 Commission itself has not approved a specific

16 design-basis earthquake or mean return period

17 for an ISFSI under a probabilistic seismic

18 hazard analysis, have they?)

19 DR. CORNELL: It's my understanding that

20 the Commission has not made a statement as to the

21 precise quantitative number for that. My

22 understanding is the Staff has suggested to the

23 Commission that their preferred value is 2000

24 years.

25 MS. NAKAHARA: Ms. Chancellor is

1 distributing what I'd like to have marked as
2 State's Exhibit 180, which is a November 19th, 2001
3 memorandum to William D. Travers, with the subject
4 Staff Requirements SECY-01-0178 Modified Role
5 Making Plan, 10 CFR Part 72, Geological and
6 Seismological Characteristics for Siting and Design
7 of Dry Cask Independent Spent Fuel Storage
8 Installations.
9 JUDGE FARRAR: All right. We'll have
10 the reporter mark that for identification.
11 (EXHIBIT-180 MARKED.)
12 JUDGE FARRAR: All right, the reporter
13 has marked the document. Mr. Turk, you wanted to
14 interject something?
15 MR. TURK: Yes, Your Honor. I would
16 note that this document is identical in virtually
17 all respects with the document that is included in
18 Staff Exhibit U that has been prefiled in this
19 proceeding.
20 MS. NAKAHARA: That's fine, Your Honor.
21 I'd be happy to use Staff Exhibit U.
22 JUDGE FARRAR: Okay. Then we will not
23 admit this.
24 MS. NAKAHARA: A point of clarification.
25 Since this has already been marked, do we start

1 with the next number 181 with our exhibits?

2 JUDGE FARRAR: Yes, right.

3 Q. (By Ms. Nakahara) Dr. Cornell, are you

4 familiar with this document?

5 A. Yes, I've read this letter before.

6 Q. And isn't it true that the Commission

7 has directed the Staff to solicit comments on a

8 range of probability of exceedance levels from 5.0E

9 to the minus four through 1.0E to the minus four?

10 A. Yes.

11 Q. And that's notwithstanding the Staff's

12 recommendation for a 2,000-year return period; is

13 that correct?

14 A. Yes.

15 Q. And could you clarify for us lay people

16 what a return level earthquake, a 5.0E to the minus

17 four would correspond with?

18 A. 2000 years.

19 Q. Thank you.

20 A. I should point out, this is not the same

21 E we discussed earlier, the lower case E.

22 Q. And would you clarify what 1.0E to the

23 minus four would correspond with?

24 A. A 10,000-year mean return period.

25 Q. Thank you.

1 JUDGE FARRAR: Ms. Nakahara, are you
2 going to examine further on that point?
3 MS. NAKAHARA: No.
4 JUDGE FARRAR: How and when does the
5 Board learn of the status of that rule making? For
6 example, the Board would hate on November 10th of
7 this year to have worked for two months writing an
8 opinion based on the current state of the rules and
9 to find a new rule come out that low and behold is
10 applicable to this case. How do we guard against
11 that possibility?
12 MR. TURK: Your Honor, I can give you a
13 current status report, at least current as of some
14 date in the past that I left Washington to come to
15 these hearings. The Staff had in process a
16 proposed rule making for publication. To my
17 knowledge, it has not been issued yet.
18 JUDGE FARRAR: As a proposal or as a --
19 MR. TURK: As a proposed rule.
20 JUDGE FARRAR: So assuming nothing has
21 happened since you were gone, the next step would
22 be a proposal would be out, public comment?
23 MR. TURK: Yes.
24 JUDGE FARRAR: Final rule?
25 MR. TURK: Yes. And that would be many

1 months away from where we are today.

2 JUDGE FARRAR: Right. So we --

3 presumably, there's no way that's going to happen

4 before we write a decision in November?

5 MR. TURK: I don't --

6 JUDGE FARRAR: That's not a criticism of

7 the Commission or its processes, it's just that's

8 the way proposals go.

9 MR. TURK: We can't tell you how long

10 that process would take. There's a minimum comment

11 period, I believe that's -- I believe the comment

12 period is a 75-day period, but there's thought that

13 goes into it. There's receipt of the comments,

14 analysis of the comments. I would point out,

15 however, that Staff Exhibit U consists of several

16 documents, including a letter from Chairman Meserve

17 to the State of Utah, specifically Dr. Nielson and

18 to Mr. Silberg, advising them that there are two

19 separate tracks and one doesn't affect the other.

20 There's the PFS proceeding track and there's the

21 rule-making track, and you'll see that when we

22 introduce that exhibit.

23 JUDGE FARRAR: Oh, okay. What's the

24 date of that letter?

25 MR. TURK: November 27, 2001.

1 JUDGE FARRAR: Okay. All right, thank
2 you for that clarification. Go ahead,
3 Ms. Nakahara.
4 Q. (By Ms. Nakahara) Dr. Cornell, if
5 you'll look at your response to Question 22. And
6 you state the public should be provided comparable
7 levels for safety -- of safety for various societal
8 activities; correct?
9 A. Yes.
10 MS. NAKAHARA: Ms. Chancellor is
11 distributing what is now State's Exhibit 181, which
12 is a copy of appendix -- an Internet copy of
13 Appendix H entitled Dry Spent Fuel Storage Licenses
14 to NRC's Information Digest 2001, Edition
15 NUREG-1350 Volume 13.
16 (EXHIBIT-181 MARKED.)
17 Q. (By Ms. Nakahara) Dr. Cornell, I'll
18 represent my understanding that this table
19 represents licenses issued by the NRC for dry spent
20 fuel storage at the time that this digest was
21 published.
22 Isn't it true that the ISFSI act, Surry
23 1 and 2 for Virginia Electric & Power Company, was
24 not designed to a 2,000-year mean return period?
25 A. I do not know.

1 Q. If you know, isn't it true, that the
2 H. B. Robinson 2 Carolina Power & Light Company
3 ISFSI was not designed to a 2,000-year mean return
4 period?
5 A. I do not know.
6 Q. If you know --
7 JUDGE FARRAR: Wait, wait, can we
8 shorten this?
9 MS. NAKAHARA: Yes.
10 JUDGE FARRAR: Assuming your point is
11 eventually to establish that these were not
12 designed the way you want, let's ask Dr. Cornell,
13 do you know about the design of any of these?
14 MS. NAKAHARA: I do not know the
15 design-basis. I can speculate, but I do not know.
16 JUDGE FARRAR: Oh, we don't like that.
17 DR. CORNELL: And I know you don't.
18 JUDGE FARRAR: Does that satisfy
19 your --
20 MS. NAKAHARA: Just one more.
21 JUDGE FARRAR: Go ahead.
22 Q. (By Ms. Nakahara) So Dr. Cornell, you
23 are not aware of any dry cask storage ISFSI other
24 than the INEEL ISFSI for Three Mile Island 2 that
25 has been designed to a 2,000-year mean return

1 period; is that correct?

2 A. I'm aware of none, no.

3 Q. Thank you.

4 JUDGE FARRAR: Ms. Nakahara, did you

5 want to have this document admitted?

6 MS. NAKAHARA: Yes. Thank you for

7 reminding me, Your Honor.

8 JUDGE FARRAR: It looks like even though

9 this witness is unfamiliar, it looks like an

10 official Staff, NRC Staff publication. Mr. Turk,

11 any objection?

12 MR. TURK: Your Honor, I'm not familiar

13 with the document. I see that it's titled

14 Information Digest 2001 Edition. I don't know that

15 this document is probative of anything. Because

16 the witness is not familiar with the facility

17 listed, and therefore, no relevance has been shown

18 for the document. I would oppose its admission on

19 that ground.

20 JUDGE FARRAR: Mr. Gaukler.

21 MR. GAUKLER: I agree and I would oppose

22 it on the same ground.

23 JUDGE FARRAR: All right. Then why

24 don't we withhold admission at this time, but if --

25 well, no. This witness hasn't added anything to

1 the document. Do you want to argue against the
2 Applicant and Staff's claim of lack of relevance?
3 MS. NAKAHARA: Your Honor, Dr. Cornell
4 states in Answer 22 that the public should be
5 provided comparable levels of safety for various
6 societal activities. I would submit that a
7 comparable level of safety is required with other
8 license ISFSIs, and this merely illustrates the
9 number of licenses issued to various ISFSIs, and
10 that Dr. Cornell is unaware of the design-basis
11 earthquake in which these licenses were granted,
12 and it shows that he has not considered that in his
13 opinion.
14 JUDGE FARRAR: Okay. So even though
15 this document does not on its face show us how
16 these are designed --
17 MS. NAKAHARA: Yes.
18 JUDGE FARRAR: -- the relevance is it
19 shows even though he's looking for comparability,
20 he's unfamiliar with this, so is unable to speak to
21 the comparability. Mr. Turk.
22 MR. TURK: May I ask counsel again for
23 the answer in which he makes that statement?
24 JUDGE FARRAR: 22. Which appears to
25 talk about comparable levels of safety for societal

1 activities generally, and I take it Ms. Nakahara is
2 saying --
3 MS. NAKAHARA: At least.
4 JUDGE FARRAR: That may be true
5 generally, but here's one that's specific, and he
6 has not made that inquiry, you know, is the PFS
7 facility comparable to these other similar
8 facilities -- or other facilities.
9 MR. TURK: Your Honor, I believe the
10 statement in the testimony was rather general in
11 nature and it didn't go to comparison with other
12 ISFSIs. But in a very limited sense, I can see the
13 relevance, but I don't think it's -- I don't think
14 it's highly probative of anything, except as a
15 reference point.
16 JUDGE FARRAR: Mr. Gaukler.
17 MR. GAUKLER: I think he already
18 testified that he didn't know of the design-basis
19 of other ISFSIs, other than I guess the INEEL TMI 2
20 design-basis. So it doesn't really add anything to
21 the record.
22 (Judges conferred off the record.)
23 JUDGE FARRAR: The Board agrees this
24 does have relevance to some extent to the Answer
25 22, so we will overrule the objections and let the

1 State Exhibit 181 be admitted.

2 MS. NAKAHARA: Thank you, Your Honor.

3 (EXHIBIT-181 ADMITTED.)

4 MS. NAKAHARA: Ms. Chancellor has or is

5 in the process of distributing what I would like to

6 mark as State's Exhibit 182. It is a very brief

7 portion of the Applicant's brief on the Regulatory

8 Standard for Aircraft Crash Hazards at the Private

9 Fuel Storage Facility which was submitted to the

10 NRC Commission and includes a table of contents and

11 pages six and -- something happened to my page

12 numbering. Six and seven. And I modified Page 7

13 to remove the extra discussion.

14 JUDGE FARRAR: And you want this marked

15 as an exhibit?

16 MS. NAKAHARA: Yes, state's Exhibit 182.

17 JUDGE FARRAR: And I suppose even though

18 this is a part of the record in the proceeding,

19 it's not a part of the record in this little part

20 of the proceeding, so we should go ahead and the

21 reporter will mark this as State's 182.

22 (EXHIBIT-182 MARKED.)

23 JUDGE FARRAR: Go ahead, Ms. Nakahara.

24 MS. NAKAHARA: Thank you.

25 Q. (By Ms. Nakahara) Dr. Cornell, I would

1 not expect you to be familiar with this document
2 since it was submitted in another proceeding.
3 However, I would direct you to Page 6. And to
4 avoid the criticism of Mr. Travieso-Diaz in
5 counting lines, I hope I'm correct, if you look on
6 the eighth line from the bottom --
7 MR. TRAVIESO-DIAZ: Please don't use me
8 as a standup for that.
9 Q. (By Ms. Nakahara) PFS in its argument
10 states, "In doing so, the Commission intended to
11 make the design-basis for Part 60 repositories
12 comparable to that for Part 72 facilities."
13 Do you see that line?
14 A. Yes, I do.
15 Q. The above ground operations at Yucca
16 Mountain are not designed to a 2,000-year mean
17 return period; isn't that correct?
18 MR. TURK: I'm sorry, could I hear the
19 question again.
20 MS. NAKAHARA: Let me try and connect
21 that a little better.
22 JUDGE FARRAR: You were reading from the
23 document, but then your question was not read from
24 the document.
25 MS. NAKAHARA: Well, let me try and

1 connect it.

2 JUDGE FARRAR: No, no, that's all right,

3 let her read it back.

4 (The record was read as follows: "The

5 above ground operations at Yucca Mountain are

6 not designed to a 2,000-year mean return

7 period; isn't that correct?")

8 MR. TURK: Your Honor, I would object to

9 the question on two grounds. One, I'm not aware

10 there's a design for the Yucca Mountain above

11 ground facility. I'm not sure what she's referring

12 to, whether it's a preliminary design or something

13 else. Secondly, this is a legal brief, it is not

14 evidence, it is not a commission statement. And I

15 would object to it on the grounds that it's not

16 evidence. And third, I believe that there's been

17 some development with respect to Part 60 insofar as

18 Part 63 has now been issued.

19 JUDGE FARRAR: You said there's been

20 some what with respect to Part 60?

21 MR. TURK: A legal development with

22 respect to Part 60, and that we now have Part 63.

23 JUDGE FARRAR: Even if all of what you

24 just said is true, the question in a sense is not

25 connected to this brief. The question is a factual

one about Yucca Mountain, as I took it.

MS. NAKAHARA: Yes. Let me try and connect them a little better.

JUDGE FARRAR: I don't know if you need to connect them so much as restate the question about Yucca Mountain more directly.

Q. (By Ms. Nakahara) Given that PFS has taken the position in the past --

JUDGE FARRAR: No, forget PFS. You said let's find out about -- I think you've got a good question about Yucca Mountain. Depending on the answer, we'll see where we go with that in light of the brief and in light of Mr. Turk -- the developments Mr. Turk had mentioned. Let's just ask him about Yucca Mountain.

Q. (By Ms. Nakahara) Dr. Cornell, based on your understanding, will Yucca Mountain -- the above ground operations at Yucca Mountain be designed to a 2,000-year mean return period?

A. Yucca Mountain is a complicated situation, as we know. It's ongoing. It's my understanding that the Department of Energy, as the Applicant in that case, has proposed seismic design criteria of a graded type, of 1,000 years for some components and 10,000 years for others. And I

1 believe the NRC accepted that proposal.

2 JUDGE FARRAR: Do you know enough about

3 it to tell us about those different components?

4 Ms. Nakahara asked about the above ground. I take

5 it, Ms. Nakahara, the above ground, you mean there

6 will be fuel -- the canisters or transportation

7 casks would come in above ground?

8 MS. NAKAHARA: Yes.

9 JUDGE FARRAR: And there would have to

10 be something done to them before they're sent below

11 ground. Are you familiar enough with the design --

12 proposed design of Yucca Mountain to speak to this?

13 DR. CORNELL: I'm familiar enough to

14 know that there's a very substantial difference in

15 what happens at that transfer operation. The fuel

16 rods come into Yucca Mountain from the various --

17 proposed, come in from Yucca Mountain from the

18 various power plants, and then they are packaged.

19 The transportation casks are opened and the fuel

20 assemblies taken and put and stored in various

21 storage casks.

22 JUDGE FARRAR: That's done above --

23 DR. CORNELL: That's done in this

24 above-ground transfer of building operation. So in

25 contrast to the situation at PFS where the

canisters are not open, this packaging of the fuel -- of the fuel is done at the site of the power plant and then put into the canisters, and the canisters are switched without being opened.

JUDGE FARRAR: So at Yucca Mountain, do I understand you to say that fuel assemblies would be removed from what we called the multipurpose canisters?

DR. CORNELL: Yes.

JUDGE FARRAR: And put in --

DR. CORNELL: They're equivalent, yes.

JUDGE FARRAR: And put in something different that would then go underground?

DR. CORNELL: Correct.

JUDGE FARRAR: Okay. And I take it, it's that above-ground transfer operation that Ms. Nakahara was asking you about?

Q. (By Ms. Nakahara) Yes. Are you aware of the design-basis earthquake standard proposed by DOE for that transfer facility at Yucca Mountain?

A. I stated in my previous answer to the degree I'm aware of it, the numbers are 1,000 years and 10,000 years for the two types of -- two categories of SSCs.

JUDGE FARRAR: But the above ground

one --

DR. CORNELL: The above ground is that -- the primary part of that is that transfer building.

JUDGE FARRAR: And would that be 1,000 or 10,000?

DR. CORNELL: Some elements would be 1,000 and some would be 10,000.

Q. (By Ms. Nakahara) Thank you.

MS. NAKAHARA: And I will not move for the admittance of this exhibit to move this along.

MR. TURK: This is just 182 for identification only?

MS. NAKAHARA: Yes.

MR. TURK: Thank you.

Q. (By Ms. Nakahara) Dr. Cornell, in your response to Question 30, is it generally your opinion -- not generally. Is it your opinion that the design acceptance criteria and procedures guidelines provided by NRC SRPs contain many conservatisms that result in risk reduction factors in the range of 5 to 20?

A. Yes, that's my position.

Q. Your opinion is based on SRPs for nuclear power plants; correct?

1 A. It's based on SRPs for nuclear power
2 plants and the similarity between the seismic
3 criteria in those and those for ISFSIs.
4 Q. Is the key to qualitatively obtaining a
5 risk reduction factor in the range of 5 to 20, is a
6 quantitative assessment that the nuclear power
7 plants are designed to meet acceptance criteria
8 specified in the SRPs? Is that -- let me try that
9 again.
10 A. Yes.
11 Q. Is one of the bases for your deriving
12 that the SRPs for nuclear power plants relate to a
13 range of 5 to -- a risk reduction factor in the
14 range of 5 to 20 is one of the basis for your
15 arriving at that opinion, is that nuclear power
16 plants are designed to meet the acceptance criteria
17 specified in the applicable SRPs?
18 A. Yes.
19 Q. And you cannot ensure that a nuclear
20 power plant will have a risk reduction ratio of 5
21 to 20 if the acceptance criteria is not satisfied;
22 isn't that true?
23 A. Not based on that argument, I couldn't,
24 but perhaps based on other arguments.
25 Q. And other arguments being a specific

1 analysis for the SSC in question?

2 A. A specific analysis which would

3 accumulate, let's say, an estimate of the margin.

4 Q. In Answer 37, you state that the seismic

5 elements of the ISFSIs -- and I'll wait. In Answer

6 37, you state that the seismic elements of the

7 ISFSIs and nuclear power plant SRPs are similar;

8 correct?

9 A. Yes, I do.

10 Q. And you also state that you're familiar

11 with the testimony of Dr. Krishna Singh, is that

12 correct, filed in this proceeding?

13 A. Yes, I have looked at that.

14 Q. Dr. Singh states that the HI-STORM

15 canister is designed and constructed with the same

16 ASME or American Society of Mechanical Engineers

17 code used for reactor pressure vessels and primary

18 coolant system piping. Do you recall that

19 statement?

20 A. Not exactly those words, but I remember

21 that concept, yes.

22 Q. Reactor vessels are rigidly connected to

23 the foundation; isn't that true?

24 A. Yes.

25 Q. Reactor vessels are not allowed to

1 freely slide, rotate or uplift under seismic
2 forces; correct?
3 A. Correct.
4 Q. Similarly, primary coolant system piping
5 are not allowed to freely slide, rotate or uplift
6 under seismic forces; correct?
7 A. Correct.
8 Q. In Answer 33 of your testimony, you
9 state a freestanding storage cask is not typical of
10 commercial nuclear power plants; correct?
11 A. Correct.
12 Q. And that the level of conservatism for
13 HI-STORM 100 storage casks must be demonstrated by
14 specific analysis; correct? I'm not saying you
15 state that.
16 A. I don't see those words, no, but I --
17 Q. Do you agree with that?
18 A. I agree with the notion that I cannot
19 apply the similarity argument to the casks.
20 Q. In your answer to Question 38, does your
21 answer imply that your opinion -- that the storage
22 casks has a risk reduction level in the range of 5
23 to 20 relies on the testimony filed by Drs. Alan
24 Soler and Dr. Krishna Singh?
25 A. Yes, it does.

1 Q. Are you familiar with the seismic
2 analysis of the casks discussed in their testimony
3 filed in this matter?
4 A. I have some familiarity with it, yes.
5 Q. You also state in Answer 38 that they
6 have stated they generally use the same design
7 criteria and procedures applicable to nuclear power
8 plants; correct?
9 A. Yes, I did.
10 Q. What do you mean by the term generally?
11 A. Well, I mean that, for example, when
12 assessing the behavior of the casks, let's say, in
13 their fixed condition, the inertial effects on
14 those casks, the seismic induced forces in those
15 casks are assessed in the same way that one would
16 do that for 0800. When one goes further to look at
17 their behavior, in terms of their stability,
18 sliding and tipping stability, that gets into a
19 range which is beyond that or outside of that
20 normally encountered in the assessment of nuclear
21 power plant components. But, for example, they use
22 the same ASME stress and allowable criteria when
23 evaluating the effects of tipover accidents, et
24 cetera.
25 Q. So is it fair to say you would not use

1 generally, in terms of whether Holtec complied with
2 some aspects of the acceptance criteria for the
3 SRPs or all aspects of acceptance criteria?
4 MR. GAUKLER: Objection, the question is
5 vague and ambiguous.
6 Q. (By Ms. Nakahara) I'm trying to
7 understand your use of the term generally, so let
8 me -- you wouldn't use your term as you used it in
9 Answer 38, to refer to whether Holtec only complied
10 with some aspects of the acceptance criteria --
11 applicable acceptance criteria versus all
12 applicable acceptance criteria?
13 A. To my knowledge, every time they
14 encountered a situation where they needed, for
15 example, stress criteria and so on, where there
16 were analogous criteria in the SRPs for ISFSIs and
17 the SRPs for nuclear power plants, they applied the
18 same kinds of criteria that we would use for
19 nuclear power plants. As I said earlier, there are
20 elements that appear in the -- for example,
21 design-basis accidents that appear in the ISFSIs
22 for dry storage casks, such as the hypothetical
23 drop accident and the hypothetical tipover
24 accident, that they would not find in the nuclear
25 power plant 0800. So they in that sense did not --

1 they did not find there what they needed for those
2 conditions.
3 Q. Do you agree that Holtec's cask
4 stability analysis is dependent upon a computer
5 model, a nonlinear analysis computer model that
6 mathematically projects the cask behavior?
7 A. Their analysis of the design-basis
8 earthquake includes a computer analysis that is,
9 say, mildly nonlinear.
10 Q. And to follow up a little bit on Judge
11 Lam's questions --
12 JUDGE FARRAR: What was that word, it
13 was what not --
14 DR. CORNELL: At the design-basis
15 earthquake, the analysis shows that there's very
16 little sliding and very little tipping. Prior to
17 that sliding or tipping, there was -- I don't
18 believe there's any nonlinearity in the cask model,
19 the pad model. There may be some in the soil
20 springs.
21 Q. (By Ms. Nakahara) Following up a little
22 bit on Judge Lam's questions about uncertainties,
23 have you evaluated the uncertainties in determining
24 the failure probability of the cask as analyzed by
25 Holtec?

1 A. I have not quantified those
2 uncertainties, no.
3 Q. Are you familiar with Holtec's analysis
4 entitled Multi Cask Response at PFS ISFSI from
5 2,000-year Seismic Event Rev.2, which was dated
6 August 20th, 2001?
7 MS. NAKAHARA: I'm assuming that if
8 counsel for PFS has no problem with the proprietary
9 nature, we'd be glad to show Dr. Cornell a copy.
10 MR. GAUKLER: That's already been
11 admitted into the record, correct?
12 MS. NAKAHARA: Yes.
13 MR. GAUKLER: Fine.
14 JUDGE FARRAR: Ms. Nakahara, these are
15 one of those proprietary documents we got?
16 MS. NAKAHARA: Yes.
17 JUDGE FARRAR: Mr. Gaukler, is
18 Mr. Cornell authorized, sworn, whatever? We don't
19 want to breach any proprietary issues here.
20 MR. GAUKLER: We made arrangements for
21 that previously.
22 JUDGE FARRAR: Oh, okay, fine. Is there
23 a question pending?
24 DR. CORNELL: Is there a question?
25 Q. (By Ms. Nakahara) Are you familiar with

1 this document?

2 A. Yes, I've read this document before.

3 Q. Did you agree with the methodology used

4 in this document?

5 A. Yes, in general, to the best of my

6 ability to evaluate it.

7 Q. Did you agree with the input parameter

8 values selected for this document, this analysis?

9 A. I certainly agree with the seismic input

10 values and the properties with which I'm familiar,

11 such as the material properties of the concrete

12 steel. There are others such that I am probably

13 less familiar with because I have not been involved

14 directly in these kind of analyses.

15 Q. And are you aware that no experimental

16 test data is available to support the results of

17 this analysis?

18 A. I think you have to qualify that for me.

19 There's certainly experiments have been done to

20 evaluate nonlinear analyses of soils, of pads and

21 soils and so on.

22 Q. For example, there is no experimental

23 test data to support the input parameters used by

24 Holtec for this analysis to estimate the cask

25 response?

1 A. I'm not sure if that's been testified
2 to. I'm not aware of such testing in any case.
3 Q. Do you believe it's important to obtain
4 SHAKE table data to support input parameters for a
5 nonlinear analysis in the absence of any real
6 physical data?
7 MR. GAUKLER: Is that a hypothetical
8 question?
9 MS. NAKAHARA: Yes, hypothetically.
10 DR. CORNELL: No, generally not. We do
11 lots of nonlinear analyses in advanced earthquake
12 engineering. We have done some SHAKE table
13 testing, and the SHAKE table testing is used to
14 verify the general nature of those models. But
15 it's not used for each and every application.
16 Q. (By Ms. Nakahara) And just quickly, are
17 you familiar with Holtec's analysis entitled PFS
18 Site Specific HI-STORM Drop Tipover Analysis
19 Revision 2 dated October 31st, 2001, which
20 Ms. Chancellor will show you a copy?
21 MR. GAUKLER: What document is this?
22 MS. NAKAHARA: Pardon me, the drop
23 tipover analysis.
24 MR. TURK: Just for clarification, does
25 this have an exhibit number on it?