UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

June 22, 2010

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador **/RA/**
Technical Secretary, ACRS

SUBJECT: CERTIFICATION OF THE MEETING MINUTES FROM THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS 554th FULL COMMITTEE MEETING HELD ON JULY 9-11, 2008 IN ROCKVILLE, MARYLAND

The minutes of the subject meeting were certified on September 11, 2008 as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment:
As stated

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

September 11, 2008

MEMORANDUM TO: Sherry Meador, Technical Secretary
Advisory Committee on Reactor Safeguards

FROM: Cayetano Santos, Chief **/RA/**
Reactor Safety Branch
Advisory Committee on Reactor Safeguards

SUBJECT: MINUTES OF THE 554th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), July 9-11, 2008

I certify that based on my review of the minutes from the 554th ACRS Full Committee meeting, and to the best of my knowledge and belief, I have observed no substantive errors or omissions in the record of this proceeding subject to the comments noted below.

OFFICE	ACRS	ACRS:RSB
NAME	SMeador	CSantos/sam
DATE	09/11/08	09/11/08

OFFICIAL RECORD COPY

CERTIFIED

Date Issued: 09/03/2008
Date Certified: 09/11/2008

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MINUTES OF THE 554th ACRS MEETING

July 9-11, 2008

APPENDICES

During its 554th meeting, July 9-11, 2008, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports, letters, and memoranda

REPORTS

Reports to Dale E. Klein, Chairman, NRC, from William J. Shack, Chairman, ACRS:

- Security and Aircraft Impact Rulemaking for Nuclear Power Plants, dated July 18, 2008

- Stretch Power Uprate Application for the Millstone Power Station, Unit 3, dated July 23, 2008

LETTERS

Letters to R. W. Borchardt, Executive Director for Operations, NRC, from William J. Shack, Chairman, ACRS:

- Interim Letter 4: Chapter 3 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Certification of the ESBWR Design, dated July 21, 2008

MEMORANDA

Memoranda to R. W. Borchardt, Executive Director for Operations, NRC, from Frank P. Gillespie, Executive Director, ACRS:

- Draft Regulatory Guides 1149 and 1189, dated July 15, 2008

MINUTES OF THE 554th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
July 9-11, 2008
ROCKVILLE, MARYLAND

The 554th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on July 9-11, 2008. Notice of this meeting was published in the *Federal Register* on June 20, 2008 (72 FR 35172-35173) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting schedule and outline (Appendix II). The meeting was open to public attendance.

A transcript of selected portions of the meeting is available in the NRC's Public Document Room at One White Flint North, Room 1F-19, 11555 Rockville Pike, Rockville, Maryland. Copies of the transcript are available for purchase from Neal R. Gross and Co., Inc., 1323 Rhode Island Avenue, NW, Washington, DC 20005. Transcripts are also available at no cost to download from, or review on, the Internet at http://www.nrc.gov/ACRS/ACNW.

ATTENDEES

ACRS Members: Dr. William J. Shack (Chairman), Dr. Mario V. Bonaca (Vice-Chairman), Dr. Said Abdel-Khalik (Member-at-Large), Dr. George E. Apostolakis, Dr. Sam Armijo, Dr. Sanjoy Banerjee, Dr. Dennis Bley, Mr. Charles Brown, Dr. Michael Corradini, Mr. Otto L. Maynard, Dr. Dana A. Powers, Mr. Harold Ray, Dr. Michael Ryan, Mr. John Sieber, and Mr. John Stetkar. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

[Note: Mr. Sam Duraiswamy was the Designated Federal Official for this portion of the meeting.]

Dr. William J. Shack, Committee Chairman, convened the meeting at 8:30 a.m. In his opening remarks he announced that the meeting was being conducted in accordance with the provisions of the Federal Advisory Committee Act. He reviewed the agenda items for discussion and noted that no written comments or requests for time to make oral statements from members of the public had been received. Dr. Shack also noted that a transcript of the open portions of the meeting was being kept and speakers were requested to identify themselves and speak with clarity and volume. Dr. Shack welcomed Mr. Harold Ray and Dr. Michael Ryan as new official members and stated that the Committee was now at its statutory strength of 15 members.

II. Stretch Power Uprate Application for Millstone Power Station

[Note: Mr. David Bessette was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff, Dominion Nuclear Connecticut, Inc., and members of the public to discuss Dominion's license amendment request to increase the power level of Millstone Unit 3 by 7%. Topics of discussion included fuel system and nuclear design as well as containment and design basis accident analyses. Also discussed were proposed modifications to the plant to support the increased power level such as changes to the core design and disabling automatic control rod withdrawal. The plant safety analyses have been performed in many cases, with more modern methods than when the plant was initially licensed. These analyses show substantial margins to licensing limits for containment design pressure, peak cladding temperature, and departure from nucleate boiling. Mr. Gunderson of the Citizens Against Millstone described a concern that the containment design pressure could be exceeded. The Committee issued a report to the NRC Chairman on this matter dated July 23, 2008, recommending that the application for power uprate at Millstone Unit 3 be approved.

III. Selected Chapters of the Safety Evaluation Report (SER) Associated with the Economic Simplified Boiling Water Reactor (ESBWR) Design Certification Application

[Note: Mr. Harold Vandermolen was the Designated Federal Office for this portion of the meeting.]

The Committee met with representatives of the NRC staff and General Electric-Hitachi Nuclear Energy to discuss Chapter 3, "Design of Structures, Components, Equipment, and Systems," of the NRC Staff's SER with Open Items related to the ESBWR Design Certification Application. The discussion focused on classification of Structures, Systems, and Components (SSCs) and the seismic analysis. The SSCs are classified as Safety Class 1, 2, 3, or N depending on whether the SSC is needed to preserve the integrity of the reactor coolant pressure boundary, to shut down the reactor and maintain it in a safe shutdown condition, or to prevent or mitigate potential offsite exposures. Thus, the reactor coolant pressure boundary components and supports are classified as Safety Class 1 whereas nonsafety-related SSCs are classified as Class N. Safety Classes 1 through 3 are very closely related to Quality Groups A through C. The quality groups are defined in terms of their pressure retaining functions. Pressure retaining components of the reactor coolant pressure boundary are Quality Group A. Finally, there is a seismic classification; all safety related SSCs are placed in Seismic Category I, which means they must remain functional in the event of a design basis earthquake. Nonsafety-related SSCs may be placed in Seismic Category II, which means that they need not remain functional, but must not fail in such a way as to interfere with safety-related SSCs. The remaining nonsafety-related SSCs may be assigned to Seismic Category NS, which means that they must conform to the International Building Code but have no further seismic design requirements. Regarding the seismic design, the Combined Seismic Design Response Spectra (CSDRS) are based on Regulatory Guide 1.60 spectra with the addition of the North Anna site-specific spectra at high frequencies, i.e., the CSDRS is the envelope of the generic and North Anna spectra.

North Anna is representative of most severe rock sites in the eastern US, and thus the CSDRS envelopes most candidate sites with considerable conservatism. The fluids in the reactor building pools are modeled as a mass-spring (sloshing) component and an impulsive (rigid) component. However, for conservatism, the entire water mass of each pool is considered as an impulsive mass in the seismic stick model for predicting overall building response. The sloshing component generally responds at very low frequencies (below 0.5 Hz), where no structural modes of vibration exist. The seismic loads used in the stress analysis of pool structures include both the global loads calculated from the seismic response analysis and local hydrodynamic pressure loading on the pool boundaries.

The Committee issued a letter to the EDO on this matter, dated July 21, 2008, stating that the evolving nature of the ESBWR design makes it difficult to perform an effective review, and that additional information is needed to demonstrate that dynamic forces from seismic events are treated properly in the analyses of heat exchangers immersed in elevated water pools.

IV. Safeguards and Security Matters

[Note: Ms. Maitri Banerjee was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff and a member of the pubic to discuss the draft final rules on security and aircraft impact assessment. Consistent with the Commission direction in the October 31, 2003, Staff Requirements Memorandum, the Committee did not review the elements of the security rule that dealt with physical security. The ACRS review was limited to three parts of the rule: (i) 10 CFR 50.54(hh), “Mitigative Strategies and Response Procedures for Potential or Actual Aircraft Attack;” (ii) 10 CFR 73.54 “Protection of Digital Computer and Communication Systems and Networks;” and (iii) 10 CFR 73.58, “Safety/Security Interface Requirements for Nuclear Power Reactors.” The Committee also reviewed the draft final rule, “Consideration of Aircraft Impacts for New Nuclear Power Reactor Designs.” The staff discussed the essential elements of each rule, how comments from the public were addressed, and the status of the associated regulatory guidance. Mr. James Riccio of Greenpeace stated that the aircraft impact rule, in his opinion, lacks substantive acceptance criteria, and the requirements that the containment remains intact may not be sufficient.

The Committee issued a report to the NRC Chairman on this matter, dated July 18, 2008, recommending that the draft final rules be approved. The committee agreed with the staff that it is appropriate to treat aircraft attacks as beyond-design-basis events.

V. Status of NRC Staff Activities Associated with Seismic Design Issues at Nuclear Power Plants

[Note: Mr. Mike Lee was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff to discuss the evaluation of earthquakes and how that information will be used to evaluate the safety of nuclear power reactors. The staff's presentation focused on four topics: (1) NRC Seismic Research Program Plan; (2) Generic Safety Issue-199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern U.S. on Existing Nuclear Power Plants;" (3) Interim Staff Guidance on Seismic Issues Associated with High Frequency (HF) Ground Motion; and (4) the July 2007 Earthquake at the Kashiwazaki-Kariwa (KK) Nuclear Power Plant Site. The NRC staff issued a Seismic Research Program Plan in January 2008 that identifies approximately 40 research projects in the areas of (1) Earth Science and Natural Hazards Research; (2) Earthquake Engineering Analysis and Earthquake Resistant Design; (3) Cooperation in Ongoing International Research Activities; and (4) Updates to NRC Regulatory Guides. This plan will be implemented over the next three years.

In a 2004 analysis prepared for the NRC, the U.S. Geological Survey reported that the Peak Ground Acceleration (PGA) reference probability for the 29 Central Eastern United States (CEUS) nuclear power reactor sites had increased. As a result, the probability of exceeding the Safe Shutdown Earthquake at some nuclear power plants east of the Mississippi River is now believed to be higher than previously understood. The NRC staff is reviewing documentation to better understand what seismic margin currently exists for the current fleet of nuclear power plants and will then evaluate the new PGA estimates against those margins. The NRC staff recently learned that for some current and future nuclear power plant sites, the site specific ground motion may exceed the ground motion derived from a Lawrence Livermore National Laboratory- or Electric Power Research Institutes-based Probabilistic Seismic Hazard Analysis.

On July 16, 2007, a magnitude 6.8 earthquake occurred about 9 kilometers offshore of the KK nuclear power plant site. PGAs as high as 0.69g were recorded at the bases of some of the reactor buildings, and PGAs at the tops of some building roofs were reported to be twice as high. In the U.S. the largest PGA assumed in reactor designs is 0.3g. The PGA associated with this earthquake was 2 to 2.5 times greater than the acceptable earthquake design for the KK site. Although SSCs important to safety appeared to have performed well, there were several incidents involving noncritical SSCs at the KK site. Inspections for all seven reactor units were completed and no abnormalities were found that could impact the functional or structural integrity of the reactor units. All seven reactor units at the KK site remain off-line while additional site inspections and assessments take place. This was an information briefing. Future subcommittee meetings will get into more details related to seismic issues.

V. Containment Overpressure Credit

[Note: Mr. Harold Vandermolen was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff and the Tennessee Valley Authority (TVA) to discuss technical issues related to crediting of containment overpressure during design basis accidents and special events in support of the extended power uprate for Browns Ferry Units 1, 2, and 3. Representatives of TVA described how credit for containment overpressure is part of the current licensing basis for Appendix R and Loss-of-Coolant-Accident requirements. Of the two, the Appendix R is the more limiting. During a postulated fire event in two specific locations, if all of the equipment in these locations is rendered inoperable by the fire, there will not be sufficient net positive suction head for the residual heat removal pumps if the containment overpressure resulting from primary system blowdown is not credited. However, TVA claims that this assumption, when taken together with other licensing basis assumptions, is overly conservative. Moreover, based on discussions with the pump vendor, TVA claimed that there is a high likelihood that the pumps would survive the period of low suction head. Several members expressed an interest in the pump data and how they were obtained, since the pump tests were performed when the pumps were new in the 1970s and the pump rotors were replaced in the 1990s. In addition, the fire hazard analysis is a deterministic analysis; the licensee does not have a fire Probabilistic Risk Assessment. Several Members discussed the lack of any means for quantifying the degree of conservatism claimed by TVA. This was an information briefing. No Committee action was necessary.

VI. Executive Session

[Note: Mr. Frank Gillespie was the Designated Federal Official for this portion of the meeting.]

A. Reconciliation of ACRS Comments and Recommendations/EDO Commitments

- The Committee considered the EDO's response of June 20, 2008, to conclusions and recommendations included in the April 30, 2008, ACRS report on the Draft NUREG-1902, "Next generation Nuclear Plant Licensing Strategy Report." The Committee decided that it was satisfied with the EDO's response.

B. Report of the Planning and Procedures Subcommittee Meeting

Review of the Member Assignments and Priorities for ACRS Reports and Letters for the July ACRS Meeting

Member assignments and priorities for ACRS reports and letters for the July ACRS meeting were discussed. Reports and letters that would benefit from additional consideration at a future ACRS meeting were also discussed.

Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through October 2008 was discussed. The objectives were:

- Review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate
- Manage the members' workload for these meetings
- Plan and schedule items for ACRS discussion of topical and emerging issues

Staff Requirements Memorandum Resulting from the ACRS Meeting with the Commission

In a Staff Requirements Memorandum (SRM) dated June 26, 2008 which resulted from the ACRS meeting with the Commission on June 5, 2008, the Commission stated the following:

- At the next Commission briefing on digital I&C, the staff should report the progress made with respect to identifying and analyzing digital I&C failure modes, and discuss the feasibility of applying failure mode analysis to the quantification of risk associated with digital I&C.

- The staff should continue working to address Committee concerns, such as SOARCA, digital I&C, and containment overpressurization, and, as necessary and appropriate, provide timely policy decision papers to the Commission to resolve any disagreements.

- Direction to the staff regarding SOARCA will be provided in the SRM for SECY-08-0029, "State-of-the-Art Reactor Consequence Analysis — Reporting Offsite Health Consequences," which is currently before the Commission for voting.

Status of the Quality Assessment of Selected NRC Research Projects

The Committee is in the process of assessing the quality of the following NRC research projects:

- FRAPCON/FRAPTRAN Code work at the Pacific Northwest National Laboratory and
- NUREG-6948, "Study of Remote Visual Methods to Detect Cracking in Reactor Components"

The Panel Chairmen provided a brief report on their preliminary findings.

Visit to the Braidwood Nuclear Plant and Meeting with the Region III Administrator

Several members of the Committee plan to visit the Braidwood Nuclear Plant on July 23 and meet with the Regional Administrator on July 24, 2008. An itinerary and logistics for the plant visit and meeting with the Regional Administrator were discussed.

Proposed Regulatory Guides,

- DG-1149, Qualification of Safety Related Motor Control Centers for Nuclear Power Plants"

  DG-1149 is a new Regulatory Guide, which describes a method for qualification of safety-related motor control centers for nuclear power plants. This Guide endorses, with certain exceptions, IEEE Standard 649-2006, "Standard for Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations." This Standard provides the basic principles, requirements, and methods for qualifying safety-related motor control centers for applications in both harsh and mild environments in nuclear power plants.

  The staff plans to issue DG-1149 for public comment and would like to know whether the Committee wants to review this Guide prior to being issued for public comment.

- Proposed Revision 2 to Regulatory Guide 1.126, (DG-1189), "An Acceptable Model and Related Statistical Methods for Analysis of Fuel Densification"

  This Guide describes an analytical model and related assumptions and procedures for predicting the effects of fuel densification in LWR plants. To meet these objectives, this Guide describes statistical methods related to product sampling that will ensure that this and other approved analytical models will adequately describe the effects of densification for each initial core and reload fuel quantity produced.

  Revision 1 to Regulatory Guide 1.126 was issued in 1978. The proposed revision 2 includes more recent information on in-reactor densification. There are no substantive changes to the existing technical guidance.

  The staff plans to issue DG-1189 for public comment and would like to know whether the Committee wants to review this Guide prior to being issued for public comment.

  Visit to the US-APWR Simulation Facility

During its June 2008 meeting, Mitsubishi Heavy Industries, LTD (MHI) provided an overview of the US-Advanced Pressurized Water Reactor (US – APWR) design. During that meeting, MHI invited interested ACRS members to visit a simulation facility related to US-APWR in Pittsburgh, Pennsylvania.

The meeting was adjourned at 1:00 p.m. on July 11, 2008.

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards (ACRS); Subcommittee Meeting on Planning and Procedures; Notice of Meeting

The ACRS Subcommittee on Planning and Procedures will hold a meeting on June 3, 2008, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed pursuant to 5 U.S.C. 552b (c) (2) and (6) to discuss organizational and personnel matters that relate solely to the internal personnel rules and practices of the ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows:

Tuesday, June 3, 2008, 8 a.m. Until 9 a.m.

The Subcommittee will discuss proposed ACRS activities and related matters. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Officer, Mr. Sam Duraiswamy (telephone: 301–415–7364) between 7:30 a.m. and 4 p.m. (ET) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public. Detailed procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 26, 2007 (72 FR 54695).

Further information regarding this meeting can be obtained by contacting the Designated Federal Officer between 7:30 a.m. and 4 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes in the agenda.

Dated: May 13, 2008.

Cayetano Santos,

Chief, Reactor Safety Branch.

[FR Doc. E8–11230 Filed 5–19–08; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting Notice

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on June 4–6, 2008, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Monday, October 22, 2007 (72 FR 59574).

Wednesday, June 4, 2008, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 a.m.–10 a.m.: ARTIST Test Program (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the findings from the ARTIST Tests on aerosol retention in the secondary side of a steam generator, and related matters.

10:15 a.m.–11:45 a.m.: Risk Assessment Standardization Project (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the Risk Assessment Standardization Project (RASP) and related matters.

1:45 p.m.–3:45 p.m.: Overview of the U.S. Evolutionary Power Reactor (EPR) Design (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and AREVA Nuclear Power Inc., regarding design features of the EPR and related matters.

4 p.m.–5 p.m.: Status of the Development of Rules and Regulatory Guidance in the areas of Safeguards and Security (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the status of activities associated with the development of rules and regulatory guidance in the safeguards and security areas.

5 p.m.–5:30 p.m.: Status of the Quality Assessment of Selected Research Projects (Open)—The Committee will hold discussions with the Chairmen of the ACRS Panels regarding the status of the quality assessment of the research projects on: FRAPCON/FRAPTRAN Code work at the Pacific Northwest National Laboratory; and NUREG–6943, ''Study of Remote Visual Methods to Detect Cracking in Reactor Components.''

5:45 p.m.–7 p.m.: Preparation of ACRS Report (Open)—The Committee will prepare and discuss the proposed ACRS report on the ARTIST Test Program.

Thursday, June 5, 2008, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 a.m.–9:30 a.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings. It will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

9:30 a.m.–9:45 a.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

10 a.m.–11:15 a.m.: Preparation for Meeting with the Commission (Open)—The Committee will hold discussions in preparation for their meeting with the Commission on the following topics: Safety Research Program Report, Digital I&C Matters, State-of-the-Art Reactor Consequence Analysis Program, ESBWR Design Certification, and Extended Power Uprates and related Technical Issues.

1:30 p.m.–3:30 p.m.: Meeting with the Commission (Open)—The Committee will meet with the Commission to discuss topics noted above.

3:45 p.m.–6 p.m.: Preparation of ACRS Report (Open)—The Committee will continue its discussion of a proposed ACRS report on the ARTIST Test Program.

Friday June 6, 2008, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 a.m.–10:30 a.m.: Overview of the US-Advanced Pressurized Water Reactor (US-APWR) Design (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Mitsubishi Heavy Industries, Ltd., regarding design features of the US-APWR and related matters.

10:45 a.m.–11:45 a.m.: Status of NRC Staff Activities Associated with the Resolution of Generic Safety Issue (GSI)-191, ''Assessment of Debris Accumulation on Pressurized-Water Reactor (PWR) Sump Performance'' (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the status of NRC staff activities associated with the resolution of GSI–191.

1:15 p.m.–1:30 p.m.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 26, 2007 (72 FR 54695). In accordance with those procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting. Persons desiring to make oral statements should notify the Cognizant ACRS staff named below five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Cognizant ACRS staff prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Cognizant ACRS staff if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, as well as the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Girija S. Shukla, Cognizant ACRS staff (301–415–6855), between 7:30 a.m. and 4 p.m., (ET). ACRS meeting agenda, meeting transcripts, and letter reports are available through the NRC Public Document Room at *pdr@nrc.gov*, or by calling the PDR at 1–800–397–4209, or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS) which is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html or http://www.nrc.gov/reading-rm/doc-collections/ACRS/.

Video teleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician (301–415–8066), between 7:30 a.m. and 3:45 p.m., (ET), at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment and facilities that they use to establish the video teleconferencing link. The availability of video teleconferencing services is not guaranteed.

Dated: May 14, 2008.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E8–11232 Filed 5–19–08; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Notice

AGENCY HOLDING THE MEETINGS: Nuclear Regulatory Commission.

DATES: Weeks of May 19, 26, June 2, 9, 16, 23, 2008.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

Week of May 19, 2008

There are no meetings scheduled for the Week of May 19, 2008.

Week of May 26, 2008—Tentative

Tuesday, May 27, 2008

1:30 p.m.—NRC All Hands Meeting (Public Meeting), Marriott Bethesda North Hotel, 5701 Marinelli Road, Rockville, MD 20852.

Wednesday, May 28, 2008

9:30 a.m.—Briefing on Equal Employment Opportunity (EEO) and Workforce Planning (Public Meeting) (Contact: Kristin Davis, 301–492–2266).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

Week of June 2, 2008—Tentative

Wednesday, June 4, 2008

9 a.m.—Briefing on Results of the Agency Action Review Meeting (AARM) (Public Meeting) (Contact: Shaun Anderson, 301–415–2039).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

Thursday, June 5, 2008

1:30 p.m.—Meeting with Advisory Committee on Reactor Safeguards (ACRS) (Public Meeting) (Contact: Tanny Santos, 301–415–7270).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

Week of June 9, 2008—Tentative

There are no meetings scheduled for the Week of June 9, 2008.

Week of June 16, 2008—Tentative

There are no meetings scheduled for the Week of June 16, 2008.

Week of June 23, 2008—Tentative

Friday, June 27, 2008

9:30 a.m.—Periodic Briefing on New Reactor Issues (Public Meeting) (Contact: Donna Williams, 301–415–1322).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

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* The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings, call (recording)—301–415–1292. Contact person for more information: Michelle Schroll, 301–415–1662.

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Additional Information

The start time for the Briefing on Results of the Agency Action Review Meeting (AARM) (Public Meeting) on Wednesday, June 4, 2008, has been changed from 9:30 a.m. to 9 a.m.

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The NRC Commission Meeting Schedule can be found on the Internet

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

Appendix II

REVISED
June 25, 2008

AGENDA
554th ACRS MEETING
JULY 9-11, 2008

WEDNESDAY JULY 9, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:30 - 8:35 A.M. — Opening Remarks by the ACRS Chairman (Open) (WJS/CS/SD)
- 1.1) Opening statement
- 1.2) Items of current interest

2) 8:35 - ~~10:30~~ A.M. **10:40** — Stretch Power Uprate Application for Millstone Power Station, Unit 3 (Open/Closed) (JDS/DEB)
- 2.1) Remarks by the Subcommittee Chairman
- 2.2) Briefing by and discussions with representatives of the NRC staff, Dominion Nuclear Connecticut, Inc. and its contractor Westinghouse Electric Company LLC regarding the proposed 7% stretch power uprate for Millstone Power Station, Unit 3.

Members of the public may provide their views, as appropriate.

~~**[NOTE: A portion of this session may be closed to protect information that is proprietary to Dominion Nuclear Connecticut and its contractor pursuant to 5 U.S.C. 552b (c) (4).]**~~

~~**10:30 - 10:45**~~ **A.M.** 10:40 – 10:50 — ***** BREAK *****

3) ~~10:45 - 2:15~~ P.M. **10:50 – 2:30** — Selected Chapters of the Safety Evaluation Report (SER) Associated with the Economic Simplified Boiling Water Reactor (ESBWR) Design Certification Application (Open/Closed) (MLC/HJV)

~~**12:15 - 1:15**~~ **P.M.** 12:00 - 1:20 ***** LUNCH *****

- 3.1) Remarks by the Subcommittee Chairman
- 3.2) Briefing by and discussions with representatives of the NRC staff and General Electric - Hitachi Nuclear Energy (GEH) regarding selected chapters of the NRC staff's SER With Open Items associated with the ESBWR design certification application.

Members of the public may provide their views, as appropriate.

~~**[NOTE: A portion of this session may be closed to protect information that is proprietary to GEH and its contractors pursuant to 5 U.S.C. 552b (c) (4).]**~~

~~2:15 - 2:30~~ P.M. ***** BREAK *****
2:30 – 2:40

4) 2:30 - 6:00 P.M. <u>Safeguards and Security Matters</u> (Open/Closed) (MVB/MB) (**Room T-8E8)**

4.1) Remarks by the Subcommittee Chairman

4.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final/proposed rules and associated regulatory guidance in the area of safeguards and security.

~~4:30 - 4:45~~ P.M.
5:00 – 5:15
*****BREAK*****

~~**[NOTE: A portion of this session will be closed to protect information classified as National Security Information as well as Safeguards Information pursuant to 5 U.S.C. 552b (c) (1) and (3).]**~~

6:00 - 6:15 P.M. ***** BREAK *****

5) 6:15 - 7:30 P.M. <u>Preparation of ACRS Reports</u> (Open/Closed)
Discussion of proposed ACRS reports on:

5.1) Stretch Power Uprate Application for Millstone Power Station, Unit 3 (Open) (JDS/DEB)

5.2) Selected Chapters of the SER Associated with the ESBWR Design Certification Application (Open) (MLC/HJV)

5.3) Safeguards and Security Matters (Closed) (MVB/MB)

<u>THURSDAY JULY 10, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND</u>

6) 8:30 - 8:35 A.M. <u>Opening Remarks by the ACRS Chairman</u> (Open) (WJS/CS/SD)

7) 8:35 - ~~10:30~~ A.M. **11:25** <u>Status of NRC Staff Activities Associated with Seismic Design Issues at Nuclear Power Plants</u> (Open) (DAP/MPL)

7.1) Remarks by the Subcommittee Chairman

7.2) Briefing by and discussions with representatives of the NRC staff regarding the 2008 seismic research program plan, the interim staff guidance on high frequency ground motion, the July 2007 Japan earthquake, and the status of resolution of Generic Safety Issue-199 (GSI-199).

Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

~~10:30 - 10:45~~ **A.M.** 10:40 – 10:50 ***** BREAK *****

8) ~~10:45~~ – 12:30 P.M. **11:30** Containment Overpressure Credit (Open/Closed) (MVB/ZA)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff and Tennessee Valley Authority (TVA) regarding technical issues related to crediting of containment overpressure during design basis accidents and special events in support of the extended power uprate for Browns Ferry Units 1, 2, and 3.

Members of the public may provide their views, as appropriate.

~~[NOTE: A portion of this session may be closed to protect information that is proprietary to TVA and/or its contractors pursuant to 5 U.S.C. 552b (c) (4).]~~

~~**12:30 - 1:30**~~ **P.M.** 1:15 – 2:15 ***** LUNCH *****

9) ~~1:30 - 2:15~~ P.M. **2:15 – 3:00** Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (WJS/FPG)

9.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.

9.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

~~10) 2:15 - 2:30 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (WJS/CS/AFD)~~
~~Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters~~

2:30 - 2:45 P.M. ***** BREAK *****

11) 2:45 - 7:30 P.M. Preparation of ACRS Reports (Open/Closed)
Continued discussion of proposed ACRS reports listed under Item 5.

FRIDAY JULY 11, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

12) 8:30 - 1:00 P.M. Preparation of ACRS Reports (Open/Closed)
Continued discussion of proposed ACRS reports listed under Item 5. There may be a 15 minute break at some point during this activity.

13) 1:00 - 1:30 P.M. Miscellaneous (Open) (WJS/FPG)
Discussion of matters related to the conduct of Committee activities and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTES:

- During the days of the meeting, phone number 301-415-7360 should be used in order to access anyone in the ACRS Office.

- Presentation time should not exceed 50 percent of the total time allocated for a given item. The remaining 50 percent of the time is reserved for discussion.

- Thirty five (35) hard copies and one (1) electronic copy of the presentation materials should be provided to the ACRS in advance of the briefing.

SIGN-IN SHEETS
554TH ACRS FULL COMMITTEE MEETING
July 9, 2008

	NAME	NRC ORGANIZATION
1	J. Lamb	NRR
2	J. Miller	NRR
3	S. Ray	NRR
4	G. Cranston	NRR
5	B. Parks	NRR
6	S. Miranda	NRR
7	G. Georgiev	NRR
8	A. Cubbage	NRO
9	M. Caruso	NRO
10	P. Shemanski	NRO
11	A. Pal	NRO
12	D. Jeng	NRO
13	R. McNally	NRO
14	M. Shams	NRO
15	M. Abid	NRO
16	J. Huang	NRO
17	P. Y Chen	NRO
18	T. Spicher	NRO
19	M. Chakravorty	NRO
20	S. Chakrabarti	NRO
21	M. Jardaneh	NRO
22	P. Sekerak	NRO
23	D. Shum	NRO
24	R. Hernandez	NRO
25	T. Scarbrough	NRO
26	C. Ng	NRO
27	A. Hsia	NRO
28	C. Patel	NRO

SIGN-IN SHEETS
554TH ACRS FULL COMMITTEE MEETING
JULY 9, 2010

29	J. Fair	NRR
30	B. Thomas	NRO
31	P. Patel	NRO
32	T. Reed	NRR
33	D. Nelson	NRR
34	E. Bowman	NRR
35	R. Chazell	NRO
36	J. Vaughn	NSIR
37	B. Schneider	NSIR
38	D. Gordon	NSIR
39	J. Zimmerman	NRR
40	S. Schneider	NRR
41	M. Case	NRR
42	P. Holahan	NSIR
43	G. Tartell	NRO
44	S. Ali	RES
45	G. Mizuno	OGC
46	P. Madden	NRO
47	K. Lois	NRR

SIGN-IN SHEETS
554TH ACRS FULL COMMITTEE MEETING
July 10, 2008

	NAME	NRC ORGANIZATION
1	G. Bjorkman	NMSS
2	M. Stutzke	RES
3	M. Chakravorty	NRO
4	P. Shemanski	NRO
5	A. Murphy	RES
6	J. Thompson	NRO
7	P. Y. Chen	NRO
8	J. Kauffman	RES
9	R. Karas	NRO
10	C. Munson	NRO
11	J. Perez	RES
12	M. Shams	NRO
13	M. Shah	NMSS
14	A. Campbell	NMSS
15	H. Graves	RES
16	Y. Orechwa	NRR
17	L. Killiam	RES
18	E. Miller	NRO
19	F. Vega	NRO
20	V. Graizer	NRO
21	E. Brown	NRR
22	D. Frumkin	NRR
23	J. cusing	NRR
24	T. Orf	NRR
25	T. Collins	NRR
26	K. Martin	NRR
27	T. McGinty	NRR
28	T. Boyce	NRR

July 9, 2008

	NAME	OUTSIDE ORGANIZATION
1	D. Dominicis	Westinghouse
2	T. Williams	Westinghouse
3	J. Killimaher	Westinghouse
4	B. Gillard	Dominion
5	B. Papken	Dominion
6	B. Kellerman	Westinghouse
7	G. Wang	Westinghouse
8	W. Barton	Dominion
9	A. Price	Dominion
10	J. Craffey	Dominion
11	D. Fink	Westinghouse
12	R. Patel	Dominion
13	A. Gundersen	Fairewinds
14	D. Kovack	Westinghouse
15	J. Deaver	General Electric –Hitachi (GEH)
16	D. Hamon	GEH
17	P. Campbell	GEH
18	C. Rajendra	GEH
19	A. Liu	GEH
20	R. Kingston	GEH
21	J. Alan Beard	GEH
22	A. Pfister	WEC
23	P. Hastings	Duke Energy
24	J. Weil	McGraw Hill
25	J. Weil	GEH
26	J. Keys	NEI
27	J. Guery	Dominion
28	M. Elmeghrihr	Dominion
29	M. O'Connor	Dominion

	NAME	ORGANIZATION
30	P. Russell	Dominion
31	M. Kim	Dominion
32	R. Thomas	Dominion
33	G. Gardner	Dominion
34	A. Gharakhanian	Dominion
35	K. Connor	Dominion
36	H. Beeman	Dominion
37	D. Graves	Shaw
38	R. Bain	Shaw
39	S. Ferguson	Shaw
40	L. Salyards	Dominion
41	R. MacManus	Dominion
42	H. Onoratd	FP&L
43	S. Blodgett	Dominion
44	D. Hei	Dominion
45	S. Andre	Westinghouse
46	N. Florentine	Westinghouse
47	B. Eakin	Dominion
48	K. Descandes	Dominion
49	R. burnham	Dominion
50	J. Hantz	Westinghouse
51	A. Elms	Dominion
52	D. Huegel	Westinghouse
53	S. Antoine	Westinghouse
54	J. Riccio	Greenpeace

July 10, 2008

	NAME	OUTSIDE ORGANIZATION
1	L. Reiter	Consultant
2	R. Barrett	AdSTM Contractor
3	W. He	AdSTM Contractor
4	M. Rasmussen	TVA
5	L. Stafford	TVA
6	P. Heck	TVA
7	M. Purcell	TVA
8	J. Wolcott	TVA
9	B. Wetzel	TVA
10	C. Carey	TVA
11	R. Kalantari	EPM
12	B. Morris	TVA
13	J. Emens	TVA
14	R. Marks	TVA
15	A. Heymer	NEI
16	G. Li	GEH
17	T. Abney	GEH
18	J. Stamatakis	CNWRA
19	B. MacKissock	NMC/Monticello

SIGN-IN SHEETS
554TH ACRS FULL COMMITTEE MEETING
June 6, 2008

	NAME	OUTSIDE ORGANIZATION
1	M. Kanzda	MNES
2	M. Onozuica	MNES
3	M. Hoshi	MHI
4	H. Arikawa	MHI
5	H. Teshima	MHI
6	M. Takashima	MHI
7	H. Hamamoto	MHI
8	M. Kikuta	MHI
9	Y. Ogata	MHI
10	M. Ishida	MNES
11	S. Watanabe	MNES
12	D. Wood	MHI
13	K. Kawai	MNES
14	S. Kaawanago	MNES
15	T. Hafesa	Worley Parsons
16	S. Unkewicg	Alion
17	T. Shiraishi	MHI
18	K. Yamauchi	MHI
19	D. Fischer	NUMARK Associates
20	M. Lucas	Luminant
21	K. Paulson	MNES
22	D. Lange	MNES
23	J. Butler	NEI

Appendix IV

LIST OF DOCUMENTS FROM THE
554th ACRS MEETING JULY 9-11, 2008

Agenda Item 2:
Stretch Power Uprate Application for Millstone Power Station, Unit 3

1. Proposed Schedule
2. Status Reports
3. References

Agenda Item 3:
Selected Chapters of the Safety Evaluation Report (SER) Associated with the Economic Simplified Boiling Water Reactor (ESBWR) Design Certification Application

4. Proposed Schedule
5. Status Report
6. Attachments

Agenda Item 4:
Safeguards and Security Matters

7. Table of Contents
8. Proposed Meeting Agenda
9. Status Report

Agenda Item 7:
Status of NRC Staff Activities Associated with Seismic Design Issues at Nuclear Power Plants

10. Proposed Schedule
11. Status Report
12. Attachments

Agenda Item 8:
Containment Overpressure Credit

13. Agenda
14. Status Report
15. Draft Staff COP Safety Evaluation Input
16. Draft TVA July 10th COP Slides
17. June 12, 2008 TVA Submittal addressing ACRS COP Concerns
18. November 15, 2007 TVA Response to Round 13 RAI on COP Regarding Risk Evaluation
19. February 8, 2008 Staff ACRS Briefing Slides on Background of Containment Overpressure Credit
20. Selected ACRS Commission Slides on Containment Overpressure Credit

ACRS 554th Full Committee Meeting

NRC Staff Review of Proposed Stretch Power Uprate

For

Millstone Power Station, Unit 3

July 9, 2008

Opening Remarks

Joseph G. Giitter

Director

Division of Operating Reactor Licensing

Office of Nuclear Reactor Regulation

Opening Remarks

- NRC staff effort
  - Requests for additional information
  - Supplements to application
- Most challenging review area included:
  - Fuel and core design analysis
- Safety evaluation - no open technical issues

Introduction

John G. Lamb
Senior Project Manager
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Introduction

- Dominion Nuclear Connecticut, Inc. (DNC) is the licensee for Millstone Power Station, Unit 3 (MPS3)
- MPS3 Proposed Stretch Power Uprate (SPU)
  - 3,411 to 3,650 Megawatts Thermal (MWt)
  - Approximately 7% increase (239 MWt)
- Background
  - Licensed January 31, 1986
  - Approved License Renewal – October 2005
  - Operating License expires November 25, 2045
- Method of NRC staff review – RS-001 as guidance
- Schedule and Implementation

Topics for July 9, 2008

- Introduction and Overview of the SPU application
- Fuel & Safety Analysis
- Conclusion

Fuel and Reactor Systems Evaluation MPS3 SPU

Benjamin Parks and Samuel Miranda
Reactor Systems Branch
Leonard Ward, Ph.D.
Nuclear Performance and Code Review Branch
Division of Safety Systems
Office of Nuclear Reactor Regulation

Review Scope

- Staff reviewed the impact of SPU on
  - Fuel system and nuclear design
  - Thermal-hydraulic design
  - Overpressure Protection
  - Accident & Transient analyses
  - LOCA
  - ATWS
  - Westinghouse methods

Review Method

- Scope of EPU evaluations generally followed NRC-accepted, generic SPU guidelines and evaluations

- Analyses and evaluations are based on NRC-approved methodologies, analytical methods, and codes

- Followed the EPU review standard (RS-001)

Fuel System and Nuclear Design

- Evaluations:
  - Mechanical based on multiple fuel types
  - Nuclear/Thermal-hydraulic on RFA/RFA2
- Uprate effects:
  - Slight increase to linear heat rate
  - Slightly less peaked core design
- Licensee's evaluations demonstrate that acceptable core design may be achieved at uprated power level
- Cycle-specific analyses and evaluations will demonstrate compliance in accordance with NRC-approved reload licensing process

Accident & Transient Analyses

- Review included those transients covered in Matrix 8 of RS-001; results were acceptable as noted in staff's SER.
- Several accidents/transients warranted additional staff review:
  - Overpressure Protection
  - Inadvertent ECCS Actuation/P-19 Permissive
  - Rod Withdrawal at Power – Low Power

Overpressure Protection

- Limiting Overpressure event is Loss of Load/Turbine Trip
- Applicable ANSI Condition II Acceptance Criterion:
  - Limit peak pressure to 110% of reactor coolant system design pressure
- Two trips terminate event:
  - High Pressurizer Pressure
  - Overtemperature-ΔT

Overpressure Protection Continued

- Pursuant to staff request for additional information, licensee analyzed event crediting only the second (OTΔT) trip.

- Results of sequence crediting either trip were acceptable

- Peak pressure did not exceed 2750 psi (110% RCS Design Pressure)

Inadvertent ECCS Actuation

- Licensee will implement new permissive, P-19 Cold Leg Injection Permissive

AOO Acceptance Criterion

- "By itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV type without other incidents occurring independently."
- NRC reminded licensees that this criterion is in the plant licensing bases, and therefore must be met (RIS 2005-29).

AOOs That Add Mass to RCS

- Inadvertent Actuation of ECCS can develop into a small break LOCA at the top of the pressurizer, if a PORV sticks open.
- In analyses, PORVs that are not qualified for water relief are assumed to stick open after they relieve water.

Millstone Unit 3 Operating Experience

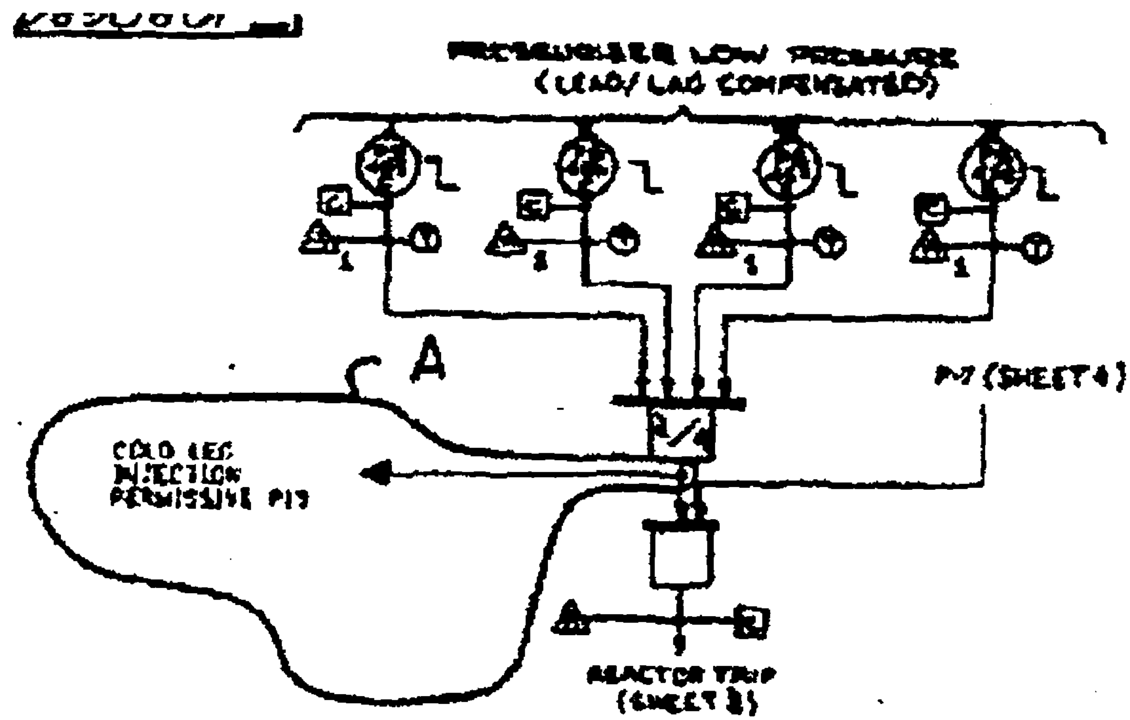
- Inadvertent actuation of ECCS incident occurred on April 17, 2005.
- Resulted in water relief through the PORVs

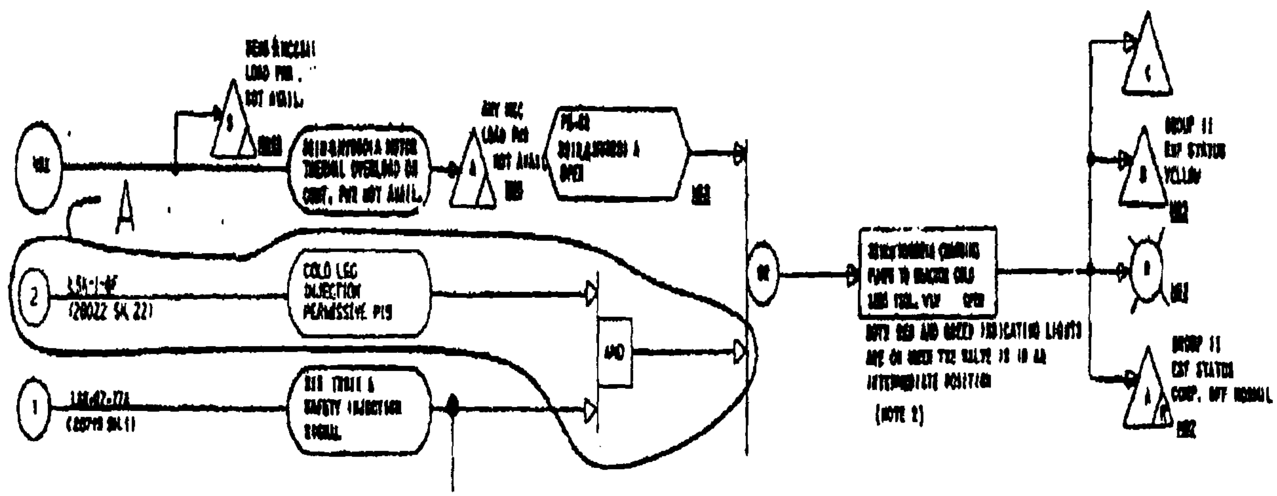
Millstone Unit 3

- PORVs are qualified for water relief
- P-19 Permissive interlocks the charging cold leg injection valves with a low pressurizer pressure signal coincident with an SI signal.

P-19 Permissive

- Charging cold leg injection valves do not open unless RCS pressure < low pressurizer pressure reactor trip setpoint and an SI signal is present.
- A single fault does not cause the cold leg injection valves to open. (P-19 would have prevented the incident of 2005.)


PRESSURIZER LOW PRESSURE
(LEAD/LAG COMPENSATED)
A
P-7 (SHEET 4)
COLD LEG
INJECTION
PERMISSIVE P13
REACTOR TRIP
(SHEET 2)



A
2
(20022 SH 22)
COLD LEG
INJECTION
PERMISSIVE P19
1
AND
OR
(NOTE 2)

Rod Withdrawal At Power

- Rod withdrawal at power evaluated in Licensing Report with acceptable results
- LR referenced a generic disposition of the potential for RCS overpressurization, given a RWAP initiated at a low power level
- Staff questioned the generic evaluation

Low Power RWAP – Generic Study

- Westinghouse evaluated the potential for overpressure conditions following a RWAP initiated at a power level where the high neutron flux-low setting can be blocked.
- Evaluation pertained to plants with water-filled loop seals on pressurizer safety valve discharge piping.
- Millstone 3 does not have water-filled loop seals; pressure relief would occur earlier.

Details of Generic RWAP Evaluation

- Performed for 4-loop Westinghouse plant
- Total power less than Millstone 3 SPU
- Pressurizer level lower than Millstone 3
- Remaining input parameters conservative relative to Millstone 3 SPU

Westinghouse Study of RWAP at Millstone 3

- Remove seal purge delay on pressurizer safety valve
- Increase core power level
- Increase pressurizer initial water level

Westinghouse Study of RWAP at Millstone 3 Continued

- Results confirmed that eliminating seal purge delay compensated for increased liquid volume in pressurizer and increased nuclear power addition capability

- Conclusion: Positive Flux Rate Trip terminates transient and Pressurizer Safety Valves mitigate pressurization effects.

LOCA

- Large Breaks evaluated with ASTRUM Best Estimate Method (Change from BART/BASH Appendix K Method)
- Small breaks evaluated using NOTRUMP (no change)
  - SBLOCA results show significant margin to regulatory limit

LOCA Results

	Small Break	Large Break	Acceptance Criterion
Peak Clad Temp, °F	1193	1781	2200
Local Cladding Oxidation, %	0.05	3.5	17
Core Wide Oxidation, %	0.01	0.12	1.0

Summary

- Transient and accident analyses demonstrate acceptable results at uprated conditions
- Fuel design remains acceptable to support the uprate
- Methods implemented acceptably

Staff Conclusion

- The staff concludes that there is reasonable assurance that the health and safety of the public will not be endangered by the proposed SPU.

GREENPEACE

Aircraft Impacts & New Nuclear Reactor Designs

U.S. Nuclear Regulatory Commission

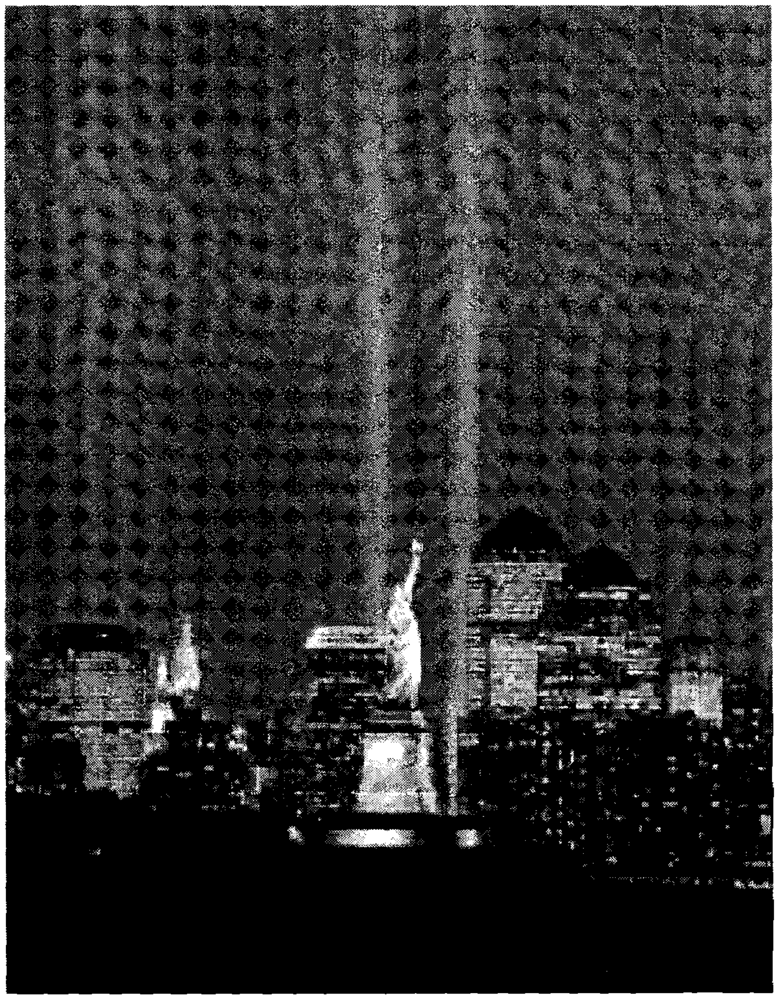
Advisory Committee on Reactor Safeguards

July 9, 2008

Jim Riccio

Nuclear Policy Analyst

Aircraft Impacts & New Nuclear Reactor Designs

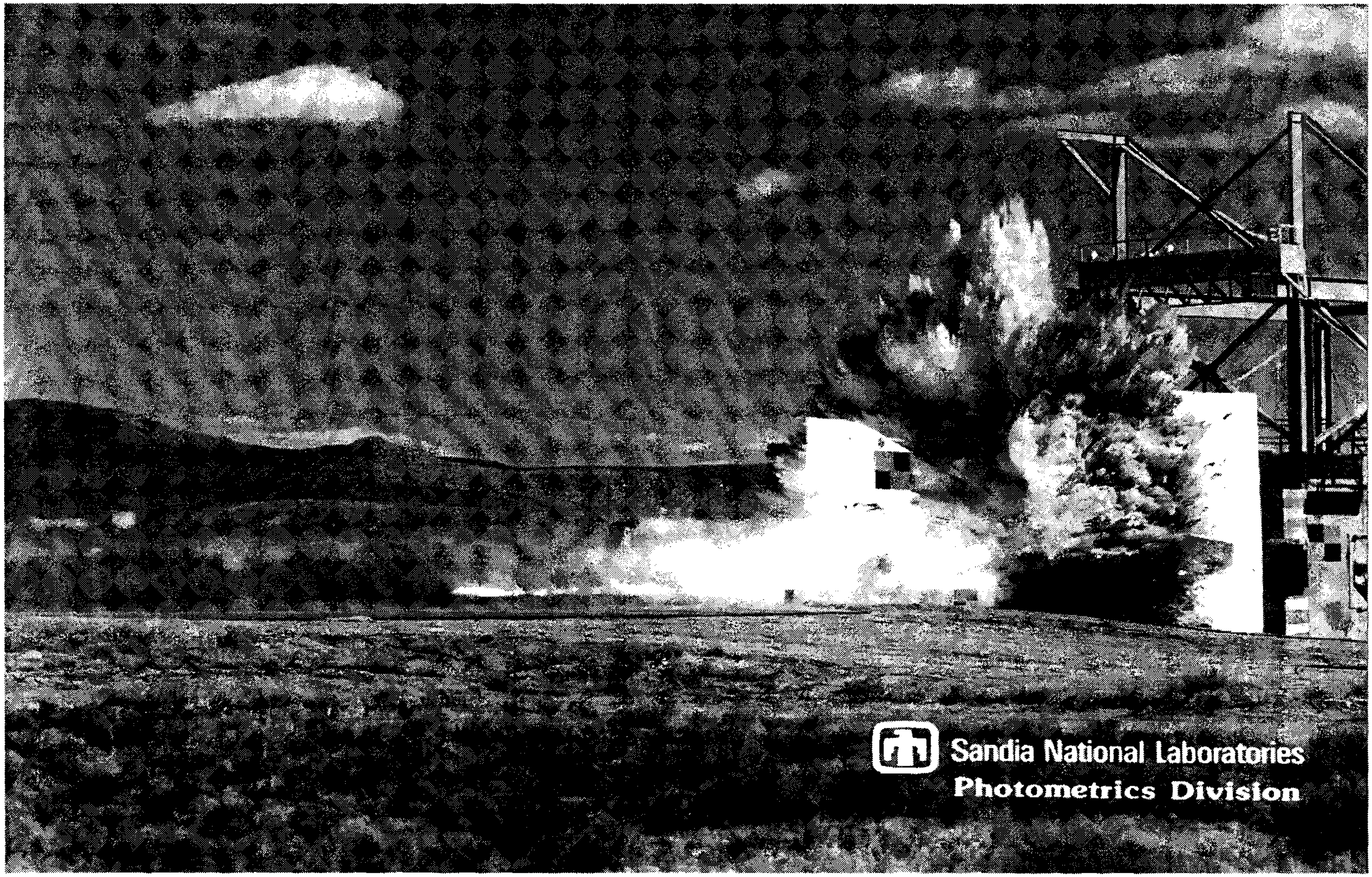


After the attacks of September 11th, the government and the nuclear industry have continued to traffic in half-truths about the vulnerability of nuclear power plants.

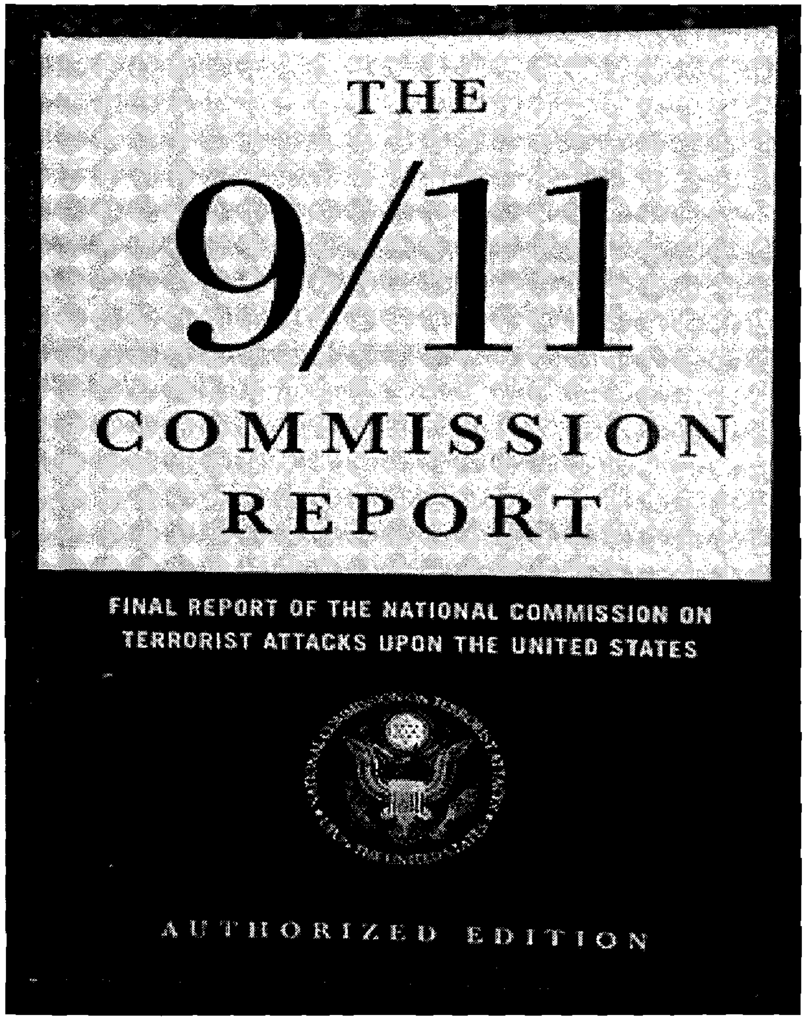
Rather than address the new reality the NRC and the nuclear industry have attempted to deceive, inveigle and obfuscate.

Aircraft Impacts & New Nuclear Reactor Designs

Aircraft Impacts & New Nuclear Reactor Designs



Aircraft Impacts & New Nuclear Reactor Designs



After 9-11, the terrorist threat is no longer hypothetical:

"KSM has admitted that he considered targeting a nuclear power plant as part of his initial proposal for the planes operation....

He also stated that Atta included a nuclear plant in his preliminary target list, but that Bin Laden decided to drop that idea."

Aircraft Impacts & New Nuclear Reactor Designs

Despite the known threat the NRC's proposed rule will not review these new nuclear reactors to ensure that they can survive a 9-11 type attack:

Corporation	Design	Site	State
Duke	AP1000	William Lee Station	SC
NuStart Energy	AP1000	Bellefonte	AL
South Carolina E&G	AP1000	Summer	SC
NRG Energy	ABWR	South Texas Project	TX
Progress Energy	AP1000	Harris	NC
Progress Energy	AP1000	Levy County	FL
Southern Nuclear	AP1000	Vogtle	GA

U.S. Nuclear Regulatory Commission, Expected New Nuclear Power Plant Applications Updated June 4,2008
http://www.nrc.gov/reactors/newlicensing/new-licensing-files/expected-new-rx-applications.pdf

GREENPEACE

Aircraft Impacts & New Nuclear Reactor Designs

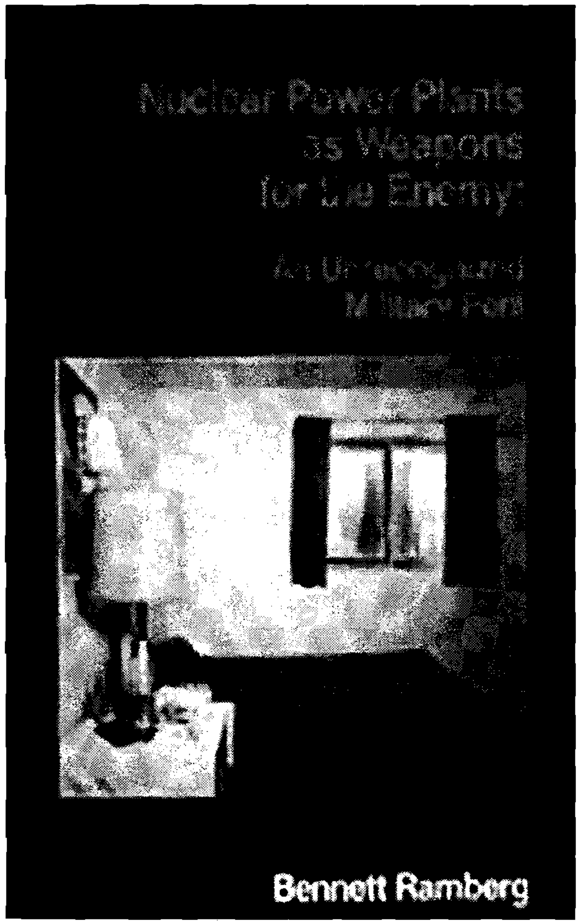
Commissioner Lyons and the Nuclear Energy Institute have offered up the strikingly similar "high level acceptance criteria."

Commissioner Lyons suggested that the industry:

- demonstrate an acceptable dose at the site boundary or
- demonstrate that the core remains cool or the containment remains intact and that spent fuel cooling is maintained.

(U.S. Nuclear Regulatory Commission, Commission Voting Record, SECY-06-204, Proposed Rulemaking - Security Assessment Requirements for New Nuclear Power Reactor Designs, April 24, 2006.)

Aircraft Impacts & New Nuclear Reactor Designs



"Keeping the terrorists guessing about our defenses was presumably one motivation for the secrecy. However, it might also reflect the commission's desire to play down its acquiescence to the nuclear industry's hubristic view that the plants are nearly invulnerable...the commission doesn't seem to have learned the lesson of those attacks."

Bennett Ramberg,
New York Times, May 20, 2003

LUMPED MASS-SPRING MODEL SEISMIC ANALYSIS vs. FINITE ELEMENT STATIC STRUCTURAL ANALYSIS

- RESULTS OF LUMPED MASS-SPRING MODEL ANALYSIS INCLUDE: NODAL MASS PT. ACCE./DISPL. RESPONSE FUNCTIONS AS WELL AS FLOOR RESP. SPECTRA (e.g., DAC-3N Code)

- DIRECT SPRING (MEMBER) FORCES, SHEAR AND BENDING MOMENTS AND JOINT DISPL./ROTATIONS

- THE ABOVE RESULTS ARE THEN APPLIED TO AN ADVANCED FINITE ELEMENT STRUCTURAL MODEL USING, SAY, ANSYS, OR SAP2000 FOR COMPUTING ELEMENT FORCES, LOCALIZED MOMEMTS AND SHEARS USED IN SIZING THE ELEMENT DIMEWNSIONS INCLUDING REBARS, STEEL PLATE SECTIONS, MEMBER DIMENSIONS, ETC.

U.S.NRC

UNITED STATES NUCLEAR REGULATORY COMMISSION

Protecting People and the Environment

Presentation to the ACRS Full Committee

ESBWR Design Certification Review

Chapter 3 – Design of Structures, Components, Equipment, and Systems

July 9, 2008

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 3 Sections & Reviewers

Richard McNally
3.2

Mohamed Shams
3.3.1
3.3.2
3.4.2

David Shum
3.4.1
3.5.1.1
3.5.1.2
3.5.1.4
3.6.1

George Georgiev
3.5.1.3
3.13

S. Rao Tammara
3.5.1.5
3.5.1.6

Renee Li
3.6.2

David Jeng
3.7.

Samir Chakrabarti
3.8

John Wu
3.9.1

Jai Rajan
3.9.2

Arnold Lee
3.9.2.2
3.9.3

Andrey Turilin
3.9.4

Patrick Sekerak
3.9.5

Thomas Scarbrough
3.9.6
3.11

P.Y. Chen
3.10

Amar Pal
3.11

John Fair
3.12

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 3 RAI Status

Total RAIs Issued – 583

Open RAIs – 57

<u>Open RAI Details</u>

- 3.8 – 19
- 3.9 – 15
- 3.6 – 8
- 3.11 - 7

ESBWR Design Certification Review
Section 3.2 – Seismic Classification and Quality Group Classification

Regulatory Basis:

10 CFR 50 Appendix A, General Design Criteria (GDC) 1 and 2

Criterion 1 --Quality standards and records. Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Criterion 2 --Design bases for protection against natural phenomena. Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

ESBWR Design Certification Review
Section 3.2 – Seismic Classification and Quality Group Classification

Classification Process

Blend of deterministic and risk-informed approach

- RG 1.26 and 1.29 for quality group and seismic classifications based on safety function
- PRA and RTNSS process establish scope and risk-significance

ESBWR Design Certification Review
Section 3.2 – Seismic Classification and Quality Group Classification

Risk Considerations Applied to Classification Process

Regulatory Treatment of Nonsafety-Related Systems (RTNSS) SSCs

- Nonsafety-Related SSCs with high risk significance are identified as RTNSS SSCs
- GEH has proposed a new Special Class to distinguish high risk significant SSCs from other nonsafety-related SSCs with low risk significance
- RTNSS SSCs will receive special treatment in terms of seismic and quality requirements
- RTNSS process is described in DCD Section 19 and evaluated in FSER Chapter 22

ESBWR Design Certification Review
Chapter 19A (SER Chap. 22)

Regulatory Treatment of Non-Safety Systems (RTNSS)

CRITERIA FOR SELECTING RTNSS SSCs:

- Non-safety SSC relied on to meet ATWS and SBO rules.
- Non-safety SSC needed for core cooling, containment heat removal or control room habitability beyond 72 hours post accident.
- Non-safety SSC that provides diagnostic info beyond 72 hours post accident.
- Non-safety SSC relied on to meet Commission's safety goals
- Non-safety SSC relied on to meet containment performance goals.
- Non-safety SSC relied upon to prevent significant adverse interaction with passive safety system.

ESBWR Design Certification Review
Section 3.7 – Seismic Design

Discussion Topics:

- ESBWR SEISMIC GROUND MOTION SPECTRA CURVES AND THEIR USE IN SEISMIC ANALYSIS
- ESBWR POOLS AND THEIR IMPACT ON SEISMIC ANALYSIS

DESIGN GROUND MOTIONS USED IN SEISMIC DESIGN OF ESBWR PLANTS

- ESBWR standard plant SSE design ground motion is rich in both low and high frequencies

- The low-frequency ground motion follows RG 1.60 ground spectra anchored to 0.3 g (Fig. 1)

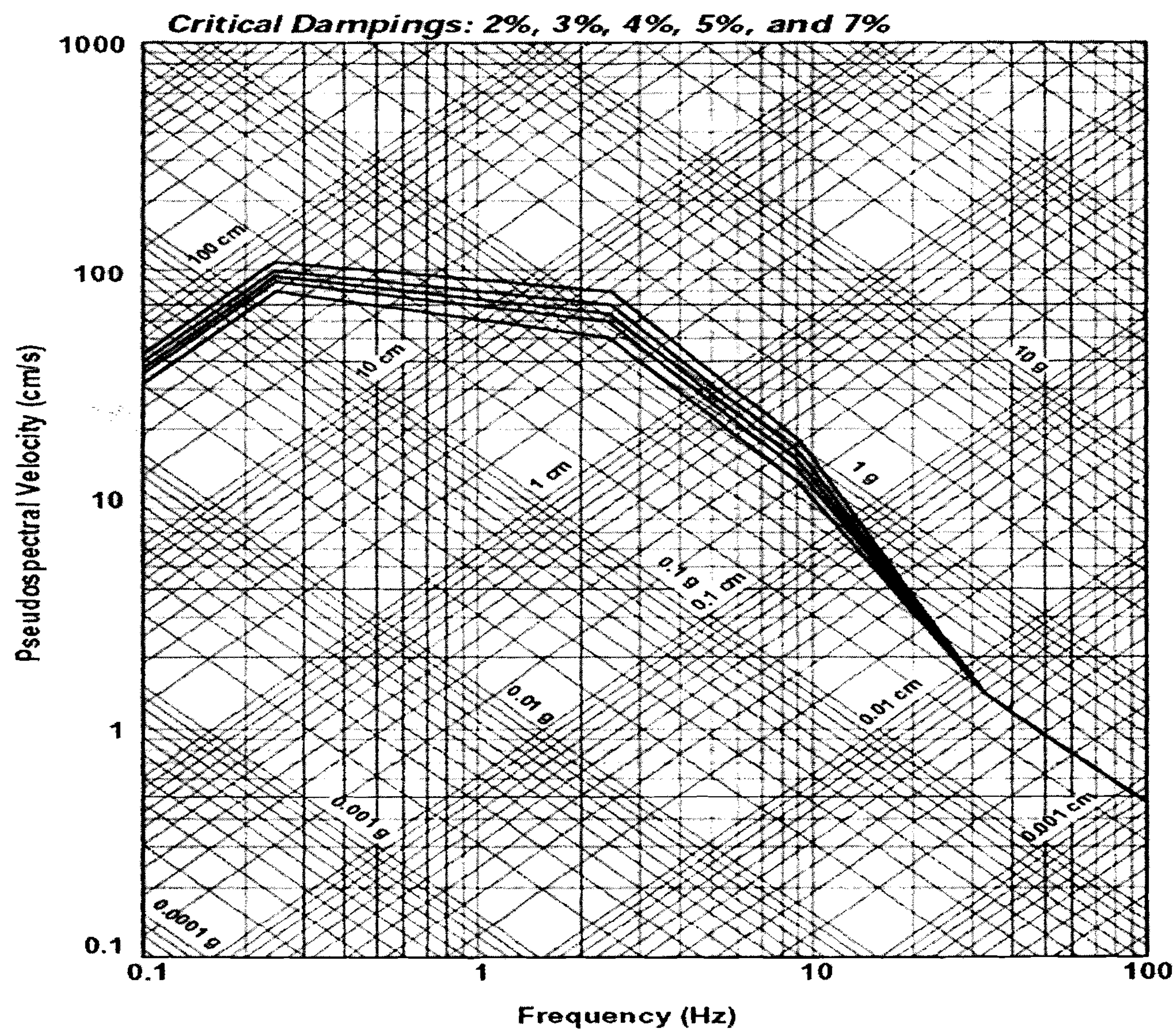
- The high frequency ground motion matches the North Anna ESP site-specific spectra as representative of most severe rock sites in the Eastern US (Fig. 2)

- These two ground motions are considered separately in the basic design (Used DAC-3N Code)

- To verify the basic design the two separate inputs (both low and high frequencies) are further enveloped to form a single envelope design ground response spectra, also termed as the Certified Seismic Design Response Spectra (CSDRS) (Fig. 3)

LOW FREQUENCY DESIGN GROUND MOTION

RG 1.60 Spectrum Anchored at 0.30 g (Fig. 1)



HIGH FREQUENCY DESIGN GROUND MOTION

North Anna ESP Site Specific Spectrum Anchored Approx. at 0.50 g (Fig. 2)

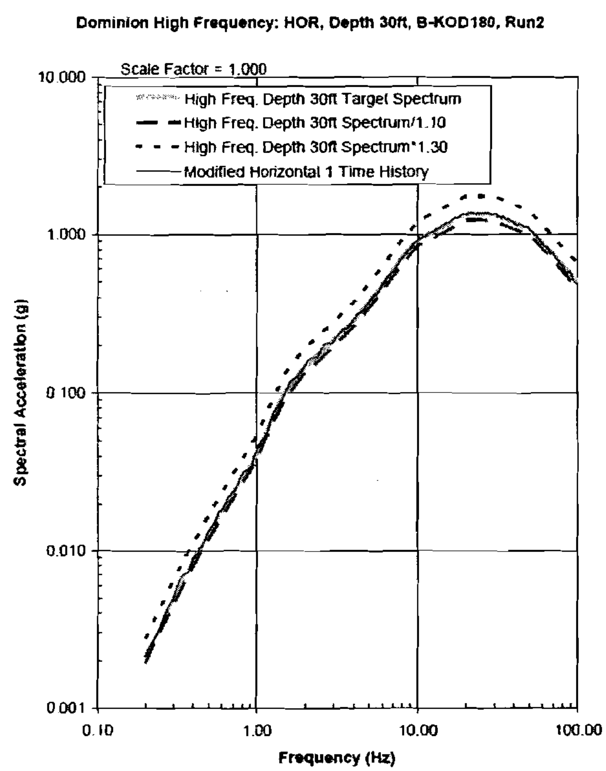
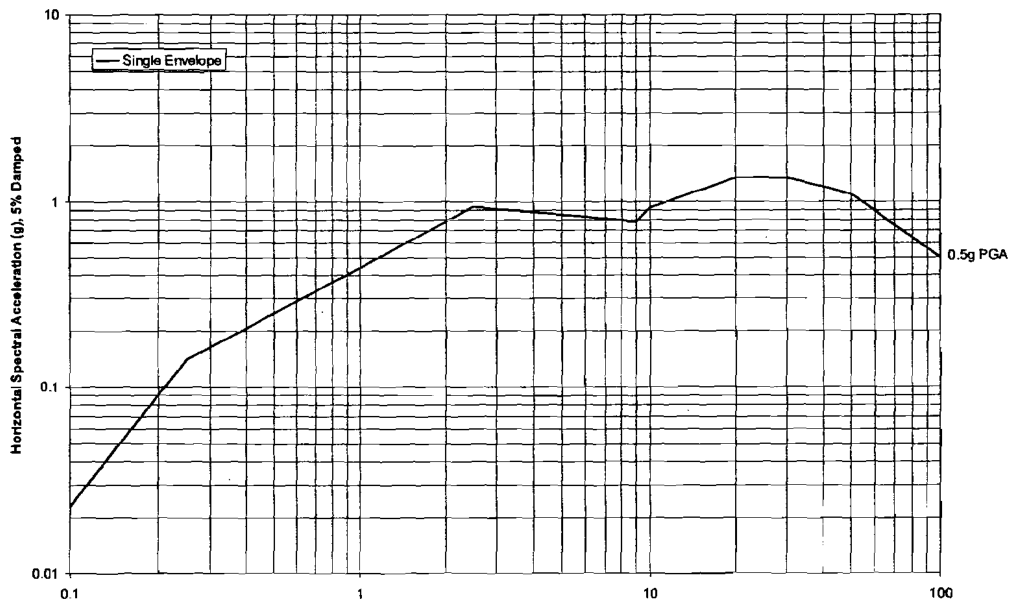
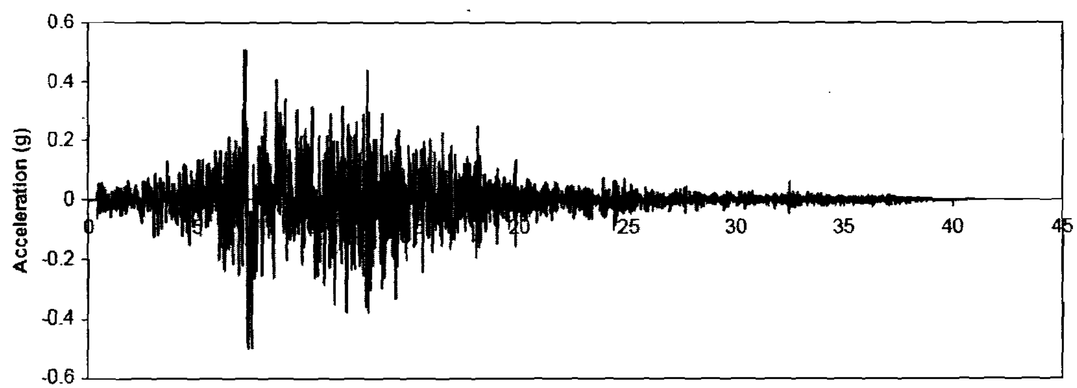


Figure 3.7-24. North Anna ESP Horizontal H1 Target Spectrum at ESBWR CB Base

ESBWR Certified Seismic Design Response Spectra (CSDRS) (GEH SLIDE) (Fig. 3)





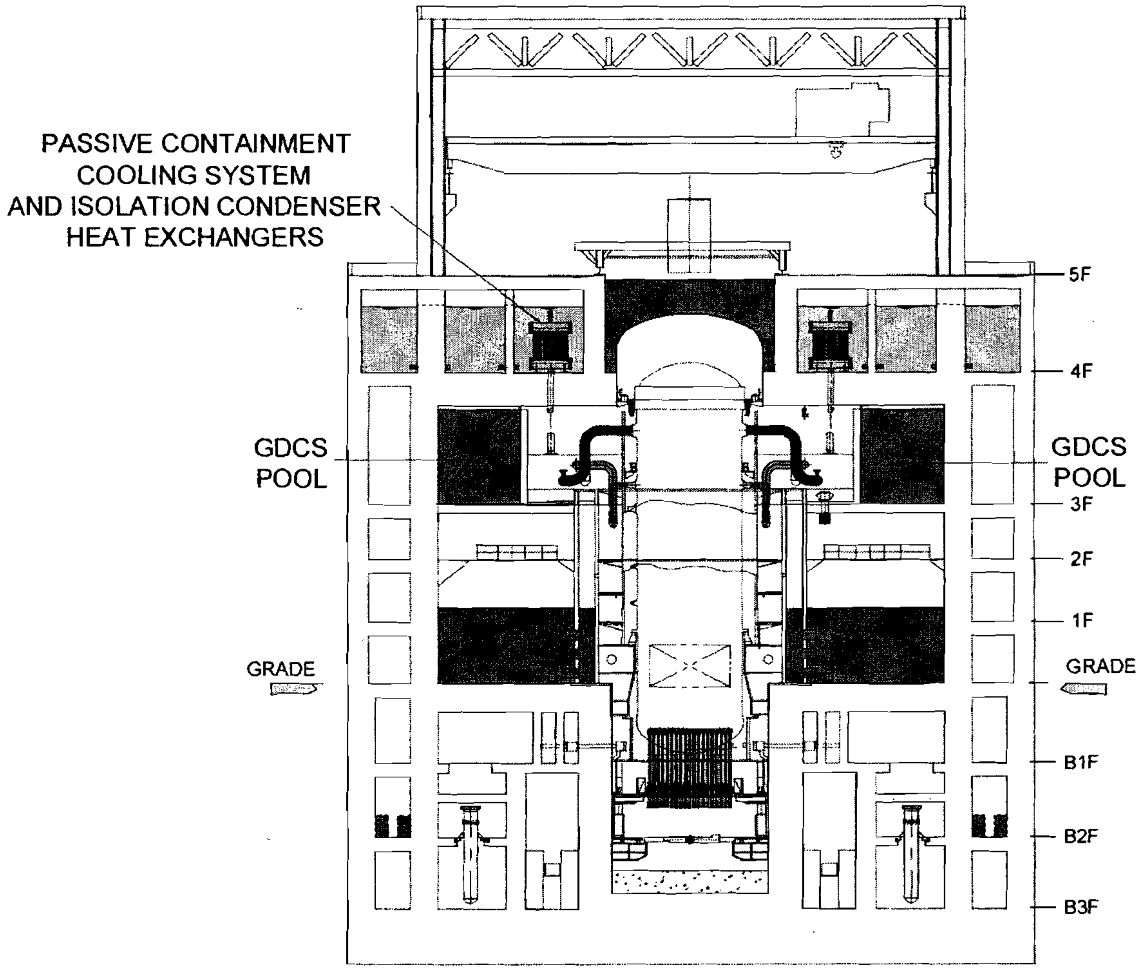
SEISMIC ANALYSIS OF ESBWR WATER TANKS AND FUEL POOLS

- ESBWR Water Tanks and Fuel Pools including Water Masses are modeled in the Seismic Analysis Models per ESBWR DCD Sections 3.7

- For Global Seismic Analysis Modeling, ESBWR Conservatively Used 100 % Water Mass for the Impulsive Mode analysis

- Design of Water Tanks and Fuel Pools Conforms with SRP Sections 3.7 and 3.8

SEISMIC ANALYSIS OF ESBWR WATER TANKS AND FUEL POOLS (GEH Slide)



REACTOR BUILDING

CONCEPTUAL MODELING OF SEISMIC ANALYSIS OF WATER TANKS

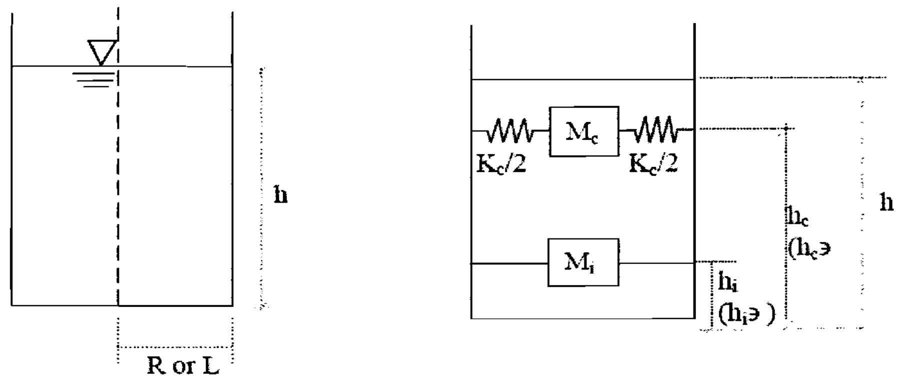


Figure 1: Description of tank dimensions and mechanical model

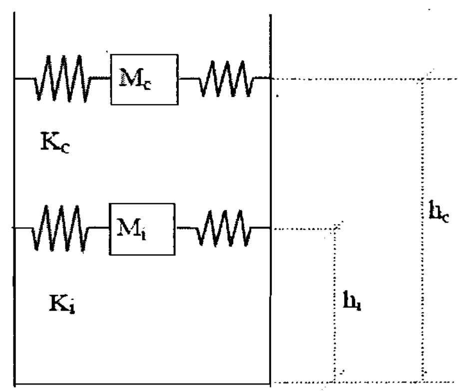
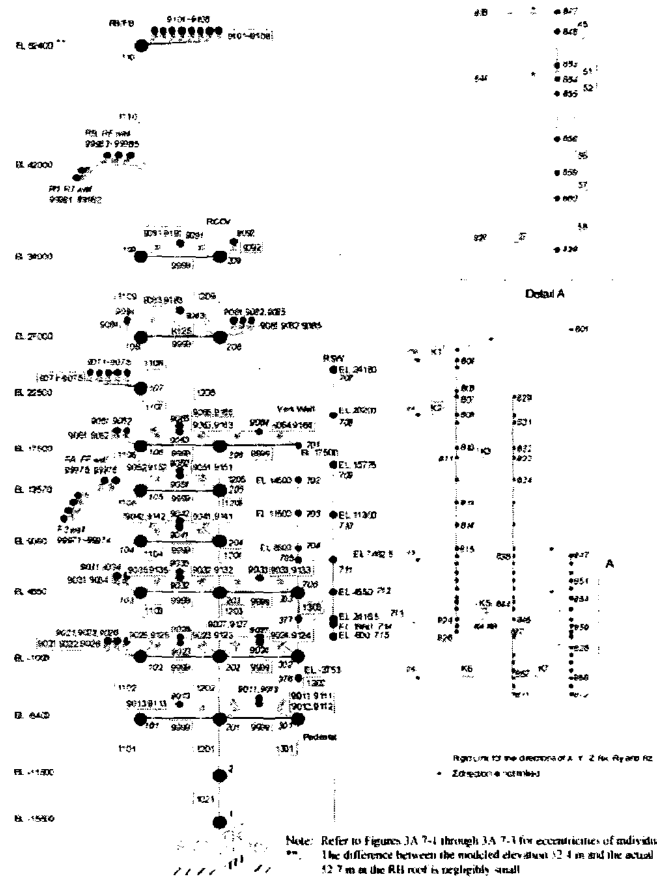


Figure 2: Mechanical models for flexible circular tanks (Malhotra et. al., 2000)

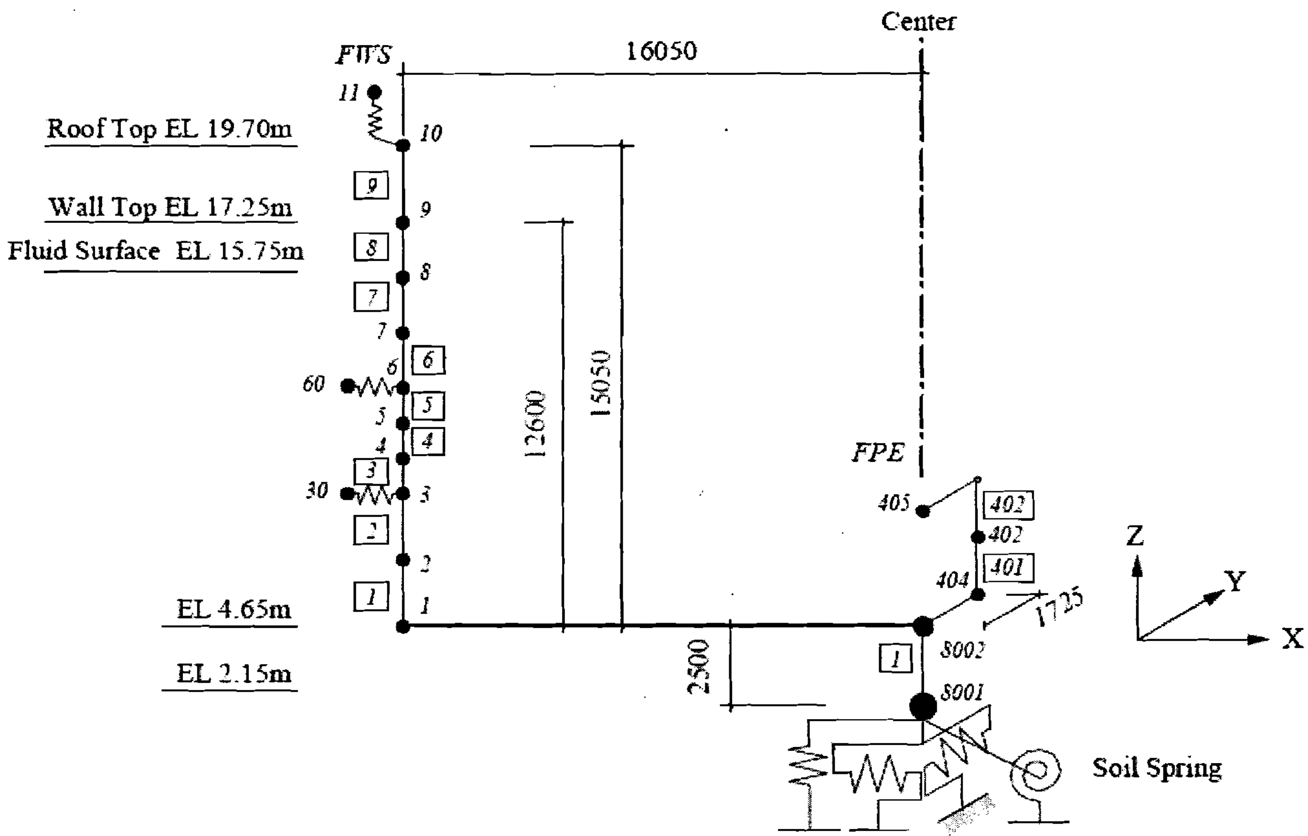
RB/FB COMPLEX SEISMIC MODEL



Note: Refer to Figures 3A.7-1 through 3A.7-3 for eccentricities of individual sticks.
**. The difference between the modeled elevation 52.4 m and the actual elevation 52.7 m at the RB roof is negligibly small

Figure 3A.7-4. RB/FB Complex Seismic Model

FWSC SEISMIC MODEL



Mass at Node 30 represents the impulsive mode.
Mass at Node 60 represents the fundamental sloshing (convective) mode.
The model is assumed to be symmetric about YZ-plane including the center line.

SEISMIC ANALYSIS OF ESBWR WATER TANKS AND FUEL POOLS (Cont'ed)

- The horizontal response analysis includes at least one impulsive mode and the fundamental sloshing (convective) mode. At least one vertical mode of fluid vibration are included in the analysis.

- The analysis models evaluate impulsive and convective masses, time period of impulsive and convective modes of vibrations, hydrodynamic pressure distribution and sloshing wave height.

- Damping values used to determine the spectral acceleration in the impulsive mode are based upon the system damping associated with the tank shell material as well as with the SSI.

- In determining the spectral acceleration in the horizontal convective mode, damping ratio is 0.5% of critical damping.

SEISMIC ANALYSIS OF ESBWR WATER TANKS AND FUEL POOLS (Cont'ed)

- The maximum overturning moment at the base of the tank and the seismically induced hydrodynamic pressures on the tank shell at any level are obtained by the modal and spatial combination methods.

- The maximum hoop forces in the tank wall are evaluated with due regard for the contribution of the vertical component of ground shaking.

- The hydrodynamic pressure at any level is added to the hydrostatic pressure at that level to determine the hoop tension in the tank shell.

- Either the tank top head is located at an elevation higher than the slosh height above the top of the fluid or else is designed for pressures resulting from fluid sloshing against this head.

ESBWR Design Certification Review
Section 3.11 – Environmental Qualification of Mechanical and Electrical Equipment

- Safety-related electrical equipment in harsh environment will be qualified by test or other methods as described in IEEE 323-1974.
- Safety-related mechanical equipment in harsh environment is qualified using test, analysis or a combination of test and analysis.
- Safety-related computer based equipment in mild environment is qualified by type testing.
- The equipment qualification method (by test or analysis or a combination of test and analysis) will be available during inspection.

CODES AND APPLICABLE TYPES OF TANKS

Table 1: Types of tanks considered in various codes

Code	Types of tanks
ACI 350.3	• Ground supported circular and rectangular concrete tanks with fixed and flexible base. • Pedestal supported elevated tanks.
AWWA D-100 & D-103	• Ground supported steel tanks with fixed and flexible base. • Elevated steel tanks with braced frame and pedestal type supporting tower.
AWWA D-110 & D-115	• Ground supported prestressed concrete tanks with fixed and flexible base.
API 650	• Ground supported steel petroleum tanks (Types of base support are not described).
NZSEE Guidelines	• Ground supported circular and rectangular tanks with fixed and flexible base. • Elevated tanks.
Eurocode 8	• Ground supported circular and rectangular tanks with fixed base. • Elevated tanks.

MAXIMUM SLOSHING WAVE HEIGHT GIVEN BY CODES

Table 7: Expressions for maximum sloshing wave height given in various codes

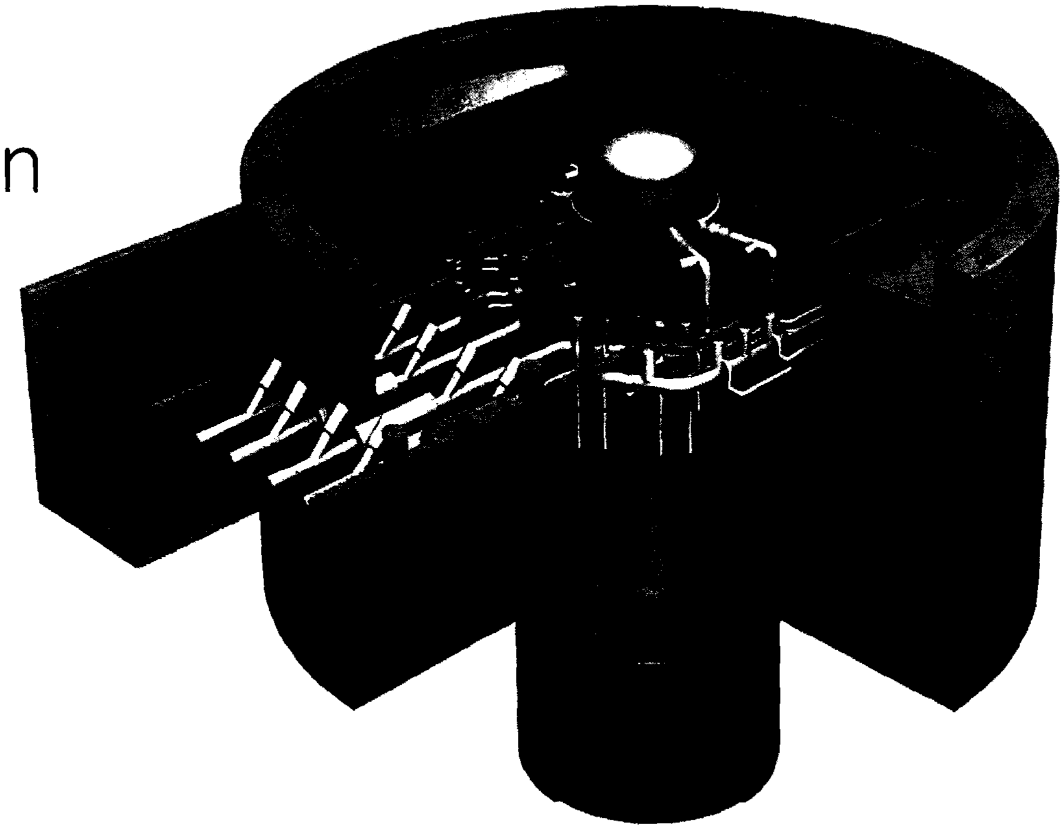
Code	**Sloshing wave height**
ACI 350.3	A_cR
AWWA D-100 & D-103	0.84 A_cR
AWWA D-110 & D-115	A_cR
API 650	Not mentioned
NZSEE Guidelines	0.84 A_cR (Considering only first mode)
Eurocode 8	$0.84A_cR$

A_c = Convective acceleration; R = Radius of tank

ESBWR DCD Chapter 3

Design of Structures, Components, Equipment and Systems

Advisory Committee on Reactor Safeguards



July 9, 2008

GE Hitachi Nuclear Energy

Introduction

> Presenters

- David Hamon, ESBWR Engineering
- Jerry Deaver, ESBWR Engineering
- Ai-Shen Liu, ESBWR Engineering
- Jeffrey Waal, ESBWR Regulatory Affairs

Presentation Content

- Chapter 3 Overview
- Selected Topics
- Summary

Chapter 3, Overview

- Chapter 3 describes the design of structures, components, equipment and systems.
  - > 3.1 – Conformance with NRC General Design Criteria.
  - > 3.2 – Classification of Structures, Systems and Components
  - > 3.3 – Wind and Tornado Loadings.
  - > 3.4 – Water Level (Flood) Design
  - > 3.5 – Missile Protection
  - > 3.6 – Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

Chapter 3, Overview (cont'd)

- Chapter 3 describes the design of structures, components, equipment and systems.
  - > 3.7 – Seismic Design
  - > 3.8 – Seismic Category I Structures
  - > 3.9 – Mechanical Systems and Components
  - > 3.10 – Seismic and Dynamic Qualification of Mechanical and Electrical equipment
  - > 3.11 – Environmental Qualification of Mechanical and Electrical Equipment
  - > 3.12 – ASME Code Class 1, 2 and 3 Piping Systems, Piping Components and Associated Supports
  - > 3.13 – Threaded Fasteners for ASME Components

Section 3.2 – Classification of SSCs

- Safety-Related Definition – 10 CFR 50.2
  > Safety-related structures, systems and components means those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

  (1) The integrity of the reactor coolant pressure boundary

  (2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

  (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

Section 3.2 – Classification of SSCs

- Safety Classification (DCD Section 3.2.3)
  - > Consistent with safety classifications used in ABWR DCD.
  - > Safety Class 1 – RCPB components and supports.
  - > Safety Class 2 – Mechanical SSCs involved in containment isolation functions not included in Safety Class 1, ECCS and RHR functions.
  - > Safety Class 3 – All other mechanical safety-related SSCs not included in Safety Classes 1 and 2. All safety-related electrical/I&C SSCs are Safety Class 3.
  - > Safety Class N – Nonsafety-related SSCs.
  - > Safety Classes 1 through 3 very closely related to Quality Groups A through C classifications for safety-related SSCs.

Section 3.2 – Classification of SSCs

- Safety Classification (DCD Section 3.2.3)
  - > Safety Classification establishes minimum requirements for all other classifications (seismic, quality group, QA)

Table 3.2-2

Minimum Safety Class Requirements

Safety Class	Minimum Design Requirements for Specific Safety Class				
	Quality Group	ASME Section III Code Class	Seismic Category[1]	Electrical Classification[2]	Quality Assurance[4]
1	A	1	I	N/A	10 CFR 50 Appendix B
2	B	2	I	N/A	10 CFR 50 Appendix B
3	C	3	I	Class 1E	10 CFR 50 Appendix B
N	D[3]	N	II or NS	Non-Class 1E	—

Section 3.2 – Classification of SSCs

- Seismic Classification (DCD Section 3.2.1)
  - > Based on RG 1.29 and SRP 3.2.1.
  - > Seismic Category I required for all safety-related SSCs.
  - > Seismic Category II required for nonsafety-related SSCs whose failure could degrade performance of safety-related SSCs and for SSCs classified as RTNSS Criterion B from PRA analyses (DCD Section 19A.3).
  - > Some nonsafety-related SSCs assigned to Seismic Category I when required by regulations.
  - > Remaining SSCs assigned to Seismic Category NS.
  - > RG 1.143 applies special seismic requirements for radioactive waste handling SSCs

Section 3.2 – Classification of SSCs

- System Quality Group Classification (DCD Section 3.2.2)
  - > Based on RG 1.26 and SRP 3.2.2.
  - > Quality Group A – Pressure-retaining portions and supports for Reactor Coolant Pressure Boundary.
  - > Quality Group B – Pressure-retaining portions and supports not in Quality Group A for safety-related containment isolation, ECCS and residual heat removal functions.
  - > Quality Group C – Pressure-retaining portions and supports for other safety-related functions not included in Quality Groups A and B.
  - > Quality Group D – Pressure-retaining portions and supports for other systems that contain or may contain radioactive material.

Section 3.2 – Classification of SSCs

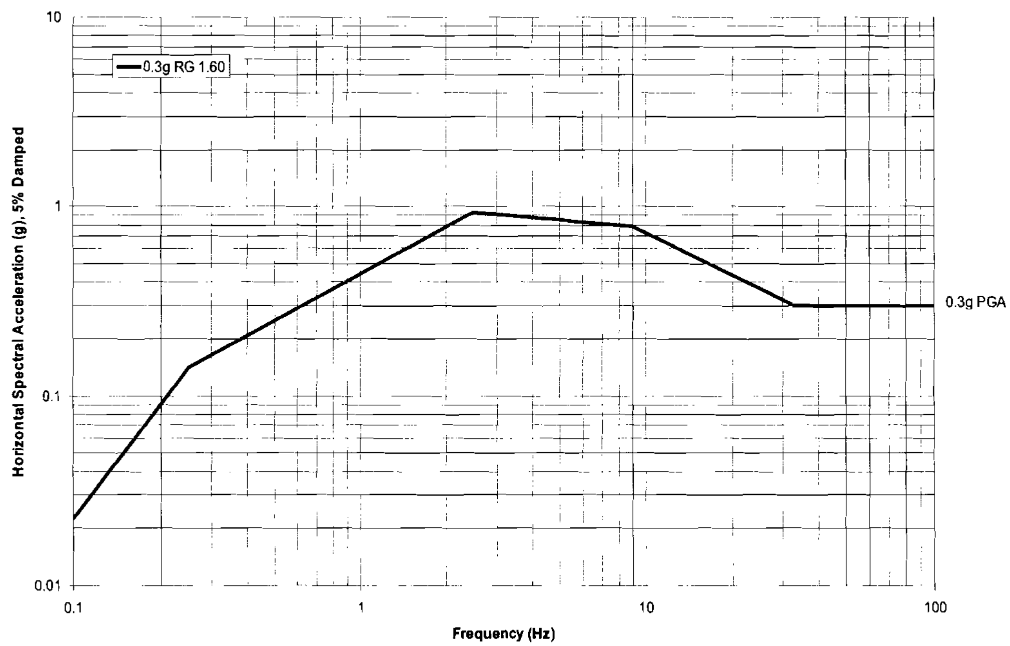
- Conclusions
  - > ESBWR classification system is consistent with previously licensed designs
  - > Safety-related SSCs are determined based on 10 CFR 50.2 definition.
  - > Safety classification establishes minimum requirements for other classifications and serves as entry point to QA program
  - > Minimum Seismic and Quality Group classifications are upgraded as required by SRPs, RGs and design practices
  - > PRA analyses determine SSCs requiring upgraded seismic design requirements due to RTNSS considerations
  - > Seismic and Quality Group classifications establish basis for NRC review under SRPs 3.2.1 and 3.2.2.

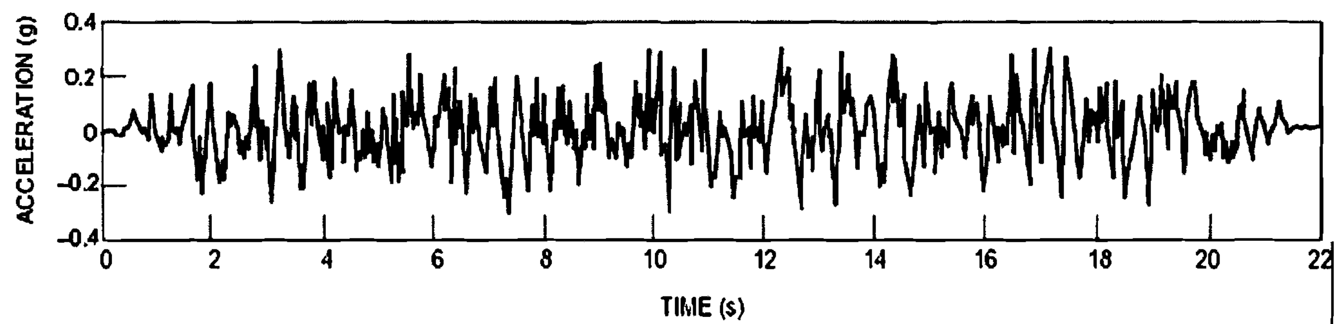
Section 3.7 – Seismic Design

- Section 3.7.1 provides seismic design parameters.
  - The CSDRS follows RG 1.60 spectra and North Anna ESP site-specific spectra at high frequencies.
  - North Anna spectra is representative of most severe rock sites in the Eastern US.
    - Note: No recorded seismic event contains simultaneously very high low-frequency and high-frequency motions. CSDRS is very conservative.
  - Artificial time histories were developed to match the CSDRS spectra per NUREG/CR-6728 criteria.

Derivation of CSDRS

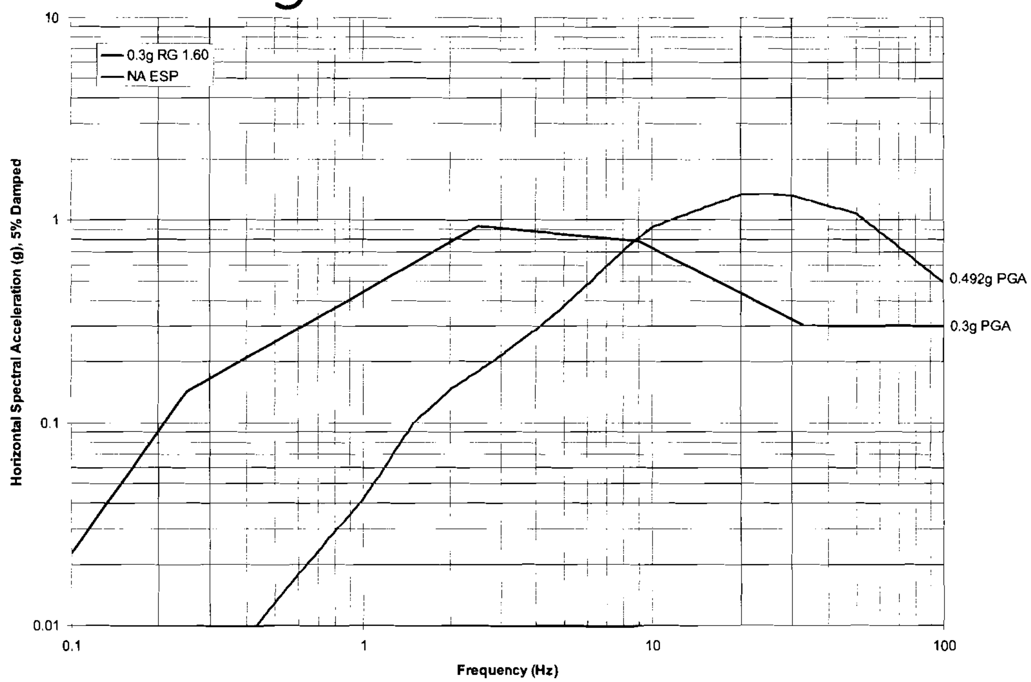
- Low-Frequency Ground Motion follows RG 1.60 with 0.3g Peak Ground Acceleration (PGA).

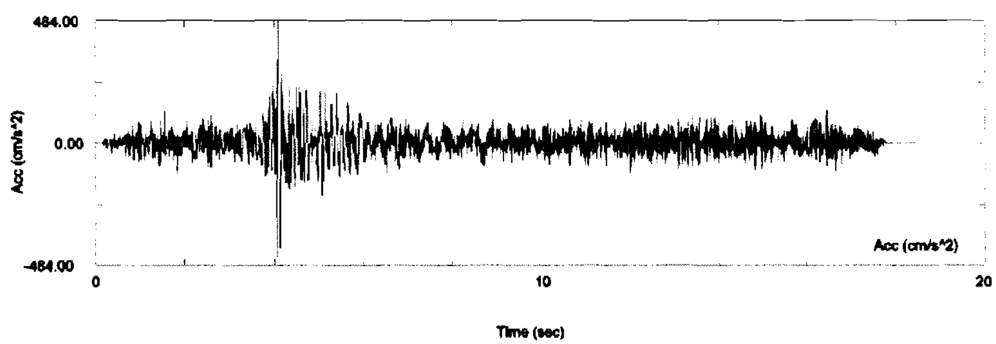




Derivation of CSDRS

- High-Frequency Ground Motion follows North Anna ESP with 0.492g PGA.

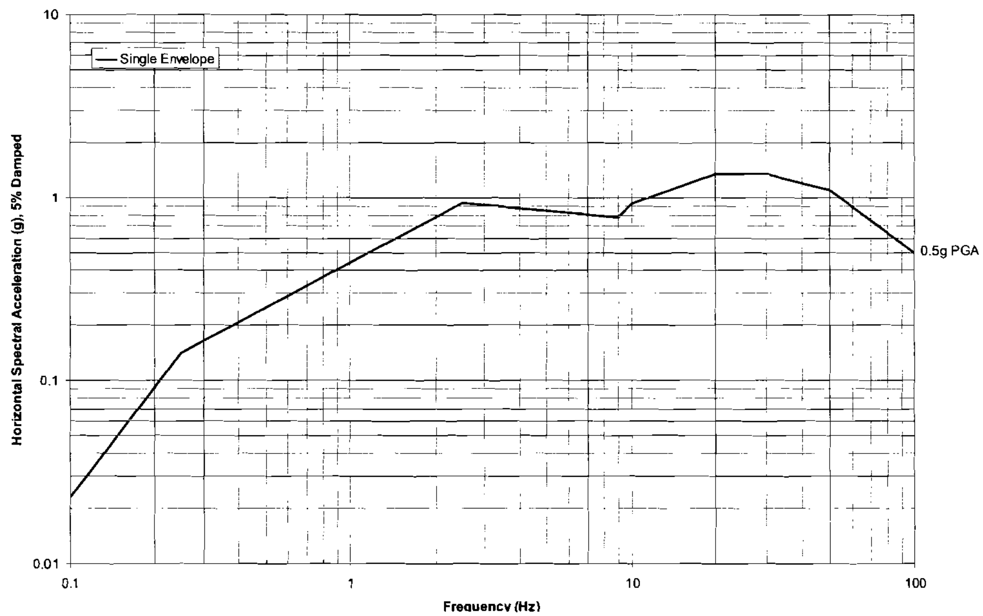


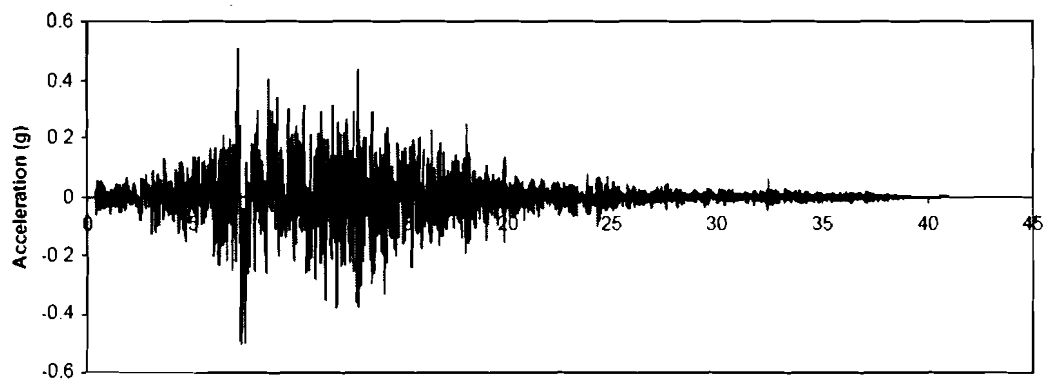


NA Time History

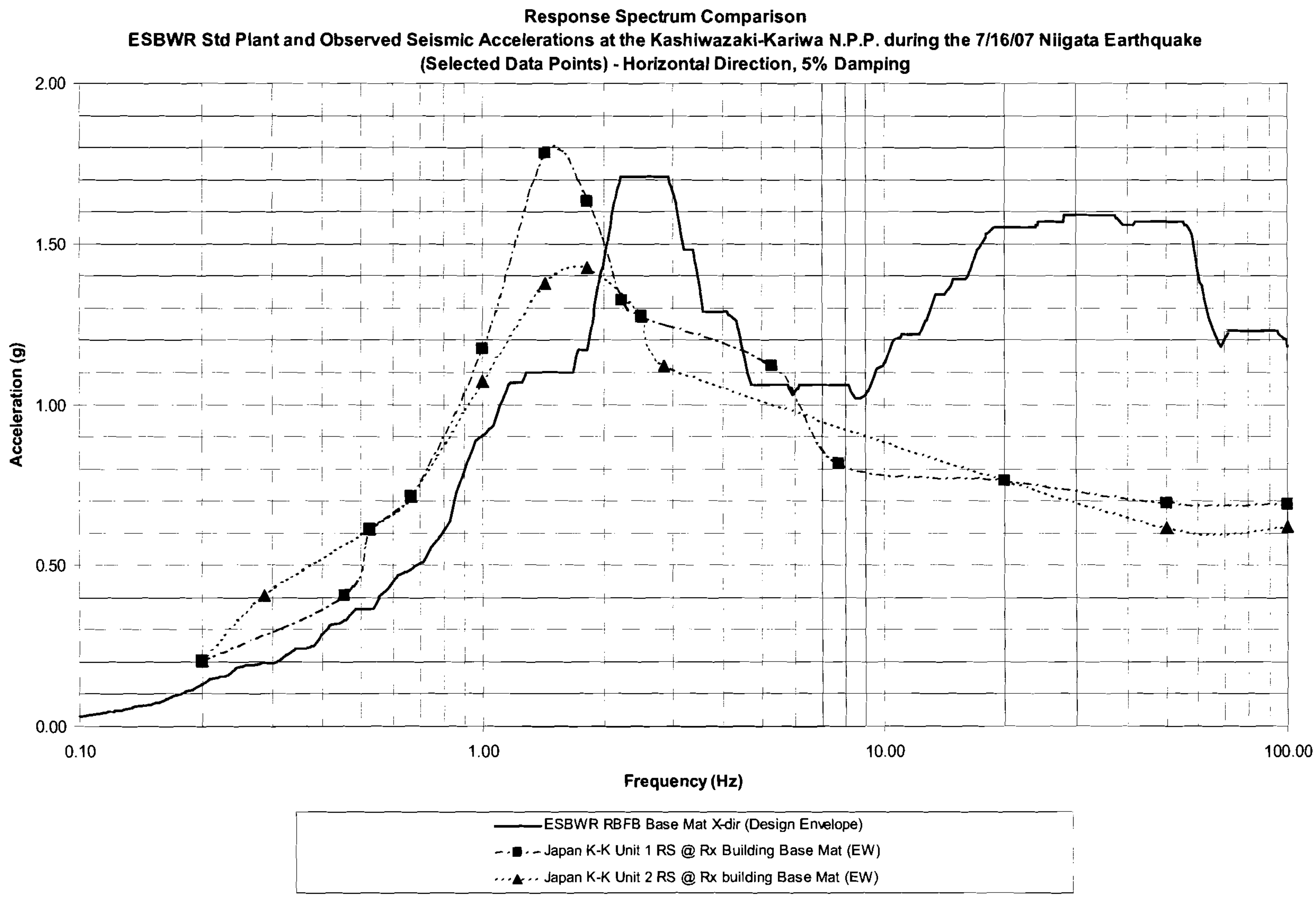
Derivation of CSDRS

- Design Ground Motion is the envelope of RG 1.60 and North Anna ESP Spectra with 0.5g PGA.





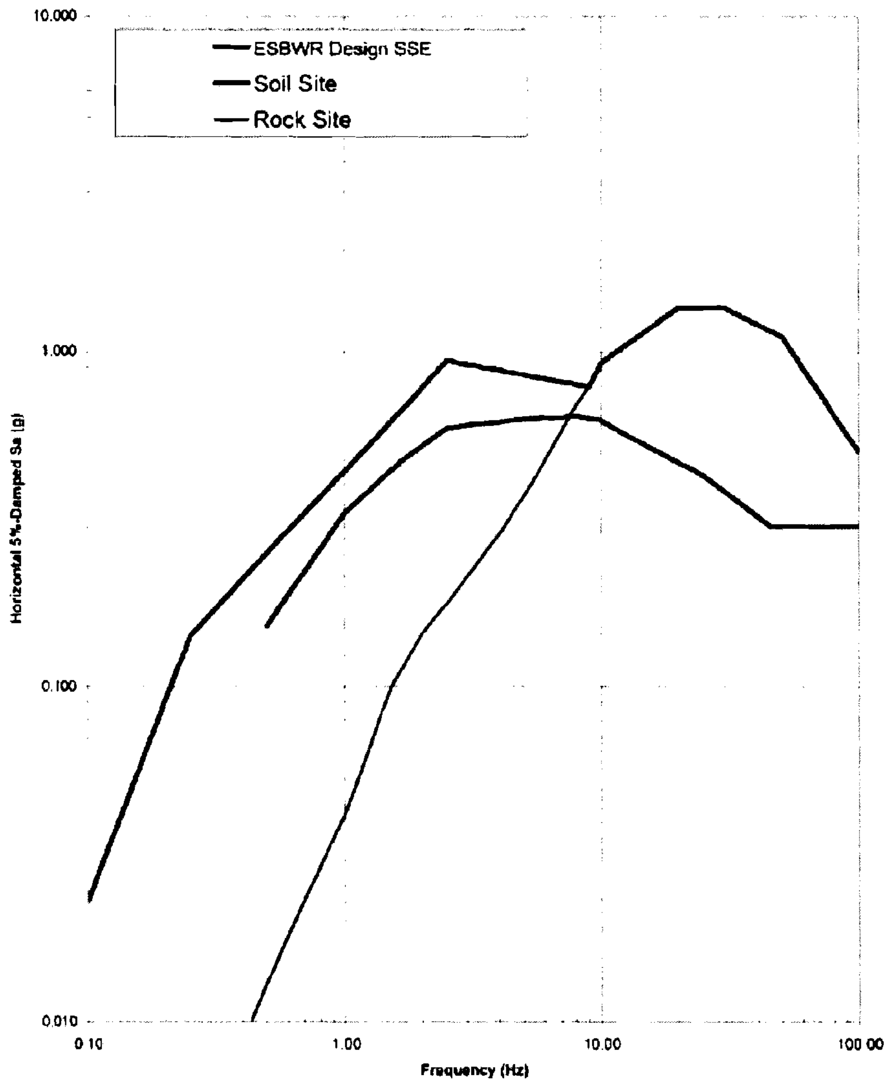
Comparison with Recent Japan Earthquake



Ground Spectra Considered in Seismic Margin Analysis

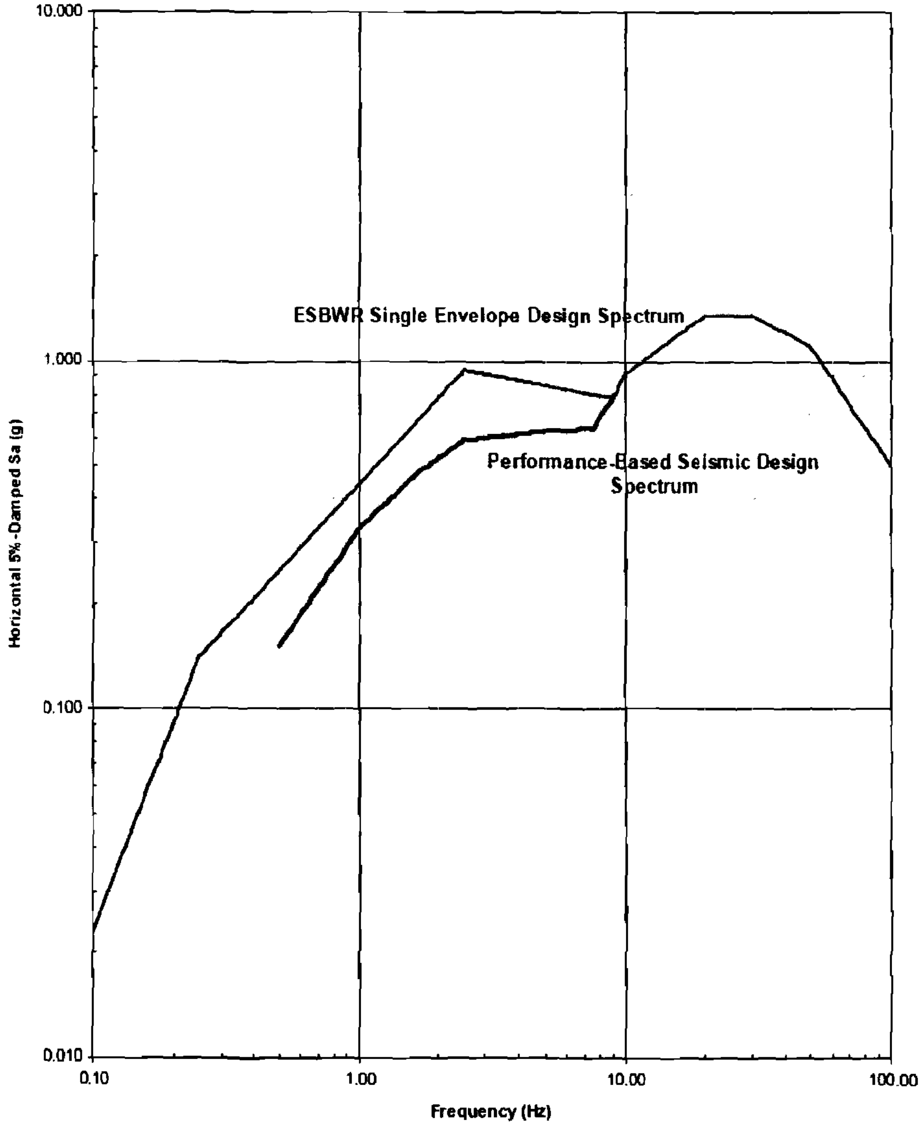
- Soil site spectrum is the bounding SSE spectrum of soil sites among the 28 sites (excluding Vogtle) included in the current EPRI study

- Rock site spectrum is North-Anna ESP spectrum

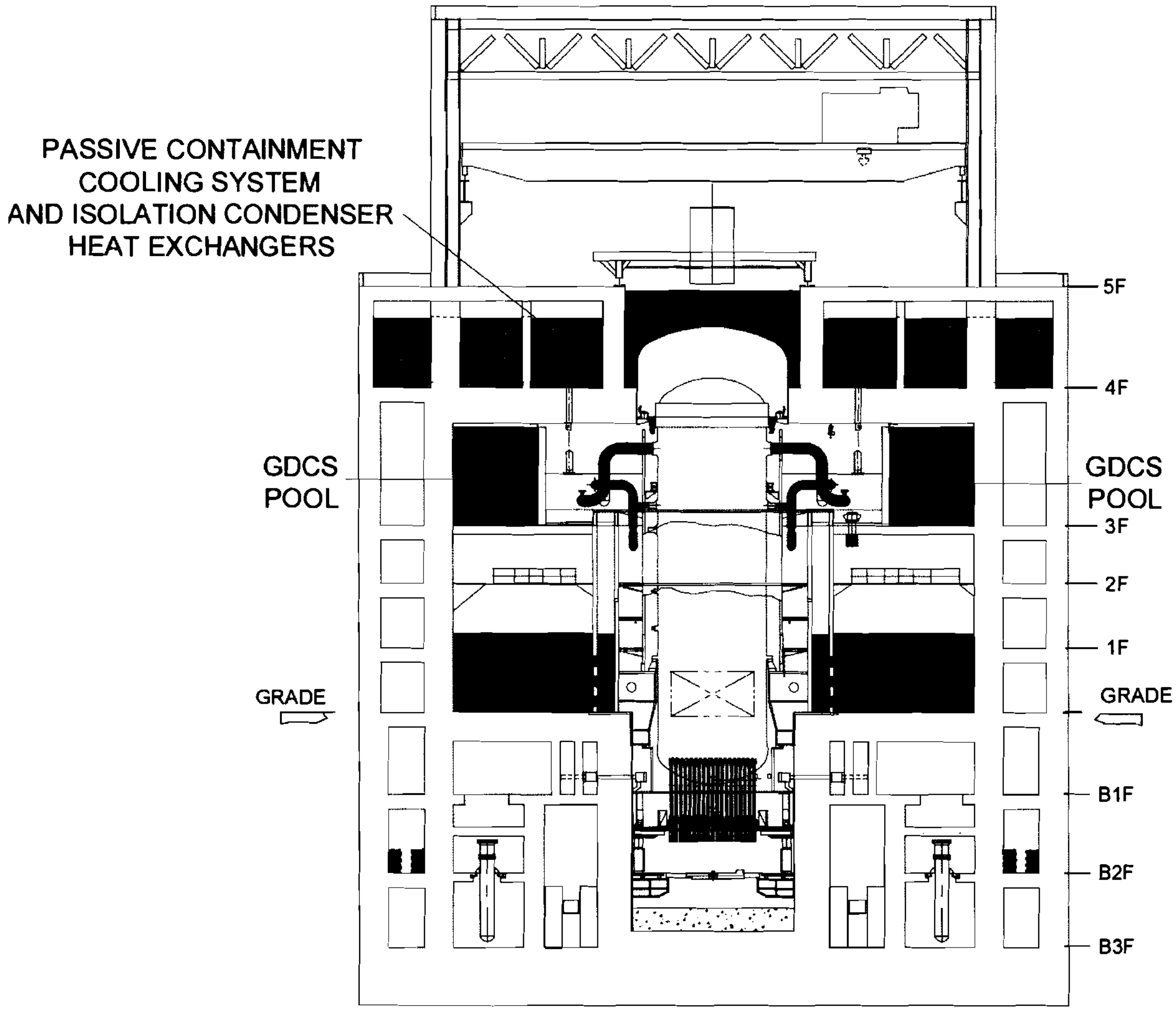


Ground Spectra Considered in Seismic Margin Analysis

- For consistency with single envelope design spectrum for all sites, performance-based soil and rock spectra are enveloped for seismic margin consideration
- Same 0.5g PGA as design spectra



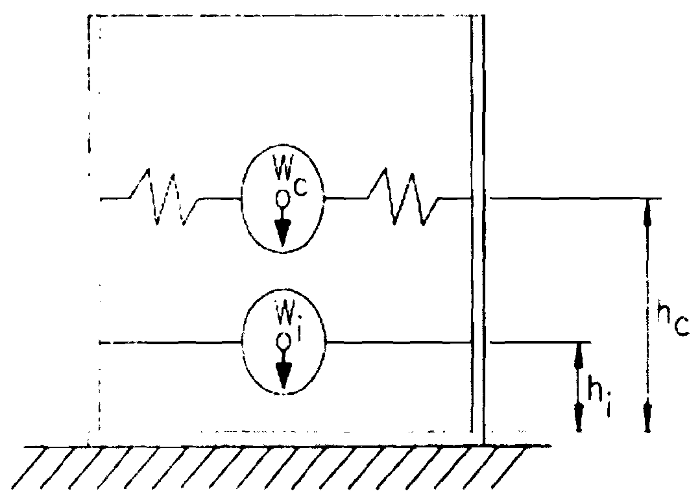
Pools in Reactor Building



REACTOR BUILDING

Modeling of Pool Water in Seismic Analysis

- Fluids contained in pools are commonly modeled as mass-spring system made of convective (sloshing) and impulsive (rigid) components.

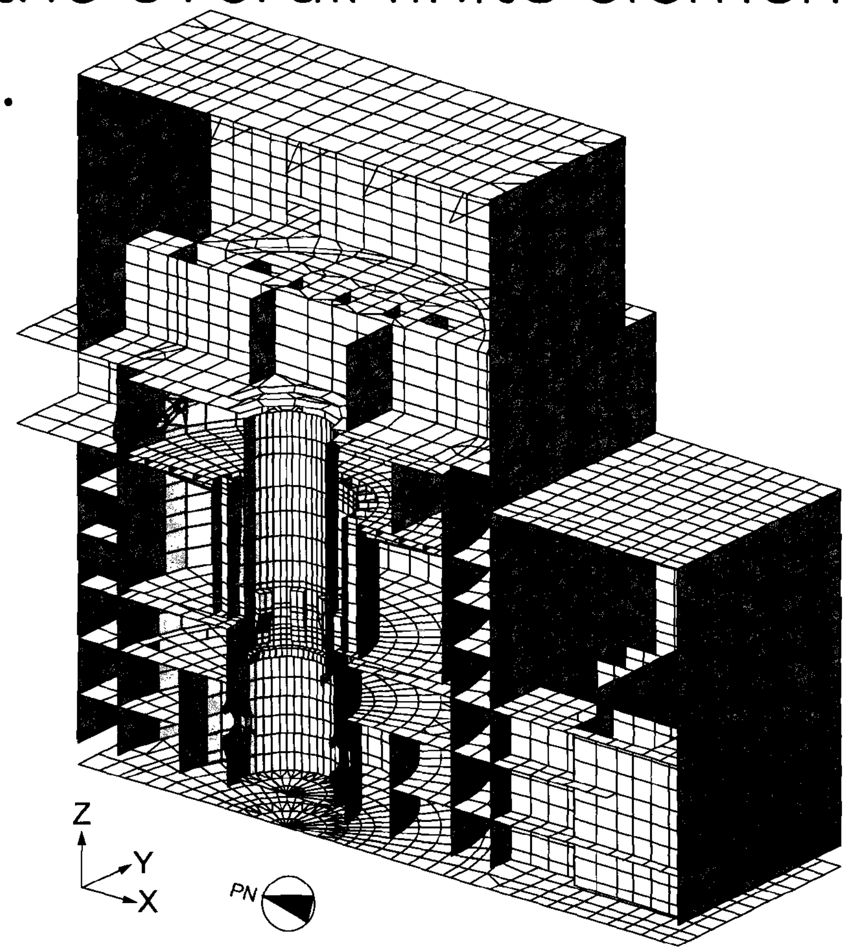


Modeling of Pool Water in Seismic Analysis

- Sloshing component responds in very low frequencies (typically<0.5 Hz) where no structural modes of vibration exist.
- Impulsive component responds in unison with the pool structure and its effect is treated as added mass.
- The sum of masses associated with each component is equal to the the total water mass in the pool.
- For conservatism, the entire water mass of each pool is considered as impulsive mass rigidly attached to structural nodes in the seismic stick model for predicting overall response of the building structure.
- All pools are included in the model, thus the effect of pool interaction is accounted for.

Stress Analysis of Pool Structures

- Each pool structure is explicitly modeled and included in the overall finite element model of the building.



Stress Analysis of Pool Structures

- Input seismic loads consist of
  - Global loads in the form of maximum shear, moment and accelerations calculated from the seismic response analysis.
  - Local loads in the form of hydrodynamic pressures due to convective and impulsive modes on the pool boundaries.
- Resulting stresses are combined with others per required load combinations to meet design code acceptance criteria.

Summary

- Chapter 3 provides design basis of structures, components, equipment and systems.

Dominion

Millstone 3 Stretch Power Uprate

ACRS Meeting
Containment

July 9, 2008

Dominion

Analysis Summary

- ❑ **Containment Analysis Methodology Updated To Current Standards.**

- ❑ **Significant Margin Remains Following SPU.**
  - 3.6 psi containment pressure margin.
  - EEQ profiles essentially unchanged.
  - No impact on current NPSH analysis.
  - Minimum pressure unaffected by SPU.
  - Subcompartment analysis remains bounding

- ❑ **Modifications Made To RSS Pipe Supports To Restore Stress Margins.**

Dominion

Containment Analysis

- Current Long Term Mass and Energy release calculations have not been updated since original licensing.
- SPU long term mass and energy releases incorporates NRC approved methodology updates.
- Containment analysis changed to in-house NRC approved methodology.
- Because of changes in both mass and energy releases and containment methodologies, comprehensive sensitivity studies performed to assure limiting conditions identified.
- Original sensitivity studies repeated as well as new sensitivity studies performed consistent with current approved updated methodologies.

Dominion Containment Analysis

- Ranges of initial conditions expanded for operational flexibility.

- Containment results used for a number of different component evaluations.
  - Containment minimum and maximum design pressure.
  - Maximum containment liner temperature.
  - Maximum pressure and temperature profiles for equipment qualification.
  - Maximum sump temperature at time of recirculation for pump NPSH.
  - Minimum and maximum temperature combinations for pipe stress evaluations.

- Bounding assumptions are dependent upon the component being evaluated.

- Reduction in cold leg temperature for SPU evaluated for impact on subcompartment analysis.

Dominion

Subcompartment Analysis

- For Most Scenarios, The SPU Mass And Energy Releases Are Bounded By The 10% Margin Provided In Current Analysis.
- SPU Analysis Credits Leak-Before-Break For Exclusion of RCS Piping Break In The Steam Generator Cubicle.
- New Analyses Performed For The Pressurizer Surge Line Break.

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Protecting People and the Environment

Final Rule — Consideration of Aircraft Impacts for New Nuclear Power Reactor Designs

George Tartal & William Reckley
Office of New Reactors

1

SRM on SECY-06-0204

- Proposed rule - security assessment requirements for new reactor designs
- Terminate the security assessment rulemaking
- Part 73 rulemaking "sets the adequate protection standard"
- Include aircraft impact assessment requirements in Part 52
- Commission-specified proposed rule language

2

Proposed Rule

- Implemented the Commission's specified rule language
- Developed a technical and legal rationale for the rulemaking
- Published on October 3, 2007 (72 FR 56287)
- Public comment period ended on December 17, 2007

3

Public Comments

- 32 comment letters received
  - 10 from industry
  - 2 from government organizations
  - 12 from non-government organizations
  - 8 from private citizens
- 31 in favor of requiring aircraft impact assessments on nuclear power plants
  - None supported it exactly as proposed

4

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Protecting People and the Environment

Public Comments (cont)

- 8 specific requests for comment
  - Inclusion of impact assessment in application
  - Acceptance criteria
  - Records retention
  - Criteria for judging DC amendments
  - Future Part 50 applicants
  - Requirements in Part 50 or Part 52
  - Design approvals and manufacturing licenses
  - Scope of the design to be evaluated

5

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Public Comments (cont)

- Overall need to address aircraft impacts
- Applicability
- Adequate protection/beyond-design-basis
- Aircraft characteristics
- Assessment
- Evaluation
- Issue Resolution / Implementation
- Safeguards/Sensitive Information
- Compliance with NEPA
- Other comments

6

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Final Rule Requirements

- Applicability – § 50.150(a)
  - New construction permits (CP)
  - New operating licenses referencing new CP
  - New standard design certifications (DC)
  - New standard design approvals (DA)
  - Combined licenses not referencing DC/DA/ML
  - Combined licenses referencing noncompliant DC
  - Manufacturing licenses (ML) not referencing DC/DA
  - Manufacturing licenses referencing noncompliant DC

7

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Protecting People and the Environment

Final Rule Requirements (cont)

- Assessment – § 50.150(b)(1)
  - Assess effects of impact of a large, commercial aircraft
  - Identify and incorporate those design features and functional capabilities that avoid or mitigate, to the extent practical and with reduced reliance on operator actions, the effects of the aircraft impact on core cooling capability, containment integrity, spent fuel cooling capability, and spent fuel pool integrity
  - NRC expects to endorse NEI guidance

8

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Protecting People and the Environment

Final Rule Requirements (cont)

- Aircraft impact characteristics – § 50.150(b)(2)
  - Large, commercial aircraft used for long distance flights in the U.S.
  - Aviation fuel loading for such flights
  - Impact speed and angle of impact considering pilot (in)experience and low altitude
  - More specific aircraft impact parameters will be provided in guidance

9

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Protecting People and the Environment

Final Rule Requirements (cont)

- Content of application – § 50.150(c)
  - Description of design features and functional capabilities identified in assessment
  - Description of how those design features and functional capabilities avoid or mitigate, to the extent practical and with reduced reliance on operator actions, the effects of the aircraft impact

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Final Rule Requirements (cont)

- Control of changes – § 50.150(d)
  - If licensee changes § 50.150-compliant information included in PSAR/FSAR
    - Redo that portion of assessment addressing changed feature or capability
    - Describe how the modified features and capabilities avoid or mitigate, to the extent practical and with reduced reliance on operator actions, the effects of the aircraft impact

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50.54(hh) & 50.150 Relationship

- 50.54(hh)
  - Preparatory actions for potential or actual aircraft attack; Guidance and mitigative strategies for loss of large areas due to fires/explosions (ICM Orders B.5.a and B.5.b)
  - Focused on human actions and operational considerations
  - Necessary for adequate protection
- 50.150
  - Assessment of newly designed facilities to avoid or mitigate the effects of aircraft impacts
  - Focused on design considerations
  - Not necessary for adequate protection

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Rulemaking Schedule

Proposed rule published	10/3/2007
Public comment period ends	12/17/2007
Final rule ACRS briefing	7/09/2008
Final rule to EDO	9/16/2008
Final rule to Commission	9/30/2008

13

Security Rulemaking for Nuclear Power Plants

ACRS Presentation

July 9, 2008

Discussion Topics

- Power Reactor Security Rulemaking
  - Currently with EDO (since 6/30/08)
  - Provided status to ACRS on June 4
- Portions requiring ACRS review
  - § 50.54(hh) "Mitigative Strategies and Response Procedures for Potential or Actual Aircraft Attacks"
  - § 73.54 "Protection of Digital Computer and Communication Systems and Networks"
  - § 73.58 "Safety/Security Interface Requirements for Nuclear Power Reactors"
- This briefing focuses on these three pieces
  - Staff requests ACRS to provide the Commission its views on acceptability of these three portions of the final rule package

10 CFR 50.54(hh) Overview

- § 50.54 (hh) Mitigative Strategies and Response Procedures for Potential or Actual aircraft Attacks
  - Reflects section B.5.a and B.5.b of 2002 ICM order
  - Staff believes that § 50.54(hh) is implementing the order requirements (i.e., it is not the intent to go beyond order requirements)
  - Initially in noticed proposed App C - moved to § 50.54, "Conditions of License" – re-noticed as supplemental proposed rule (published in Federal Register 4/10/2008)

- (hh)(1) Preparatory actions taken in the event of a potential aircraft attack (i.e., B.5.a)

- (hh)(2) Mitigative strategies for addressing the loss of large areas due to fires and explosions from beyond design basis events (i.e., B.5.b)

§ 50.54(hh)(1) Preparatory Actions

- § 50.54(hh)(1):
  - Authenticate threat source
  - Maintain communication with source
  - Contact onsite and offsite organizations
  - Take onsite actions to mitigate impact
  - Reduce visual discrimination
  - Disperse equipment and personnel
  - Recall of personnel
- Guidance under development – uses existing advisories and information (DG 50XX)

§ 50.54(hh)(2) Mitigating Measures

- § 50.54(hh)(2)
  - Fire fighting
  - Operations to mitigation fuel damage
  - Actions to minimize releases
- These requirements map into 14 strategies in current license conditions for all current licensees
- Current licensees are in compliance
- Guidance under development

§ 73.54 Protection of Digital Computer and Communication Systems and Networks

- Cyber threat was included as part of DBT (§ 73.1) issued March 2008
- These requirements were in proposed § 73.55(m)
- Moved to stand-alone section in part 73
- Required to development and submit cyber plans for NRC review and approval (intro paragraph)

§ 73.54 Cyber Security

- § 73.54(a) – Identifies protected digital assets:
  - Safety-related and ITS functions
  - Security functions
  - EP functions
  - Supports systems
- Protect from cyber attacks that:
  - Adversely impact data and/or software
  - Deny access to systems, services, data
  - Adversely impact operation of digital assets
- (b) Requires analysis to identify assets to be protected
- (c) Program design requirements
  - Protect digital assets identified in (b) – ensure function not adversely impacted
  - Apply defense-in-depth
  - Mitigate adverse affects of attacks

§ 73.54 Cyber Security Cont'

- (d) Provides training, risk mgmt, and change control requirements
  - Cyber awareness training
  - Evaluate and manage cyber risks
  - Control changes to ensure that cyber performance objectives are maintained
- (e) Cyber plan requirements
  - Cyber plan is required – requirements for content
  - Cyber plan must be submitted for NRC review and approval)
- (f) – (h) Procedures, Reviews, Records

- Guidance: DG 5022 Cyber Security Programs for Nuclear Facilities
  - Completed 6/1/08 (OUO)
  - Distributed to appropriate licensees
  - Public meeting in July

§ 73.58 Safety/Security Interface

- Addresses part of UCS petition (PRM 50-80)
- Makes explicit what is already implicitly required by regulation
- (b) Requires licensees to assess/manage potential for adverse interactions between security ↔ safety
- (c) Scope – Planned and emergent activities
- (d) Conflicts – Communicate conflicts and take compensatory and mitigative actions

§ 73.58 Safety/Security Interface

- DG 5021 Safety/Security Interface
  - Published in Federal Register July 24, 2007
  - Public Meeting held; comments received & under consideration

Summary

- Final rule requirements for § 50.54(hh), § 73.54, and § 73.58 are complete and with EDO
- Draft guidance developed – work continues to finalize guidance
- Expect further interactions with ACRS to finalize guidance (meet with ACRS when finalizing guidance)
- Staff requests ACRS provide its opinion on acceptability of the final rule provisions to the Commission

Containment Review

Ahsan Sallman
Containment & Ventilation Branch
Division of Safety Systems
Office of Nuclear Reactor Regulation

Containment Review

- Primary Containment Functional Design
- Subcompartment Analyses
- Mass and Energy Release
- Combustible Gas Control in Containment
- Containment Heat Removal
- Pressure Analysis for ECCS Performance Capability
- Reconsideration of Generic Letter 96-06

Summary of Staff Review

- RS-001, "Review Standard for Power Uprates," was followed as guidance
- Applied NRC-approved analytical methods
- RAIs were satisfactorily answered
- Applicable GDCs were satisfied
- SRP acceptance criteria were satisfied
- Met 10 CFR 50 requirements

Primary Containment Functional Design

- Application of GOTHIC 7.2a methodology to MPS3 approved by SE, dated August 30, 2006
- Conservative initial conditions for LOCA and MSLB
- Analyzed a spectrum of breaks for LOCA and MSLB

Primary Containment Functional Design Continued

- Conclusions
  - Limiting short-term LOCA & MSLB peak pressure & temperature are bounded by the containment design conditions
  - Limiting long-term LOCA & MSLB pressure & temperature responses are evaluated to be acceptable from the standpoint of EQ

Subcompartment Analyses

- NRC has approved leak-before-break (LBB) methodology for MPS3 contained in the license renewal SE – NUREG-1838
- Used LBB criteria for selection of pipe breaks
- Conclusion
  - Sufficient margin in the differential pressures across the subcompartment walls under SPU conditions

Mass and Energy Release Analyses for LOCA & Secondary Pipe Ruptures

- Analyzed a spectrum of breaks for LOCA based on NRC-approved methods: LOCA blowdown & reflood (WCAP-10325-P-A & WCAP-8264-P-A) and post-reflood (DOM-NAF-3-0-0-P-A
- Analyzed a spectrum of secondary breaks based on NRC approved methods in WCAP-8822, WCAP-8822-01-P-A, WCAP-8822-02-P-A, and WCAP-7907-P-A

Mass and Energy Release Analyses for LOCA & Secondary Pipe Ruptures Continued

- Used conservative assumptions and inputs to maximize M&E release
- Conclusion
  - Staff reviewed and agreed with the licensee's evaluation of LOCA M&E release

Combustible Gas Control in Containment

- SER, dated June 29, 2005, removed hydrogen recombiners & monitoring system from Tech Specs as per 10 CFR 50.44 and RG 1.97
- Conclusion
  - SPU does not impact combustible gas control in containment

Containment Heat Removal

- Containment accident pressure was not used for calculation of NPSHA for RSS pumps
- Input parameters are conservative or the same as the current analysis
- Used GOTHIC methodology to calculate the maximum sump temperature

Containment Heat Removal Continued

- Conclusion
  - RSS pumps NPSHA requirement is met

Pressure Analysis for ECCS Performance Capability

- Used conservative initial conditions for calculating the minimum containment backpressure transient
- Calculated containment pressure transient bounds the transient used in the ECCS performance analysis
- Conclusion
  - ECCS performance capability is unaffected by SPU

Reconsideration of Generic Letter 96-06

- GL 96-06 states, "Thermally induced overpressurization of isolated water-filled piping sections in containment could jeopardize the ability of accident-mitigating systems to perform their safety functions and could also lead to a breach of containment integrity via bypass leakage. Corrective actions may be needed to satisfy system operability requirements."

Reconsideration of Generic Letter 96-06 Continued

- Licensee reviewed GL 96-06 for piping system penetrating containment along with its relief valves as a part of SPU system design pressure & temperature evaluation
- Conclusion
  - No hardware changes are necessary for SPU conditions

Summary

- Applicable GDCs were satisfied
- SRP acceptance criteria were satisfied
- Met 10 CFR 50 requirements

Dominion

Millstone 3 Stretch Power Uprate

ACRS Meeting
Fuel & Safety Analysis

July 9, 2008

Dominion Fuel

- No Change In Fuel Design.

- Core Will Be 100% RFA-2. There Are No Mixed Core Issues.

- SPU Achieved Through An Increase In Feed Batch Size.

- Reduction In Peaking Factor Design Limits To Increase DNBR Margin.

Dominion

Fuel Design

Parameter	Current	SPU
Fuel Type	Robust Fuel Assembly (17x17 RFA-2)	Unchanged
Burnable Poison	Integral fuel burnable absorber (IFBA)	Unchanged
Blankets	Annular pellets in axial blankets	Unchanged
Maximum Enrichment	5 weight percent	Unchanged

Dominion

Initial Conditions

- Currently Analyzed For A Single Nominal Temperature At 100% Power With No Margin For Coastdown.
- SPU Analyses Performed For A 8°F Nominal Temperature Band At 100% Power And 10°F Coastdown For Added Operational Flexibility.
- SPU Operation Selected At The Same Nominal Temperature As Current Operation.
- Modest Increase In Hot Leg Temperature Will Have A Small Impact On The Life Of SG Tubes And Other Hot Leg Alloy 600 Components.
- Modest Decrease In Cold Leg Temperature Will Have A Modest Improvement In The Life Of Reactor Vessel Head Penetrations And Other Cold Leg Alloy 600 Components.
- Pressurizer Level Chosen To Balance Margins For Operation And For Design Basis Transients.

Dominion

Safety Analysis

- All plant specific safety analyses re-analyzed at SPU conditions.

- **Significant Safety Analysis Margins Remain After SPU.**
  - 11.7% DNBR margin.
  - 419 ^{o}F LB LOCA PCT margin.
  - 1007 ^{o}F SB LOCA PCT margin.
  - 3.6 psi containment pressure margin.

- **Margins Achieved Through Plant Modifications.**

- **Methodologies Updated To Current Approved Standards.**

- **SPU has small impact on currently approved AST radiological analyses.**

Dominion

DNBR Margin

- Included In Margin Management Program.
- Current DNBR Margin Used To Address Upper Plenum Anomaly.
- Modifications Will Address Upper Plenum Anomaly And Re-establish DNBR Margin.
- Preliminary Analyses Used To Establish Target SPU DNBR Margin.
- Final Analyses Resulted In Small Change To Target SPU DNBR Margin.

Dominion Pressurizer Overfill

- Included In Margin Management Program.

- Initial Pressurizer Level Selected To Balance The Margin To Letdown Isolation For Routine Reactor Trips And Margin To Pressurizer Overfill For Design basis transients.

- Current Limiting Event Is The Inadvertent ECCS Actuation At Power.

- Hardware Modification Proposed To Significantly Reduce The Severity Of The Pressurizer Overfill Rate For This Event.

- Modification Eliminates The Inadvertent ECCS Actuation As The Limiting Event. The New Pressurizer Overfill Limiting Event Changed To The CVCS Malfunction Event, Currently Considered Bounded And Not Explicitly Analyzed For Millstone Unit 3.

Dominion

Design Basis

- All Design Requirements Are Met At SPU Conditions.

- In General, SPU Has A Small Impact On The Results.

- In General, Safety Analysis Margins Are Essentially The Same With Significant Margin Remaining After SPU.

Dominion

Radiological

- Alternate Source Term Methodology Submitted In 2004 And Approved By The NRC In 2006.
- 2004 Submittal Included 6.5% Power Increase In Anticipation Of SPU.
- Alternate Source Term Methodology Resulted In Significant Increase In Available Radiological Dose Margins.
- For SPU, All Events Have Been Re-analyzed To Take Into Account The Additional 0.5% Power Increase.
- SPU Impact On Radiological Analysis Is Small.

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Status of Resolution of GI-199

Marty Stutzke, Division of Risk Analysis
Jon Ake, Division of Engineering
Office of Nuclear Regulatory Research

July 10, 2008

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Issue Summary

- The staff has identified that the estimated seismic hazard levels at some current CEUS operating sites might be higher than seismic hazard values used in design and previous evaluations.
  - Review of updates to seismic source and ground motion models provided by ESP applicants.
  - Review of recent USGS seismic hazard estimates.
  - Comparison to seismic hazard estimates developed under the IPEEE program.

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Background

- Concern identified May 2005 in a memo from NRR to RES.
  - NRR concluded that the seismic designs of operating plants in CEUS still provided an adequate level of protection.
- GI-199, "Implications Of Updated Probabilistic Seismic Hazard Estimates In Central And Eastern United States On Existing Plants," was opened in June 2005.
- In February 2008, a screening panel concluded that GI-199 should proceed to the safety and risk assessment phase.
- Public meeting held on February 6, 2008.
- NRC and EPRI are finalizing an MOU to share seismic research information.

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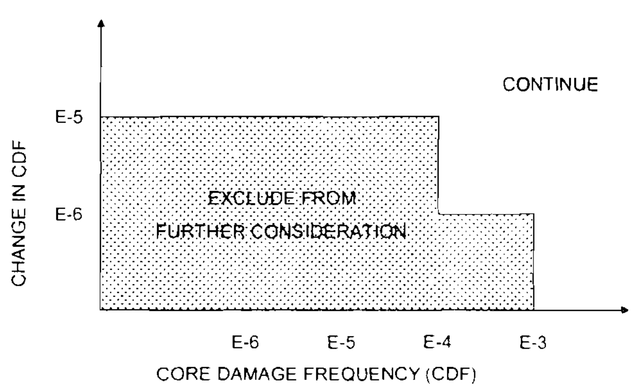
MD 6.4 Screening Criteria



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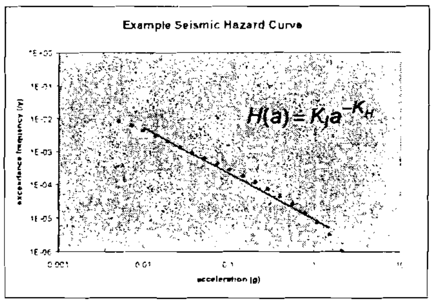
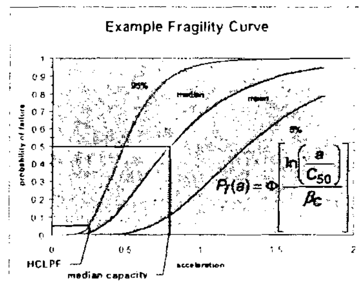
Approximate SCDF





$$SCDF = \int_0^{\infty} H(a)\left[\frac{dP_f(a)}{da}\right]da = K_I \exp\left[-K_H \ln(C_{50}) + \frac{1}{2}K_H^2\beta_C^2\right]$$

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Path Forward

- Complete the safety/risk assessment
- Hold public meeting on safety/risk assessment stage progress
- Convene the Safety/Risk Assessment Panel
- Draft memo and report for Safety/Risk Assessment Panel

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TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT

POWER UPRATE
ACRS

CONTAINMENT OVERPRESSURE (COP)

Rockville, Maryland
July 10, 2008

Introduction

- COP Part of BFN Current Licensing Basis for Appendix R and LOCA

- EPU Results in Additional COP need in Licensing Basis Analysis

- ACRS Concerns from Unit 1 105% Review
  - Magnitude and duration of Appendix R event
  - Feasibility of protecting second RHR pump
  - Consider external initiators when using risk-informed arguments for the Appendix R COP credit
  - Realistic long-term LOCA analysis needs to be supported by more defensible sensitivity analyses

Introduction

- Actions Taken to Address ACRS Concerns on Appendix R COP
  - July 2007 meeting with NRC Staff
  - Fire area analysis undertaken to compare COP needs for realistic fire versus Appendix R analysis
  - Deterministic analysis to provide risk insight
    - Submitted November 15, 2007
  - Followed-up with NPSH analysis for limiting cases
    - Submitted June 12, 2008

Alternate Fire Shutdown Analysis

- Appendix R Rule Based Fire
  - Prescribed Appendix R fire damage
    - Loss of all equipment not meeting generic separation criteria
    - Fire damage not based on analysis
  - Fire damage overly conservative for many areas of the plant

Alternate Fire Shutdown Analysis

TVA

- Fire Hazards Analysis
  - Supplement to Appendix R
  - Fire damage by analysis versus prescribed fire damage
  - Screen based on fire protection parameters
    - Combustible loading
    - Volume of fire area
    - Detection/Suppression
    - Ignition sources
  - 22/39 fire areas screened out
    - Fire limited to ignition source
    - No wide spread fire damage
  - 17 fire areas screen in
    - Evaluated for equipment availability

Alternate Fire Shutdown Analysis

- Equipment Availability Analysis
  - All equipment in fire area assumed lost in 17 fire areas
  - Unaffected equipment used per EOIs
  - Offsite power credited where unaffected
    - BOP systems available in many areas
  - 15/17 areas do not need COP
    - Sufficient equipment available to limit pool temperature
  - Only 2 fire areas need some COP
    - Electrical Board Rooms

Alternate Fire Shutdown Analysis

- NPSH Analysis – Limiting Fire Areas
  - Minimum equipment
    - Emergency depressurization
    - Reactor water level maintained with BOP
    - One RHR pump for containment cooling
  - Pool water volume increased during event
    - Peak pool temperature lower
    - Pool level/elevation head increase
  - Relaxed NPSHr based on revised vendor report
    - Based on shorter operating time consideration
  - Termination of drywell coolers not required

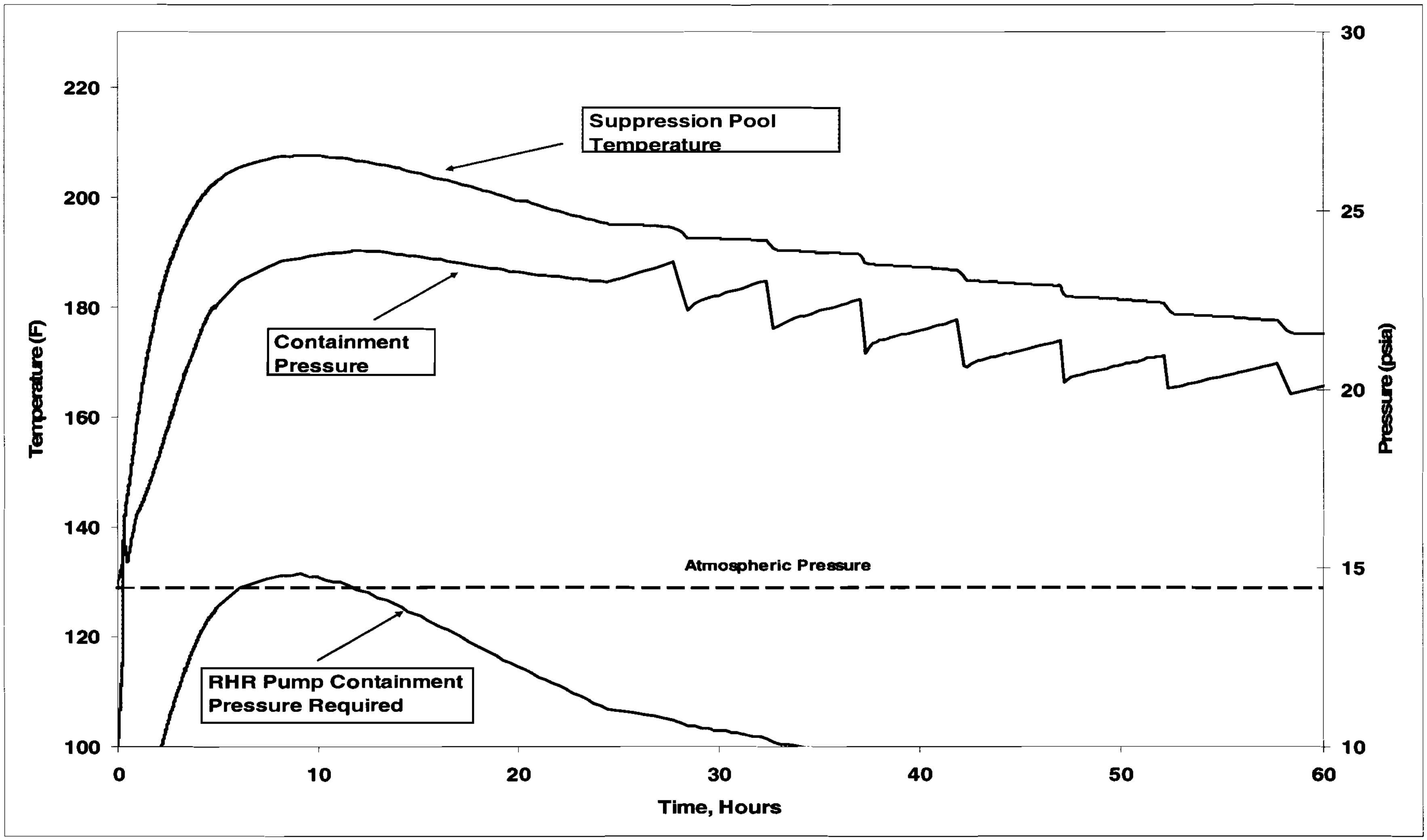
Alternate Fire Shutdown Analysis

TVA

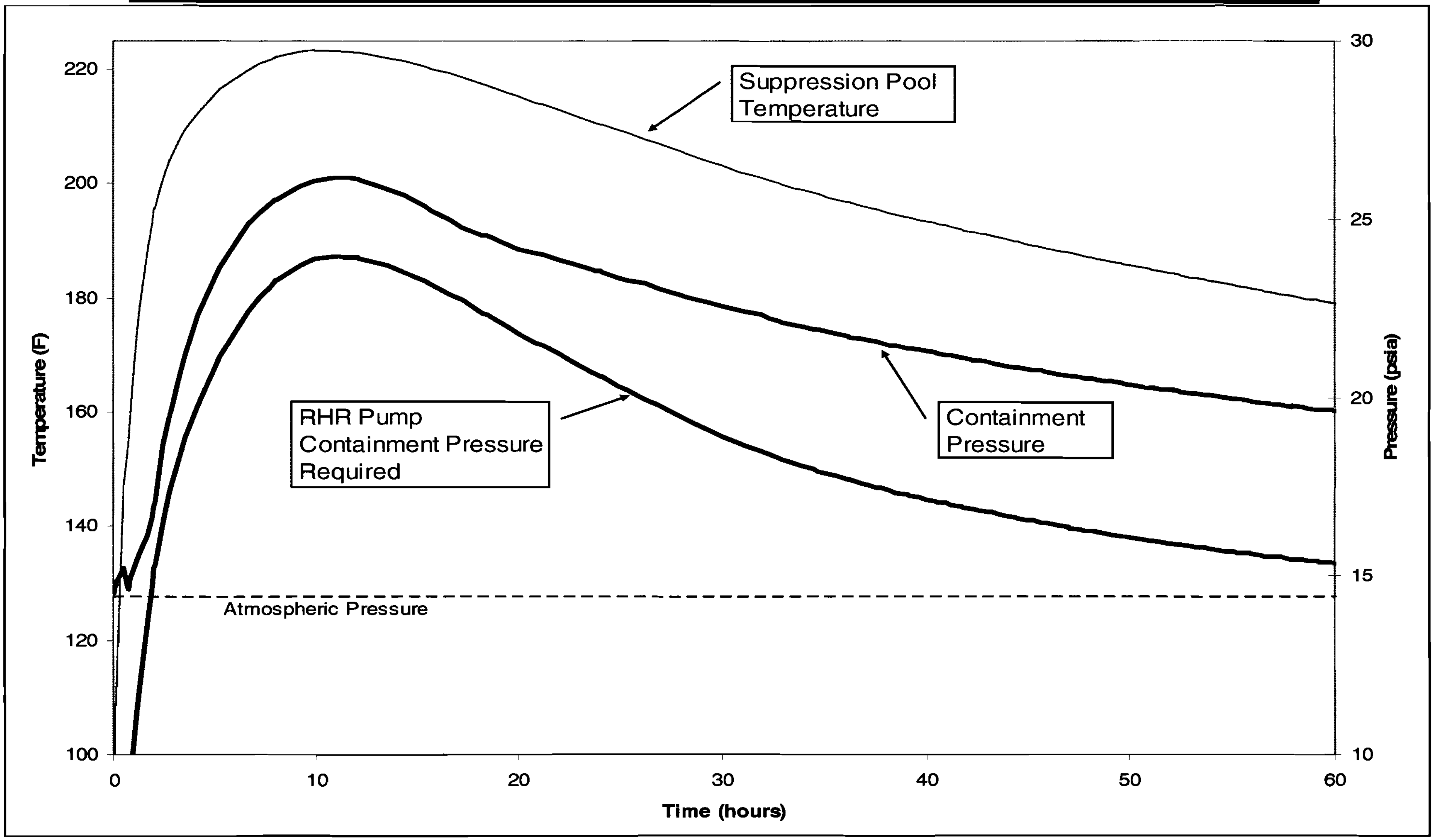
- NPSH RESULTS
  - Significant reduction in COP required
    - ½ psi COP
    - 6 hour duration
  - Significant COP margin
    - Minimum 8 psi
  - Core cooling not dependent on COP
    - Core cooling by BOP Systems
    - Defense-In-Depth consideration

COP Available and COP Required
Alternate Fire Hazards Analysis

TVA



COP Available and COP Required Licensing Basis Appendix R



Specific ACRS Recommendations

- Feasibility of Second RHR Pump for Appendix R
  - Extensive resources
    - Physical location of switchgear
    - Shared electrical system
    - Support equipment
      - Valves/controls
      - Diesel generator/controls
      - RHRSW pump/valves
    - Significant modifications
    - Significant program and licensing changes
  - Minimal safety benefit
    - Demonstrated by fire hazards analysis

Specific ACRS Recommendations

- Consideration of External Events in Fire Risk Evaluations
  - Fire risk insights from deterministic fire hazards analysis
    - Not a PRA analysis
    - Bounding fire is assumed

Appendix R Conclusions

- Licensing Basis Analysis Complies With Appendix R and Demonstrates a Success Path
- COP Magnitude and Duration for Appendix R Driven by Rule Based Assumptions
- Fire Hazards Analysis Shows Reduced or No Dependency on COP

Additional ACRS Issue

- Bias and Uncertainty in Realistic LOCA
  - Realistic LOCA used to build PRA model for COP
  - Use of 95% non-exceedance values
  - Use of probability distributions
  - Use of conservative licensing basis methods
  - Realistic NPSH analyses biased conservatively

Closing

- Concluding Remarks

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Seismic Research Program 2008-2011

ACRS Meeting
July 2008

1

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Program Overview

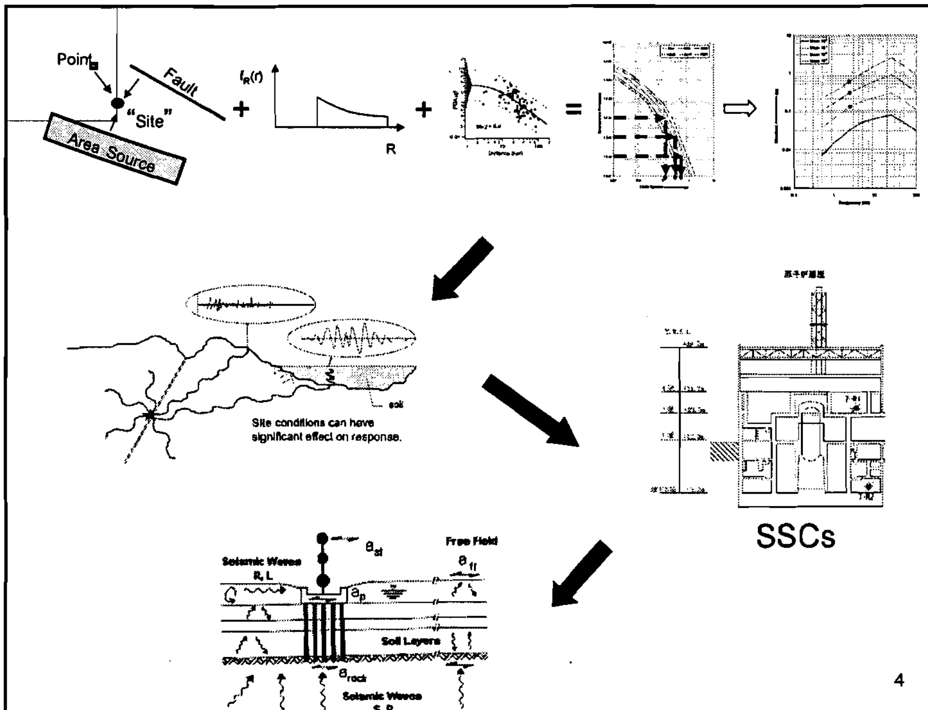
- Pu blicly available research plan
- T argeted on regulatory needs
  - Developed with NRO, NRR & NMSS
  - Efforts in many research areas to be coordinated with industry
- De veloped to both push forward the science and the level of regulatory stability
- In corporation of performance-based approaches

2

Program Philosophy

- Short- and long-term research topics
- Systematic & integrated
  - Moves from topic-by-topic approach to integrated research
  - Focused on issues with highest uncertainties
  - Risk informed plan that fills in gaps
- Cost effective
  - Piggy backing and partnering
  - Universities
  - Using “next-generation” approaches
- Highest Quality

3



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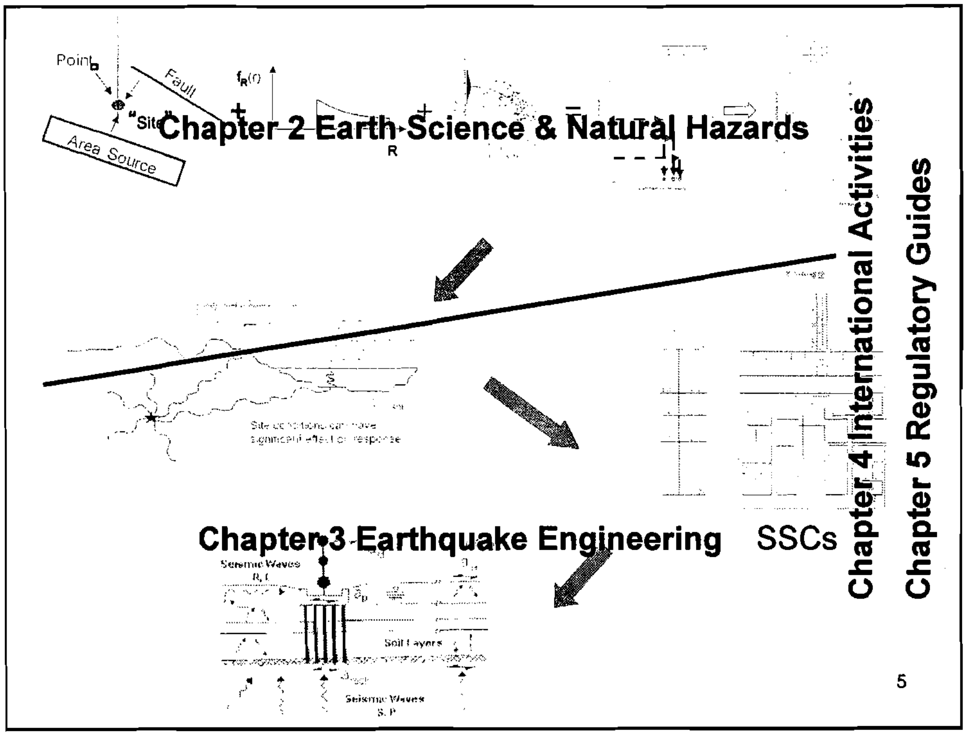
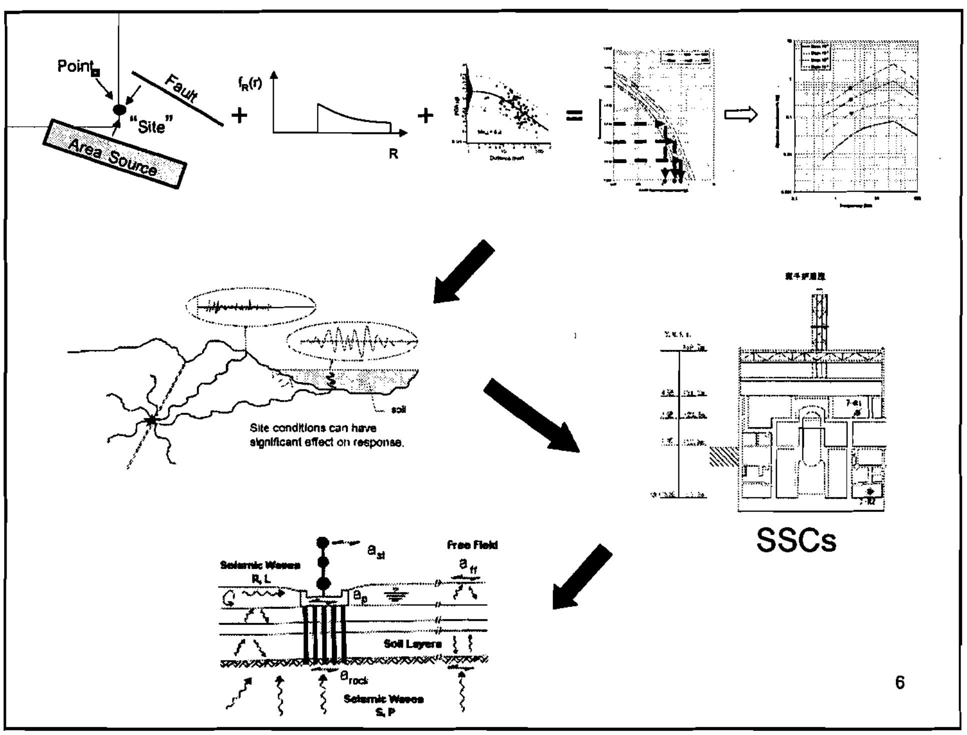
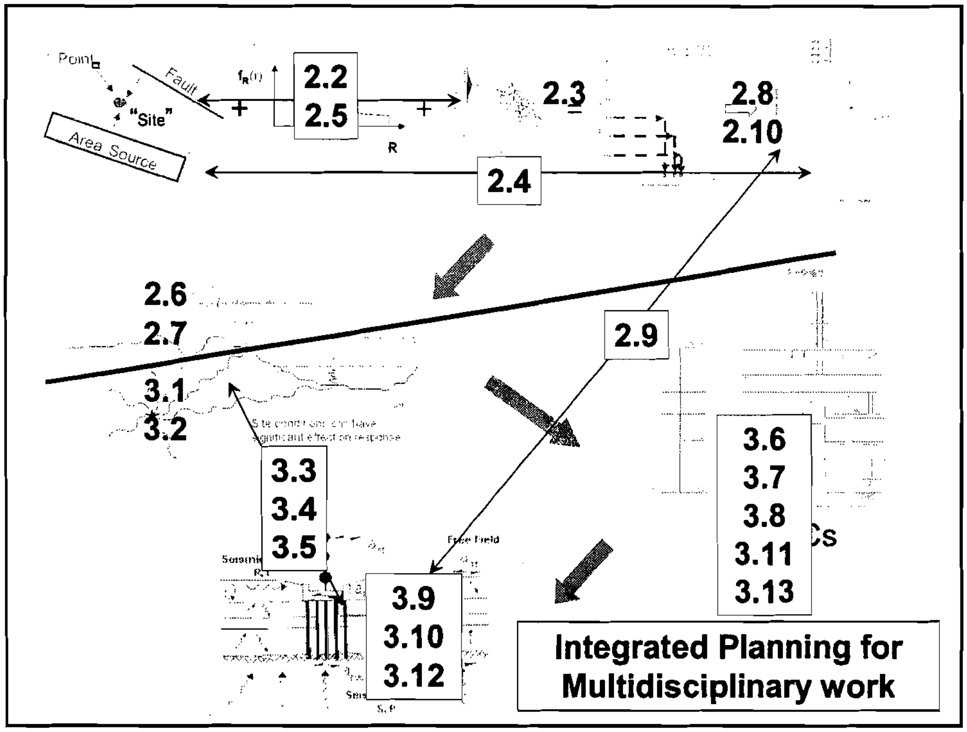

Point
Fault
"Site"
Area Source
Chapter 2 Earth Science & Natural Hazards
Chapter 3 Earthquake Engineering
SSCs
Chapter 4 International Activities
Chapter 5 Regulatory Guides
5


Point
Fault
"Site"
Area Source
$f_R(r)$
R
soil
Site conditions can have significant effect on response.
SSCs
Seismic Waves
R, L
Free Field
Soil Layers
Seismic Waves
S, P
6


Point
Fault
"Site"
Area Source
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2.5
R
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2.6
2.7
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3.2
3.3
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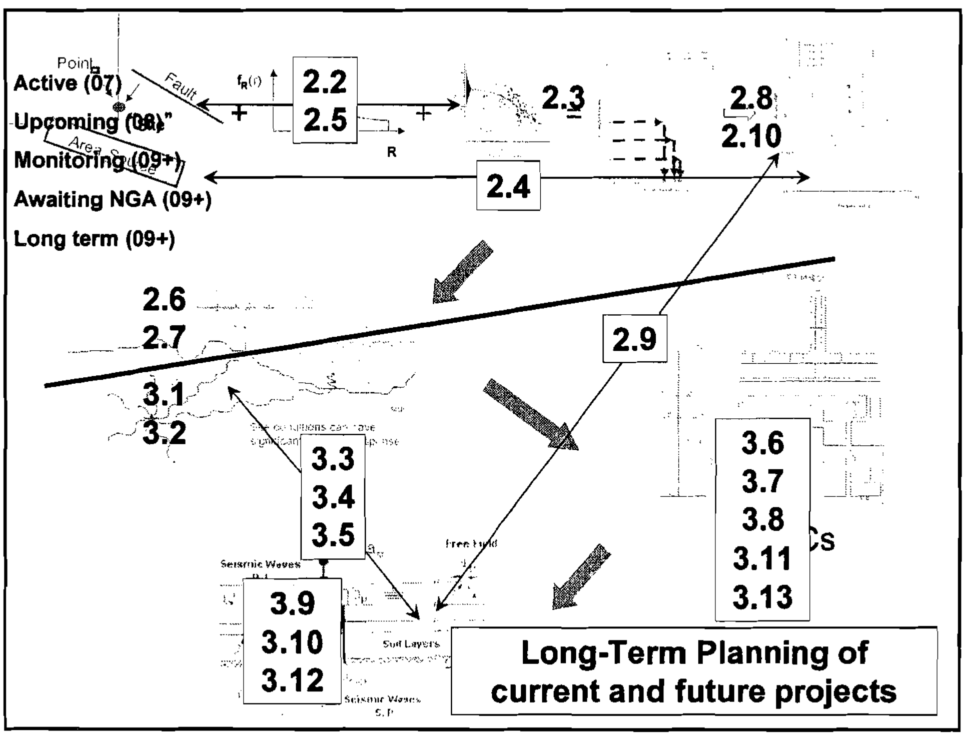
3.11
3.13
3.9
3.10
3.12
Free Field
Integrated Planning for
Multidisciplinary work


Active (07)
Upcoming (08)
Monitoring (09+)
Awaiting NGA (09+)
Long term (09+)
2.2
2.5
R
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3.7
3.8
3.11
3.13
3.9
3.10
3.12
Seismic Waves
Soil Layers
Seismic Waves
Long-Term Planning of
current and future projects

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Workshops and "Next Generation" approaches

- NRC initiated early seismic hazard work
- Seismic research moving from the development of individual tools and methods
  - Different databases, gray literature, proprietary reports, proprietary software
- The now mature field is moving to integration through workshops, working groups and "next generation" approaches
  - Common databases & inputs, community consensus, documentation of thought processes, outliers & uncertainties better understood

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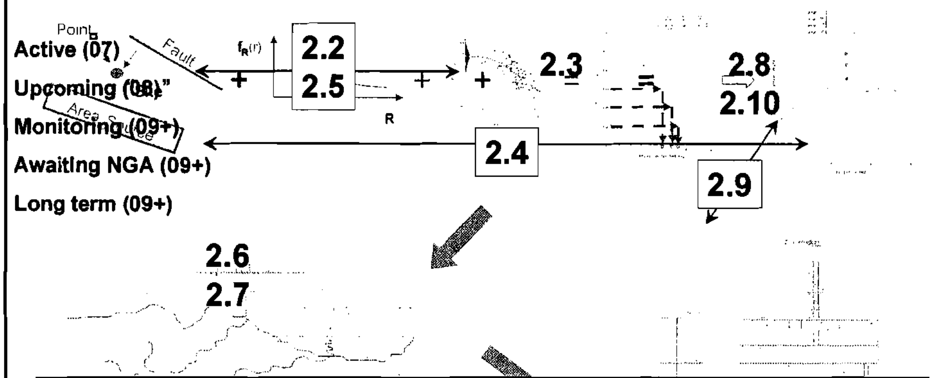
Workshops and "Next Generation" approaches

- Consistent, complete, and agreed upon data sets and information
- Key experts in the research area involved
- "Next Generation" implies fundamental redevelopment of technical tools or approaches
- Both best estimates & estimates of uncertainties

10



2.2 Mmax ('07)
CEUS SSC Project (08)
Sensitivity & uncertainty study to address issues and prioritize research needs ('08)
2.3 Next Generation Attenuation (NGA-East) ('07)
2.4 Application of SSHAC Guidelines ('07)

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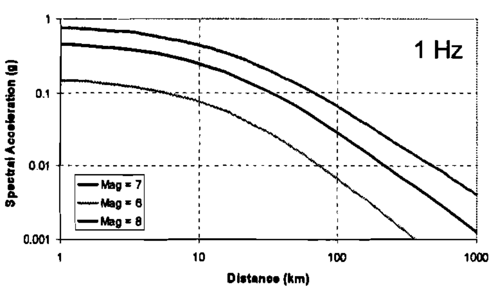
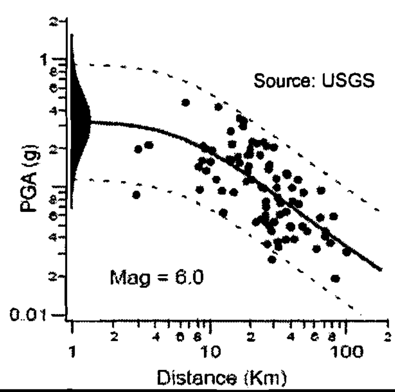
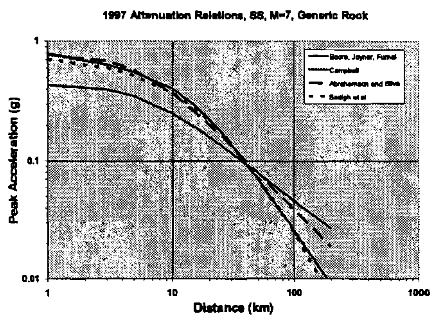
Seismic Wave Amplitudes vs. Distance

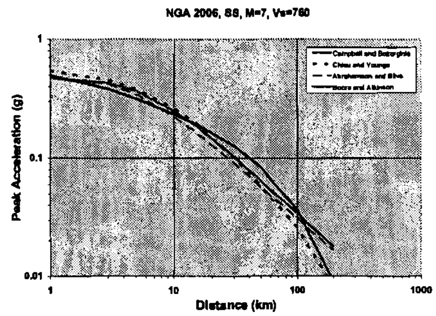




NGA-East

Before NGA-West **After**





- Went from ad hoc relationship development to unified approach
- Mutually agreed upon databases, technical bases & assumptions
- Epistemic uncertainties reduced and characterized
- Broad community consensus (removed points of contention)

13

NGA-East

- Follows up on original NGA project to address
- Approach
  - Standard agreed upon assumptions
  - Standard and complete database
  - Development program undertaken to scope project and bring in multiple agencies
  - Cooperative agreement with multiple agencies (DOE, NEHRP, EPRI)
  - USGS in-kind participation in development project
- Currently doing preliminary work
  - Technical Basis for assumptions
  - Development of earthquake record database

14

SSHAC Guidelines

- "Recommendations for PSHA: Guidance on Uncertainty and Use of Experts" NUREG/CR-6372 by Senior Seismic Hazard Analysis Committee
- General framework work well, but limited details
- Need recommendations on how to use and to update
- Much has been learned in trying to apply SSHAC
  - Yucca Mountain (two level 4s – seismic and volcano)
  - PEGASOS (level 4)

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SSHAC Guidelines

- New project created to develop practical recommendations for application of the SSHAC guidelines
  - Lessons learned
  - How and when to update
  - Understanding and characterizing uncertainties (epistemic and aleatory)
- Will develop a NUREG document to accompany the SSHAC guidelines

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CEUS Seismic Source Characterization project for Nuclear Facilities

- Major study with participation from NRC, DOE, EPRI, USGS, and other US specialists
- NRC SSHAC guidelines level-3 study
- 3 year project starting summer 2008
- Project to develop a seismic source database for the CEUS to be used as a baseline for all PSHA
- NPP applications will still study local sources
- Coordinating a CEUS SSC "International Observers Program" to allow international specialists to observe the US project firsthand

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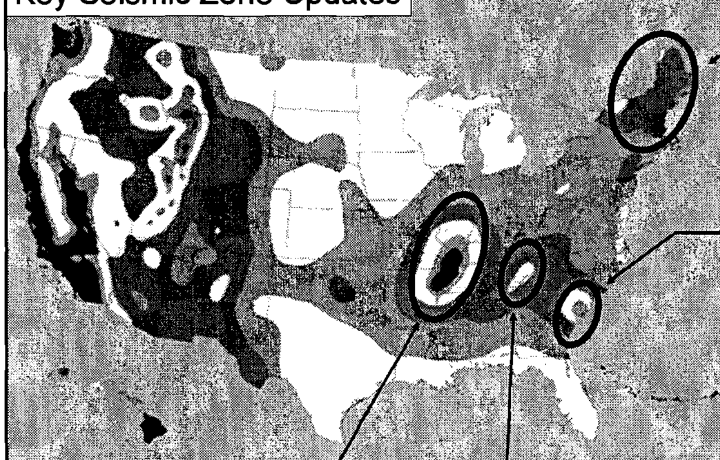
Key Seismic Zone Updates

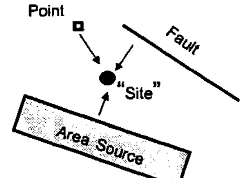


Northeast (NESZ)

Charleston (CSZ)

New Madrid (NMSZ)

Eastern Tennessee (ETSZ)



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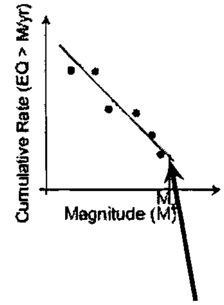
M_{max} Workshop



- M_{max} is largest magnitude for a source
- Issue for area sources in CEUS for long return periods
- Limited technical basis due to lack of systematic, integrated evaluation of existing models and new data
- Follows "best practices" for seismic workshops
  - Sensitivity study
  - Foundation document compiled & sent to participants before workshop for review. Also downloadable at USGS.
  - All key researchers sponsored, but open to anyone
- Results incorporated into USGS database

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New US NRC Projects to Assess Seismic Hazard in CEUS

Source Characterization →	Central and Eastern US Seismic Source Characterization project for Nuclear Facilities (CEUS SSC)
Ground motion prediction equations →	Next Generation Attenuation Relationships for the Central and Eastern (NGA-East)
Framework for large PSHA studies →	Recommendations for Application of the SSHAC Guidelines

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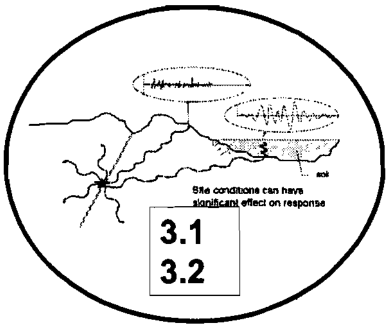
3.1 Random Vibration Theory
3.2 Site response methods



Active Projects

- Multiple methods accepted in NUREG 6728
  - Theoretical framework but few details
  - Only recently used
  - Implementation differs between practitioners
  - Focus on better understanding
- Multiple modeling tools currently in use
  - Non-linear, SHAKE, and RVT methods
  - Comparison of methods
  - Developing public RVT software with PEER

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Site SSE (GRMS) compared to CSDRS

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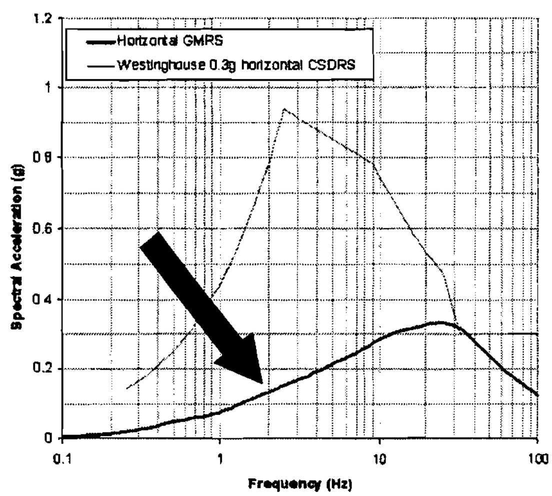
GRMS is based on site characterization and it is determined from detailed seismic hazard studies

CSDRS is based on engineering design of a plant



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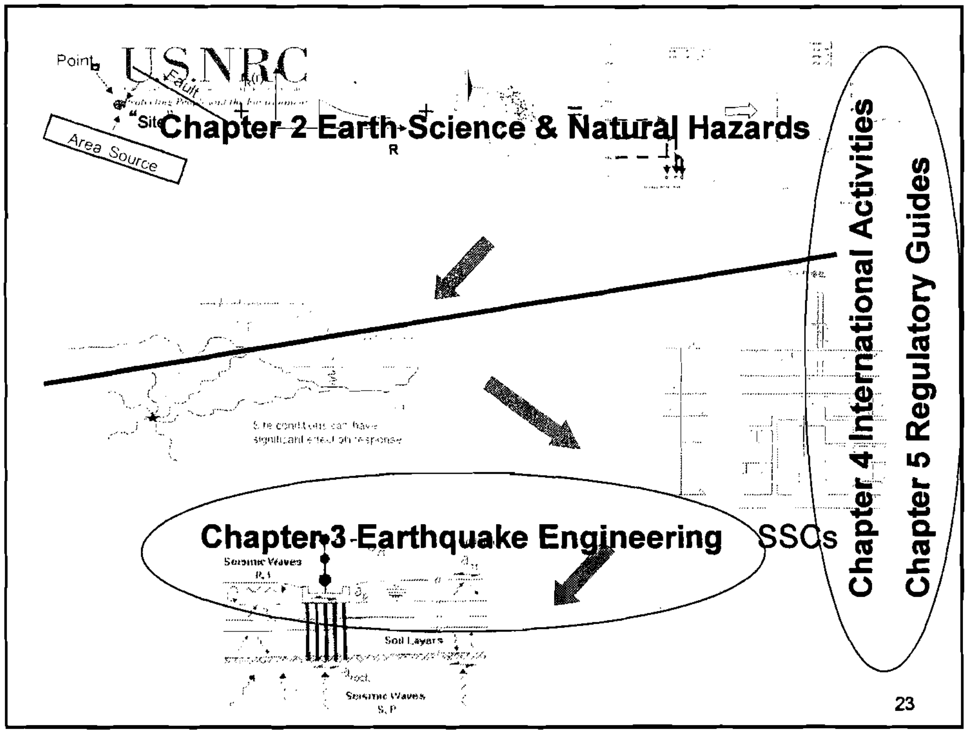

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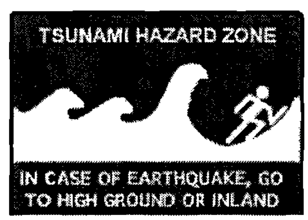
Tsunami



- Continued tsunami source development with USGS
  - Phase 1 report being used by NRC staff and industry
  - Phase 2 underway
- Continued development of modeling capabilities with USGS and NOAA

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Summary

- Seismic Research Plan publicly available
- Key drivers are both the advancement of state of practice AND greater regulatory stability
- New projects focused on seismic hazard in CEUS
- Integrated risk-informed approach
- Both short- and long-term planning
- Focus on consensus products
  - Community-wide consensus
  - Multiple stakeholders & sponsors

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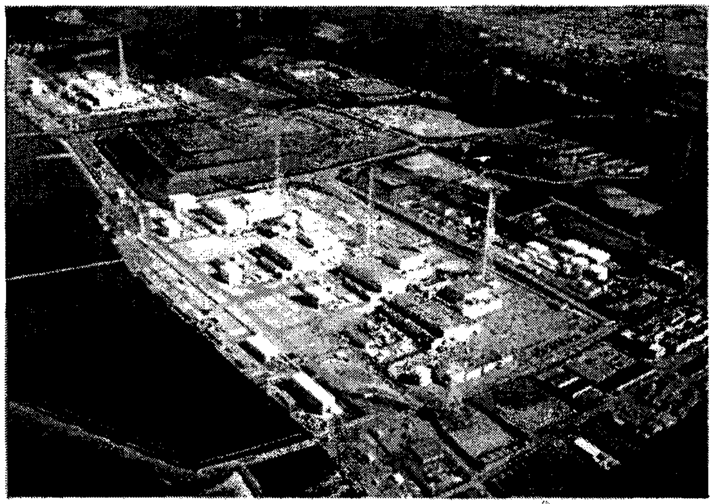
Impact of the Niigataken Chuetsu-oki Earthquake to the Kashiwazaki Nuclear Power Plant

Yong Li
Senior Geophysicist
NRO/DSER/RGS2
July 10, 2008

Agenda

- Niigataken Chuetsu-oki Earthquake and Kashiwazaki-Kariwa (KK) nuclear power plant
- Impact of the earthquake vibration to the Nuclear Power Plant (NPP)
- Major findings from post-earthquake activities at the plant site

A Bird's View of KK Site from South



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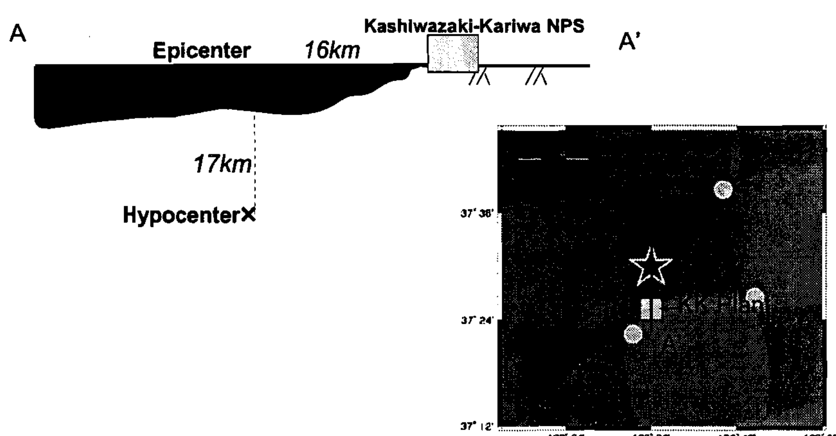
Earthquake Location relative to KK plant



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Design Peak Ground Acceleration vs. Observed (gal – cm/sec^2)

Unit	North - South direction (design / measured)	East - West direction (design / measured)	Vertical direction (design / measured)
1	274 / 311	273 / 680	235 / 408
2	167 / 304	167 / 606	235 / 282
3	192 / 308	193 / 384	235 / 311
4	193 / 310	194 / 492	235 / 337
5	249 / 277	254 / 442	235 / 205
6	263 / 271	263 / 322	235 / 488
7	263 / 267	263 / 356	235 / 355
Divide gal by 1000 to approximate g-value			

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Common Cause Failures and Potential Vulnerabilities

- Settlement and soil failures
  - ✓ Breakages of underground fire protection piping joints
  - ✓ Deformations and cracks in the ducts connected to the main stacks
  - ✓ Deformations and fire on the Unit 3 house transformer secondary bus

- Potential for adverse interaction with safety related equipment
  - ✓ Water leakage through building penetrations
  - ✓ Water leakage through leaky seals
  - ✓ Damage to thermal insulator of SLC piping

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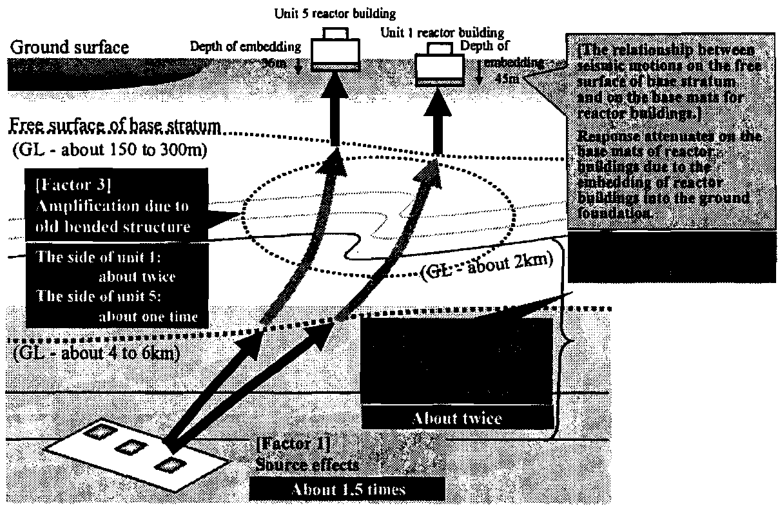
Factors to Amplify Seismic Motions



From TEPCO 6/12/2008 presentation

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Ongoing Activities at the KK Plant

- Non-destructive testing for hidden damages
- Assessment of new ground motion for reevaluation of the plant safety
- Evaluation of the plant design upgrading

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Interim Staff Guidance on Seismic Issues associated with High Frequency Ground Motion

Dr. Manas Chakravorty
Structural Engineering Branch 2
July 10, 2008

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Background

- Updated ground motion models for earthquakes in the CEUS.
- ESP reviews identified that site-specific ground motion may not be enveloped by certified design response spectra for some sites

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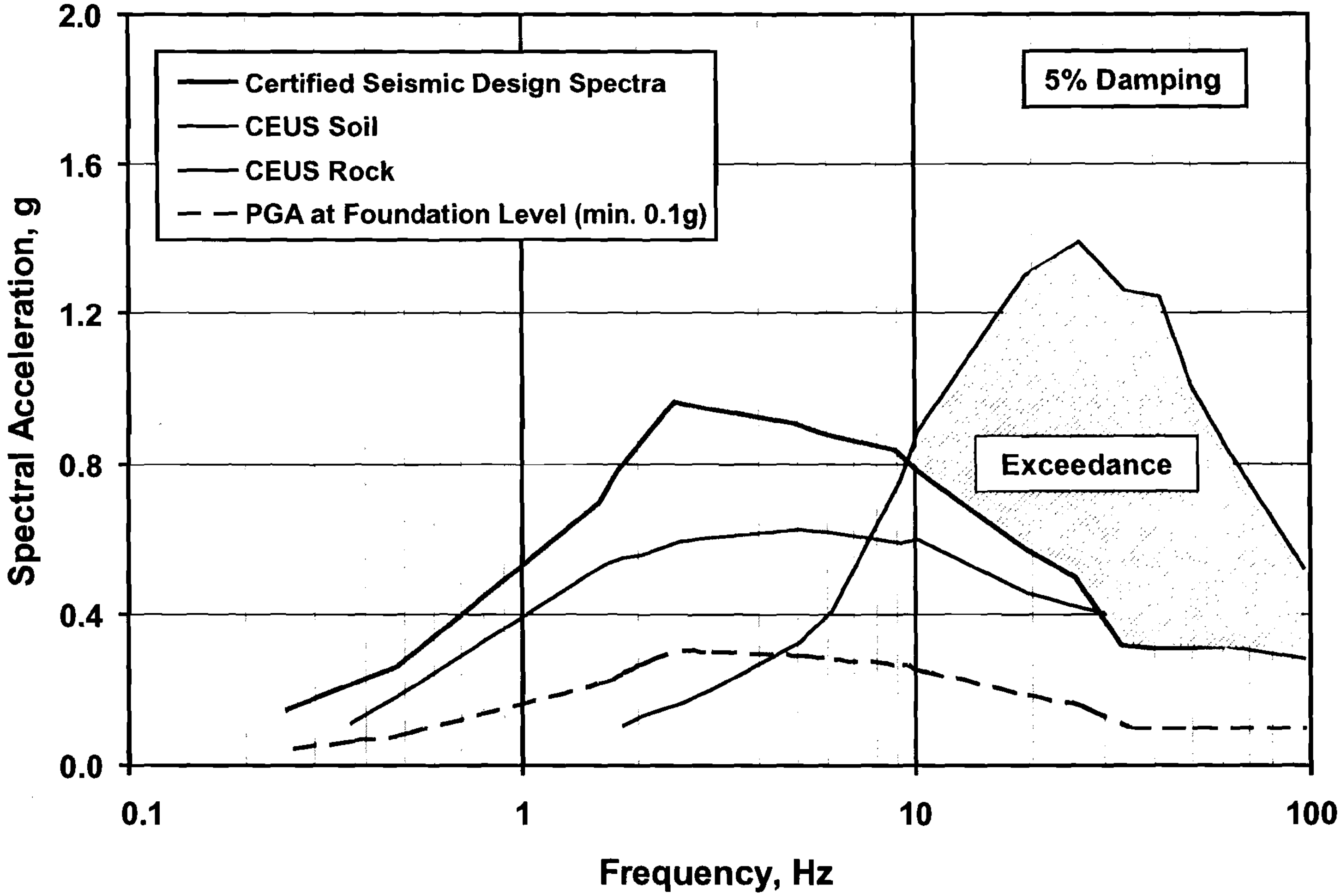
CSDRS & GMRS COMPARISON



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Resolution

- The updated SRP Section 3.7.1 " Seismic Design Parameters" provided the framework
- Issued ISG for the implementation of the SRP framework

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Issues addressed in the ISG

- Definitions of various ground motions
- Guidance on the use of the different ground motions and seismic instrumentation
- Staff position on the use of limited dynamic testing data

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Issues addressed in the ISG

- Guidance on evaluation of HF exceedance
  - Inclusion of incoherency in structural seismic response analysis
  - Screening of HF sensitive SSC's
  - Evaluation of screened components

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Summary

- SRP 3.7 & ISG provides a high frequency review framework
- AP1000 Topical report has been submitted and currently under staff review
- ESBWR has used CSDRS that envelop both soil and rock sites