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

August 11, 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 558th FULL COMMITTEE MEETING HELD ON DECEMBER 4-6, 2008 IN ROCKVILLE, MARYLAND

The minutes of the subject meeting were certified on January 19, 2009 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

January 19, 2009

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 558th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), DECEMBER 4-6, 2008

I certify that based on my review of the minutes from the 558th 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	01/ 19 /09	01/ 19 /09

OFFICIAL RECORD COPY

CERTIFIED Date Certified: 01/19/2009

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During its 558th meeting, December 4-6, 2008, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports, letter, and memorandum:

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

- Final Review of the Vogtle Electric Generating Plant Early Site Permit Application and Limited Work Authorization Request and the Associated Safety Evaluation Report, dated December 22, 2008

- Technical Basis and Rulemaking Strategy for the Revision of 10 CFR 50.46(b) Loss of Coolant Accident Embrittlement Criteria for Fuel Cladding Materials, dated December 18, 2008

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

- Interim Letter 6: Chapters 7 and 14 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Certification of the ESBWR Design, dated December 22, 2008

MEMORANDUM
Memorandum to R. W. Borchardt, Executive Director for Operations, NRC, from Edwin M. Hackett, Executive Director, ACRS:

- Proposed Rule Regarding Enhancements to Emergency Preparedness Regulations, dated December 10, 2008

MINUTES OF THE 558th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS DECEMBER 4-6, 2008 ROCKVILLE, MARYLAND

The 558th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on December 4-6, 2008. Notice of this meeting was published in the *Federal Register* on November 19, 2008 (72 FR 69681-69682). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting agenda. 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.

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 announced that Mr. Charles Brown, Jr. is an official member of the committee with an expertise in digital instrumentation and control.

II. Chapters 7 and 14 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 Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff and General Electric - Hitachi Nuclear Energy (GEH) to discuss Chapters 7 and 14 of the NRC Staff's SER with Open Items associated with the ESBWR Design Certification Application. GEH staff presented an overview of Chapter 7, which covers the Distributed Control and Information System (DCIS). The DCIS is divided into the Q-DCIS, which controls reactor trip, Emergency Core Cooling Systems (ECCS), and other safety systems, and the N-DCIS, which performs all other functions. The Q-DCIS has four divisions which are independent and are physically, electrically, and data isolated from each other as well as from the N-DCIS. The NDCIS can receive data from the Q-DCIS but cannot influence the Q-DCIS. The system is designed such that if one division is taken out of service for maintenance and a random single failure occurs in another division, the remainder of the DCIS will actuate all of the ECCS. GEH staff also discussed the ESBWR main control room and the remote shutdown system.

The NRC staff described the staff's review of Chapters 7 and 14 of the ESBWR Design control Document (DCD). The staff followed Chapters 7 and 14 of the Standard Review Plan and discussed IEEE-603 compliance, the life cycle design process, the setpoint methodology, diversity and defense-in-depth, and data communication. The staff stated that most of the remaining open items are related to clarification and consistency and that no significant technical issues remain.

Several Committee members commented that the applicant's presentation described a design in significantly more detail than what was presented in the DCD, giving an impression that the design process had progressed considerably beyond the DCD description. The NRC staff indicated that it is necessary to distinguish between what is conceptual and what was submitted for approval. The NRC staff will discuss this distinction further in a future meeting. The Committee issued a letter to the Executive Director for Operations on this matter, dated December 22, 2008, concluding that the applicant has an acceptable process for developing the Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for the Initial Plant Test Program and that the Design Acceptance Criteria (DAC) for the DCIS are incomplete. The Committee recommended that the Tier 2 DCD include additional detailed information on the architecture of the instrumentation control system and that appropriate ITAAC and DAC be added to the Tier 1 DCD.

III. Early Site Permit Application and the Final SER for the Vogtle Nuclear Plant

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

The Committee met with representatives of the NRC staff and Southern Nuclear Operating Company (SNC) to discuss the application submitted by SNC for the Vogtle Early Site Permit (ESP), SNC's request for a Limited Work Authorization (LWA), and the associated NRC staff SER. T he Vogtle ESP application is different from other ESP applications in two significant

ways. The Vogtle ESP application references parameters of a specific reactor design, the Westinghouse AP1000, rather than relying on a plant parameter envelope. The Vogtle ESP application also proposes a complete and integrated emergency plan, including emergency planning ITAAC, rather than providing only major features of an emergency plan. The SNC LWA requests permission to begin limited work on construction activities at the site, including placement of backfill, construction of retaining walls, and installation of foundation mudmats. SNC staff provided an introduction to the Vogtle ESP application and the request for an LWA. This included brief descriptions of the contents of the application and LWA, the remaining schedule for NRC staff review of documents, the site location and the nature of open items in the draft SER. SNC described the pre-construction activities requested in the LWA, including placement of engineered backfill, a concrete mudmat and water proofing, and mechanically stabilized concrete retaining walls.

NRC staff provided brief summaries of the closure of open items in the draft SER in the technical areas of hydrology, meteorology, seismology and geotechnical engineering and emergency planning. SNC provided a great deal of additional information on geotechnical engineering properties of the soils at the Vogtle site and their response to a seismic event to close several open items in the draft SER.

Several Committee members noted that seismicity is the most important site safety issue. Seismicity at the proposed Vogtle site is dominated by the Charleston seismic zone. The predicted ground motion response spectrum at the proposed Vogtle site is not bounded by the seismic design response spectrum certified for the AP1000 reactor. Therefore, this difference should be addressed in the future combined license application for new reactors at the proposed Vogtle site. The Committee issued a report to the NRC Chairman on this matter, dated December 22, 2008 recommending that the Vogtle ESP and LWA be granted.

IV. Status of Staff Activities Associated with Potential Revision to 10 CFR 50.46 (b)

[Note: Mr. Christopher Brown was the Designated Federal Official during this portion of the meeting.]

The Committee met with representatives of the NRC staff and Electric Power Research Institute (EPRI), concerning activities related to 10 CFR 50.46(b) rulemaking. The staff's presentation described the strategy for revising 10 CFR 50.46(b) fuel performance criteria. The current rule uses prescriptive criteria to ensure post quench ductility (PQD) in the cladding during loss-of-coolant accidents (LOCAs). The proposed rule would use performance based requirements to ensure PQD in the cladding. In addition, the rule would permit the use of Zirconium alloys that meet the performance acceptance requirements. These new requirements would be based on a sound understanding of the phenomena controlling cladding embrittlement and would be applicable to low and high burnup fuel for both large-break and small-break LOCAs. The rule would permit the use of current and future zirconium alloys that meet the performance acceptance requirements without the need for exemptions. The staff also mentioned that an Advance Notice of Proposed Rulemaking (ANPR) may be issued in parallel with the completion of remaining confirmatory research. The staff believes that the ANPR process will enhance

public participation and facilitate formal stakeholder interaction on the rulemaking while confirmatory data is gathered. EPRI representatives stated that industry is supportive of the NRC's overall objective to revise 10 CFR 50.46(b) to a performance-based rule; however, they expressed concerns about the implementation cost, requirements to use two-sided oxidation, and periodic testing on breakaway oxidation. They also indicated that since studies completed-to-date indicate no significant safety concerns with respect to the current design basis, there is no need to rush to rulemaking.

The Committee issued a report to the NRC Chairman on this matter, dated December 18, 2008, concluding that there are sufficient data and understanding of the cladding embrittlement phenomena to justify and proceed with rulemaking. The Committee recommended that the rule include the optional testing program to allow licensees to demonstrate compliance with PQD criteria on an alloy-specific and temperature-specific basis.

V. NRC Staff's Initial White Paper on Containment Overpressure Credit Issue

[Note: Mrs. Zena Abdullahi was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff regarding the White Paper on the use of containment accident pressure in determining the available net positive suction head (NPSH) for safety system pumps which provide: (1) core cooling and coverage, (2) suppression pool cooling, and (3) containment cooling. At issue is the reliance of safety systems on containment accident pressure to perform their design functions and successfully mitigate LOCA (10 CFR 50.46 and Appendix K), anticipated transients without scram (10 CFR 50.62), station blackout (10 CFR 50.63), and Appendix R fire events.

The NRC staff discussed the technical basis supporting its position that crediting containment accident pressure when determining the available NPSH is acceptable provided the licensees demonstrate the use of conservative assumptions for the LOCA event, including minimizing the containment pressure and maximizing the suppression pool temperature. The staff stated that its assessment indicates that neither the magnitude of containment accident pressure nor the duration of the credit needed was important. The staff also supported its conclusions by citing its evaluation of the: (1) robustness of the pumps to withstand cavitations; (2) potential loss or decrease of the containment accident pressure assumed in the calculation of the available NPSH; and (3) integrity of the seals and penetrations. The ACRS members noted that crediting of containment overpressure was first allowed in response to the emerging issue of BWR suction strainer clogging during a loss-of-coolant accident. In fact, this is an excellent example of the role maintaining NPSH margins play in ensuring safety in both addressing newly discovered issues and in mitigating additional unknowns or nonconservatisms. The ACRS members described their concerns of the staff's positions on: (1) allowing reduced or negative NPSH margins; (2) placing no limits on the amount or duration that containment accident pressure is credited; (3) justifying pump cavitations, and (4) accepting reliance on operator intervention to manage cavitations under some circumstances. The members also discussed the fact that the risk analysis does not include the risk associated with the reduction in NPSH

margin and the impact of the duration the credit is needed. Also missing was a sensitivity analysis for quantifying the impact of the cited conservatisms in the overall LOCA containment evaluation. In its concluding remarks, the NRC staff acknowledged that the technical differences appear to remain and that they intend to pursue writing a commission paper.

The Committee plans to write a report to the NRC Chairman on this matter during its February 2009 meeting.

VI. Overview of the Human Reliability Analysis (HRA) Research Activities

[Note: Dr. Hossein Nourbakhsh was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff regarding HRA research activities and the joint NRC/EPRI plan for evaluating different human reliability analysis models. The NRC staff indicated that its current focus of HRA research includes benchmarking HRA methods to understand strengths and weaknesses of existing methods and determine ways to improve them.

In a November 8, 2006 Staff Requirements Memorandum (SRM), resulting from the October 20, 2006 meeting with ACRS, the Commission directed the Committee to "work with the staff and external stakeholders to evaluate the different Human Reliability models in an effort to propose either a single model for the agency to use or guidance on which model(s) should to be used in specific circumstances." The staff briefed the Committee on its plan and status of current activities to address the November 8, 2006 SRM. The staff indicated that Phase 1 of the joint NRC/EPRI plan (to be completed by April 2009) includes reviewing the use of HRA in decision making and establishing a common terminology and HRA process. During Phase 2 of this effort (to be completed by May 2009), insights from Phase 1 and the International HRA Empirical Study will be utilized to recommend a consolidated HRA approach. In Phase 3 (to be completed by September 2010), a single HRA method or a small set of methods will be developed for use by NRC and Industry. In the final Phase 4 (to be completed by September 2010), the methods will be tested and guidance and training materials will be developed. This was an information briefing no Committee action was necessary. The Committee plans to continue its discussions on Human Reliability Analysis Research Activities in future meetings.

VII. Draft Policy Statement on Defense-in-Depth for Future Nuclear Reactors

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

The Committee met with representatives of the NRC staff regarding the development of a draft policy statement on defense-in-depth for future plants. In order to avoid any impact on the established existing regulatory processes for the light water reactors (LWRs), the staff would apply the policy statement to only non-LWR advanced reactors. The current staff effort was initiated as a part of its development of the risk-informed (RI), performance-based (PB), and technology neutral alternative regulatory framework. The staff obtained Commission direction and sought public comments through an advance notice of proposed rulemaking on the RI and

PB alternative to 10 CFR Part 50. This rulemaking is currently on hold. Previous work by international groups, the industry, and the ACRS was mentioned. The Commission has directed the staff to engage members of the public, ACRS, the industry, and other stakeholders as they develop this policy statement. The Commission also directed that the insights gained from the development of the Next Generation Nuclear Plant licensing strategy and completion of the Pebble Bed Modular Reactor pre-application review be used in developing this policy statement. The issues that received considerable Committee attention include: definition of defense-indepth, its objective and principles, the use of PRA to distinguish between a desirable design/program requirement vs. defense-in-depth measures, and implementation issues. The Committee members noted that licensing information available on Fort St. Vrain or the Clinch River Breeder Reactor could be used to test the draft policy. This was an information briefing and no Committee action was necessary. The Committee plans to review the development of the defense-in-depth policy statement.

VIII. 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 November 26, 2008, to conclusions included in the October 22, 2008, ACRS report on the status of resolution of Generic Safety Issue-191, "Assessment of Debris Accumulation on PWR Sump Performance."

  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 May ACRS Meeting

Member assignments and priorities for ACRS reports and letters for the December 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 March 2009 were discussed and the objectives were to:

- 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 (SRM)

In the January 8, 2009 SRM resulting from the November 7, 2008 ACRS meeting with the Commissioners, the Commission states the following:

+ The staff should consider what has been learned from the analyses of PWR sump performance and determine if issues have arisen that call for revising BWRs.

+ With regard to power uprates for BWRs consistent with pervious Commission direction, the staff should continue working to resolve the differences of opinion between the Committee and the staff concerning containment overpressure credit, and as necessary and appropriate, provide policy decision papers to the Commission if a resolution cannot be reached.

Browns Ferry Units 1, 2, and 3 Extended Power Uprate Applications

The staff plans to provide a draft Safety Evaluation (SE) report for Browns Ferry Units 1 and 2 in early April in support of a Subcommittee meeting in May and full Committee meeting in June. [There is a possibility a complete SE may not be available for Units 1 and 2 in April.]

For Unit 3, the steam dryer information may not be available until late (October/November) 2009. There were some discussions among the staff about providing a partial SE to the ACRS in April for discussion at the May Subcommittee and June full Committee meetings. After the steam dryer information is made available to the Committee, it needs to review only that information and provide a final report to the Commission. The staff would like to know whether the Subcommittee/full Committee will be willing to review partial SE for Unit 3 and possibly for Units 1 and 2 in May and June respectively.

It should be noted that during its October 20, 2006 meeting with the Commission, the Committee stated the following:

ACRS will review the extended power uprate application for Browns Ferry Units 1, 2, and 3 after receiving the complete Safety Evaluation report.

Biennial ACRS Report on the NRC Safety Research Program

The biennial ACRS report on the NRC Safety Research Program is due to the Commission on March 15, 2010. Drs. Shack and Powers will have the lead in coordinating the preparation of the report. Assignments for the members as well as format, content, and schedule for providing input to the report will be provided to the members during the March ACRS meeting.

Quality Assessment of Selected NRC Research Projects

During its November 2008 meeting, the Committee selected the following research projects and Panels for quality assessment in FY2009:

+ NUREG/CR-6964, “Crack Growth Rates and Metallographic Examinations for Alloy 600 and Alloy 82/182 from Field and Laboratory Materials Testing in PWR Environments” Panel: Armijo (Chair), Abdel-Khalik and Ray

+ NUREG/CR-XXXX, “Diversity and Defense-in Depth for Digital Instrumentation and Control Systems”
  Panel: Brown (Chair), Apostolakis and Sieber

The Committee report is provided to the RES Director in October of each year. Since the Committee needs to prepare its biennial report on the NRC Safety Research Program this year, Dr. Powers proposed that the Committee complete its Quality Assessment report in July 2009.

Tour of the Mitsubishi and Westinghouse Simulators in Pittsburgh

Several ACRS members and ACRS staff are scheduled to tour the Westinghouse simulator on February 18 and the Mitsubishi simulator on February 20, 2009. On February 19, 2009, a Subcommittee meeting is scheduled to discuss selected Topical reports associated with US-APWR. Proposed schedule for the Subcommittee meeting and an itinerary for touring the simulators are attached.

Reappointment of an ACRS Member

The Commission has reappointed Dr. Shack for a fifth term. He joins the elite group of members [Drs. Siess (24 yrs.), Okrent (24 yrs.), and Kerr (20 yrs.)] who served 20 or more years on the Committee.

ACRS Meeting With the Commission

The ACRS is scheduled to meet with the Commission between 1:30 and 3:30 p.m., on Thursday, June 4, 2009 to discuss items of mutual interest. A list of proposed topics will be provided to the Planning & Procedures Subcommittee and the full Committee during their March meetings.

TRACE Thermal-Hydraulic System Analysis Code

During the ACRS meeting with the Commission on November 7, 2008, Dr. Abdel-Khalik made several comments regarding the capability of the TRACE code in evaluating the passive system safety performance. Dr. Sheron, the RES Director, sent a memorandum to the Commissioners responding to the comments made by Dr. Abdel-Khalik at the Commission meeting.

Revision to the ACRS Charter

In approving the renewal of the ACRS Charter, the Commission added a new paragraph stating that the ACRS shall report to and advise the Commission on issues associated with nuclear materials and waste management. This action stems from the merger of the ACNW&M with the ACRS.

Draft Regulatory Guides

The staff plans to issue the following Draft Regulatory Guides (DG) for public comment and would like to know whether the Committee wants to review these Guides prior to being issued for public comment.

Proposed Revision 2 to Regulatory Guide 1.189 (DG-1214), “Fire Protection for Nuclear Power Plants”

Regulatory Guide 1.189 lacked clear guidance with respect to the treatment of fire-induced circuit failures. In SECY-08-0093 "Resolution of Issues Related to Fire-Induced Circuit Failures," the staff proposed a clarification to the NRC's guidance with regard to fire-induced circuit failures. The proposed Revision 2 (DG-1214) is to include the fire- induced circuit-failure clarifications described in SECY-08-0093.

Proposed Revision 4 to Regulatory Guide 1.28 (DG-1215), “Quality Assurance Program Requirements (Design and Construction)”

Regulatory Guide 1.28, Revision 3, endorsed the American National Standards Institute/American Society of Mechanical Engineers (ANSI/ASME) NQA-1-1983 standard, “Quality Assurance Program Requirements for Nuclear Power Plants.” The proposed Revision 4 endorses the ANSI/ASME NQA-1-2008 standard, “Quality Assurance Program Requirements for Nuclear Facilities Applications,” including ANSI/ASME NQA-1a-2008 (which is Addendum A to the NQA-1 standard).

Draft Regulatory Guide (DG) - 5028, “Guidance on Making Changes to Emergency Response Plans for Nuclear Power Reactors”

The NRC staff’s objectives for 10 CFR 50.54(q) are to ensure that licensees (1) follow and maintain the effectiveness of their approved emergency plans, (2) evaluate proposed changes to these plans for their impact on the effectiveness of the plans, and (3) obtain prior NRC approval for changes that would reduce the effectiveness of the plans. These actions are essential if these plans are to continue to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. The purpose of DG-5028 is to provide guidance on the implementation of 10 CFR 50.54(q) with respect to making changes to emergency response plans.

The meeting was adjourned at 12:30 p.m. on December 6, 2008.

grounding resistor to not more than 0.5 ampere and if problems are detected, the neutral grounding resistor may be adjusted to limit the ground-fault current to not more than 1.0 ampere; (5) the 1.0 ampere setting cannot be implemented until MSHA inspects the neutral grounding resistor and determines that the neutral grounding resistor and all ground-fault relays are properly adjusted to protect the high-voltage trailing cable; (6) current transformers used for the ground-fault protection will be the single-window type and will be installed to encircle all three-phase conductors; (7) high-voltage circuits will be protected against short-circuits, overloads, ground faults and under-voltage by a circuit-interrupting device of adequate interrupting capacity; and (8) within sixty days after the Proposed Decision and Order becomes final, proposed revisions for part 48 training plans will be submitted to the District Manager for the area in which the mine is located. These proposed revisions will include, but are not limited to, task training, hazard training, specialized training for qualified persons, and annual refresher training. The petitioner has listed additional procedures in this petition that will be used when the proposed alternative method is implemented. Persons may review a complete description of the procedures and training requirements at the MSHA address listed in this notice. The petitioner asserts that the proposed alternative method will at all times guarantee no less than the same measure of protection afforded the miners as the existing standard.

Docket Number: M–2008–003–M.

Petitioner: Zeigler Chemical & Mineral Corp., 366 North Broadway, Suite 210, Jericho, New York.

Mine: Little Bonanza Mines and Mills, MSHA I.D. No. 42–00876, Harrison Mine, MSHA I.D. No. 42–02484, and the Neal #4 Mine, MSHA I.D. No. 42–02317, all located in Uintah County, Utah.

Regulation Affected: 30 CFR 49.2(b) (Availability of mine rescue teams).

Modification Request: The petitioner requests a modification of the existing standard for its gilsonite mines to permit the use of two mine rescue teams with three members and one alternate for each team. The petitioner states that: (1) The underground mine is a small mine and there is hardly enough physical room to accommodate more than three or four miners in the working places; (2) an attempt to utilize five or more rescue team members in the mine's confined working places would result in a diminution of safety to both the miners at the mine and members of the rescue team; (3) no electric power is used in the mines other than battery operated hard hat lights; and (4) all work below ground is done by hand using compressed air which is fed from above ground. The petitioner asserts that a decision in their favor will in no way provide less than the same measure of protection afforded the miners under the existing standard.

Dated: November 13, 2008.

Patricia W. Silvey,

Director, Office of Standards, Regulations, and Variances.

[FR Doc. E8–27406 Filed 11–18–08; 8:45 am]

BILLING CODE 4510–43–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards

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 December 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).

Thursday, December 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.: Chapters 7 and 14 of the SER Associated with the Economic Simplified Boiling Water Reactor (ESBWR) Design Certification Application (Open/Closed)—The Committee will hear a briefing by and hold discussions with representatives of the NRC staff and General Electric—Hitachi Nuclear Energy (GEH) regarding Chapters 7 and 14, of the NRC staff's SER With Open Items associated with the ESBWR design certification application.

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

10:15 a.m.–12 p.m.: Early Site Permit Application and the Final SER for the Vogtle Nuclear Plant (Open)—The Committee will hear a briefing by and hold discussions with representatives of the NRC staff and Southern Nuclear Operating Company (SNC) regarding the Early site permit application and the NRC staff's final SER for the Vogtle Nuclear Plant.

1 p.m.–2:30 p.m.: Status of Staff Activities Associated with Potential Revision to 10 CFR 50.46(b) (Open)—The Committee will hear a briefing by the Subcommittee Chairman and hold discussions with representatives of the NRC staff regarding the status of staff activities associated with potential revision to 10 CFR 50.46(b).

2:45 p.m.–4:15 p.m.: NRC Staff's Initial White Paper on Containment Overpressure Credit Issue (Open)—The Committee will hear reports by the Subcommittee Chairman and hold discussions with representatives of the NRC staff regarding the initial White Paper on the use of containment accident pressure in determining the available net positive suction head of emergency core cooling and containment heat removal pumps, and related matters.

4:15 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters discussed during this meeting.

Friday, December 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.–10 a.m.: Overview of the Human Reliability Analysis (HRA) Research Activities (Open)—The Committee will hear a briefing by and hold discussions with representatives of the NRC staff regarding HRA research activities.

10:15 a.m.–12 p.m.: Draft Policy Statement on Defense-in-Depth for Future Nuclear Reactors (Open)—The Committee will hear a briefing by and hold discussions with representatives of the NRC staff regarding draft Policy Statement on Defense-in-Depth for Future Nuclear Reactors.

1:30 p.m.–2:30 p.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed)—The Committee will discuss of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the Full Committee during future ACRS meetings and other matters related to the conduct of the ACRS business.

[Note: A portion of this meeting may be closed pursuant to 5 U.S.C. 552b(c)(2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which would

constitute a clearly unwarranted invasion of personal privacy]

2:30 p.m.–2:45 p.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.

2:45 p.m.–3 p.m.: Election of ACRS Officers for CY 2009 (Open)—The Committee will discuss the election of the Chairman and Vice-Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee for FY 2009.

3:15 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Saturday, December 6, 2008, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

8:30 a.m.–12:30 p.m.: Preparation of ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

12:30 p.m.–1 p.m.: Miscellaneous (Open)—The Committee will discuss 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.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 6, 2008, (73 FR 58268–58269). 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.

In accordance with Subsection 10(d) Public Law 92–463, I have determined that it may be necessary to close portions of this meeting noted above 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 pursuant to 5 U.S.C. 552b(c)(2) and (6). In addition, it may be necessary to close a portion of the meeting to protect information designated as proprietary by General Electric—Hitachi or its contractors pursuant to 5 U.S.C. 552b(c)(4).

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 Girija Shukla, Cognizant ACRS staff (301–415–6855), between 7:15 a.m. and 5 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: November 13, 2008.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E8–27469 Filed 11–18–08; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Proposed Extension of Existing Collection; Comment Request

Upon Written Request, Copies Available From: U.S. Securities and Exchange Commission, Office of Investor Education and Advocacy, Washington, DC 20549–0213.

Extension:

Rule 9b–1; OMB Control No. 3235–0480; SEC File No. 270–429.

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*) the Securities and Exchange Commission (''Commission'') is soliciting comments on the existing collection of information provided for the following rule: Rule 9b–1 Options Disclosure Document (17 CFR 240.9b–1) under the Securities Exchange Act of 1934 (15 U.S.C. 78 *et seq.*). The Commission plans to submit this existing collection of information to the Office of Management and Budget (''OMB'') for extension and approval.

Rule 9b–1 (17 CFR 240.9b–1) sets forth the categories of information required to be disclosed in an options disclosure document (''ODD'') and requires the options markets to file an ODD with the Commission 60 days prior to the date it is distributed to investors. In addition, Rule 9b–1 provides that the ODD must be amended if the information in the document becomes materially inaccurate or incomplete and that amendments must be filed with the Commission 30 days prior to the distribution to customers. Finally, Rule 9b–1 requires a broker-dealer to furnish to each customer an ODD and any amendments, prior to accepting an order to purchase or sell an option on behalf of that customer.

There are six options markets that must comply with Rule 9b–1. These six respondents work together to prepare a single ODD covering options traded on each market, as well as amendments to the ODD. These respondents file approximately three amendments per year. The staff calculates that the preparation and filing of amendments should take no more than eight hours per options market. Thus, the total compliance burden for options markets per year is 144 hours (6 options markets × 8 hours per amendment × 3 amendments). The estimated cost for an in-house attorney is $295 per hour,[1]

[1] The $295/hour figure for an attorney is from SIFMA's *Management & Professional Earnings in the Securities Industry 2007,* modified by the Commission staff to account for an 1800-hour work-year and multiplied by 5.35 to account for bonuses, firm size, employee benefits and overhead.

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

November 13, 2008

AGENDA
558th ACRS MEETING
DECEMBER 4-6, 2008

THURSDAY, DECEMBER 4, 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 – 10:00 A.M. Chapters 7 and 14 of the SER Associated with the Economic Simplified Boiling Water Reactor (ESBWR) Design Certification Application (Open/Closed) (MLC/HJV)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and General Electric - Hitachi Nuclear Energy (GEH) regarding Chapters 7 and 14 of the NRC staff's Safety Evaluation Report (SER) With Open Items associated with the ESBWR design certification application.

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

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

10:00 – 10:15 A.M. ***** BREAK *****

3) 10:15 – 12:00 P.M. Early Site Permit Application and the Final SER for the Vogtle Nuclear Plant (Open) (DAP/DAW)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff and Southern Nuclear Operating Company (SNC) regarding the Early site permit application and the NRC staff's final SER for the Vogtle Nuclear Plant.

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

12:00 – 1:00 P.M. ***** LUNCH *****

4) 1:00 – 2:30 P.M. Status of Staff Activities Associated with Potential Revision to 10 CFR 50.46 (b) (Open) (JSA/CLB)

4.1) Remarks by the Subcommittee Chairman

4.2) Briefing by and discussions with representatives of the NRC staff regarding the status of staff activities associated with potential revision to 10 CFR 50.46 (b).

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

2:30 – 2:45 P.M. * BREAK**

5) 2:45 – 4:15 P.M. NRC Staff's Initial White Paper on Containment Overpressure Credit Issue (Open) (MVB/ZA)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding the initial White Paper on the use of Containment Accident Pressure in Determining the Available Net Positive Suction Head of Emergency Core Cooling and Containment Heat Removal Pumps, and related matters.

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

6) 4:15 – 7:00 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

6.1) Chapters 7 and 14 of the SER Associated with the ESBWR Design Certification Application (MLC/HJV)

6.2) Early Site Permit Application and Final SER for the Vogtle Nuclear Plant (DAP/DAW).

FRIDAY, DECEMBER 5, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

7) 8:30 – 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (WJS/AFD/SD)

8) 8:35 – 10:00 A.M. Overview of the Human Reliability Analysis (HRA) Research Activities (Open) (GEA/HPN)

Briefing by and discussions with representatives of the NRC staff regarding HRA research activities.

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

10:00 – 10:15 A.M. * BREAK ***

9) 10:15 – 12:00 P.M. Draft Policy Statement on Defense-in-Depth for Future Nuclear Reactors (Open) (WJS/MB)
Briefing by and discussions with representatives of the NRC staff regarding draft Policy Statement on Defense-in-Depth for Future Nuclear Reactors.

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

12:00 – 1:30 P.M. ***** LUNCH *****

10) 1:30 – 2:30 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed) (WJS/EMH)

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

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

[NOTE: A portion of this meeting may be closed pursuant to 5 U.S.C. 552b (c)(2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy]

11) 2:30 – 2:45 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.

12) 2:45 – 3:00 P.M. Election of ACRS Officers for CY 2009 (Open) (EMH/SD)
Election of the Chairman and Vice-Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee for CY 2009.

3:00 – 3:15 P.M. ***** BREAK *****

13) 3:15 – 7:00 P.M. Preparation of ACRS Reports (Open)
Continue discussion of the proposed ACRS reports listed under Item 6.

SATURDAY, DECEMBER 6, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14)	8:30 – 12:30 P.M.	Preparation of ACRS Reports (Open)
(10:30-10:45 A.M. BREAK)		Continue discussion of the proposed ACRS reports listed under Item 6.
15)	12:30 – 1:00 P.M.	Miscellaneous (Open) (WJS/EMH) 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.

Appendix III

LIST OF ATTENDEES
558th ACRS MEETING
DECEMBER 4-6, 2008

The list of attendees was not available at the time the minutes were finalized.

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

January 15, 2009

AGENDA
559th ACRS MEETING
FEBRUARY 5-7, 2009

THURSDAY, FEBRUARY 5, 2009, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 – 10:30 A.M. Draft Final NUREG-1855, Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking (Open) (GEA/HJV)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff regarding draft final NUREG-1855 and related matters.

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

   10:30 – 10:45 A.M. ***** BREAK *****

3) 10:45 – 11:45 A.M. Draft Final Regulatory Guide DG-5021, Safety/Security Interface (Open) (MVB/MB)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding draft final Regulatory Guide DG-5021 on Safety/Security Interface.

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

   11:45 – 12:45 P.M. ***** LUNCH *****

4) 12:45 – 2:45 P.M. Digital Upgrade of the Oconee Reactor Protection System and Engineered Safety Features (Open/Closed) (CB/CEA)
   4.1) Remarks by the Subcommittee Chairman
   4.2) Briefing by and discussions with representatives of the NRC staff and Duke Energy regarding digital upgrade of the reactor protection system and engineered safety features at Oconee Nuclear Station, Units 1, 2, & 3, and related matters.

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

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

2:45 –3:00 P.M. ***** BREAK**

5) 3:00 – 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
5.1) Draft Final NUREG-1855, Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking (GEA/HJV)
5.2) Draft Final Regulatory Guide DG-5021, Safety/Security Interface (MVB/MB)
5.3) Containment Overpressure Credit Issue (WJS/MVB/ZA)

FRIDAY, FEBRUARY 6, 2009, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

6) 8:30 – 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (MVB/EMH/SD)

7) 8:35 – 10:00 A.M. SECY-08-0197, Options to Revise Radiation Protection Regulations and Guidance Based on Recommendations of the International Commission on Radiological Protection (ICRP) (Open) (MTR/NMC)
7.1) Remarks by the Subcommittee Chairman
7.2) Briefing by and discussions with representatives of the NRC staff regarding options to revise NRC radiation protection regulations and guidance based on the recommendations of the ICRP.

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

10:00 – 10:15 A.M. ***** BREAK *****

8) 10:15 – 10:45 A.M. Subcommittee Reports (Open)
8.1) Report by and discussions with the Chairman of the Plant License Renewal Subcommittee regarding Interim Review of the Beaver Valley License Renewal Application and the Safety Evaluation Report (SER) with Open Items that were discussed during the Subcommittee meeting on February 4, 2009 (DCB/CLB)

8.2) Report by and discussions with the Chairman of the Plant License Renewal Subcommittee regarding Interim Review of the National Institute of Standards and Technology (NIST) License Renewal Application and the SER with Open Items that were discussed during the Subcommittee meeting on February 4, 2009 (JDS/PW)

9) 10:45 – 11:45 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed) (MVB/EMH)

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.

[NOTE: A portion of this session may be closed pursuant to 5 U.S.C. 552b (c)(2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy]

10) 11:45 – 12:00 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (MVB/CS/AFD)

Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

12:00 – 1:00 P.M. * LUNCH ***

11) 1:00 – 7:00 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

11.1) Draft Final NUREG-1855, Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking (GEA/HJV)

11.2) Draft Final Regulatory Guide DG-5021, Safety/Security Interface (MVB/MB)

11.3) Containment Overpressure Credit Issue (WJS/MVB/ZA)

11.4) SECY-08-0197, Options to revise Radiation Protection Regulations and Guidance based on Recommendations of the International Commission on Radiological Protection (ICRP) (MTR/NMC)

SATURDAY, FEBRUARY 7, 2009, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

12) 8:30 – 12:30 P.M. Preparation of ACRS Reports (Open)
(10:30-10:45 A.M. BREAK) Continue discussion of the proposed ACRS reports listed under Item 11.

13) 12:30 – 1:00 P.M. Miscellaneous (Open) (MVB/EMH)
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.

APPENDIX V

LIST OF HANDOUTS
558th ACRS MEETING
DECEMBER 4-6, 2008

I. Opening Remarks by the ACRS Chairman
1. Opening Remarks
2. Items of Interest

II. Chapters 7 and 14 of the SER Associated with the Economic Simplified Boiling Water Reactor (ESBWR) Design Certification Application
3. Proposed Schedule
4. Status Report

III. Early Site Permit Application and the Final SER for the Vogtle Nuclear Plant
5. Proposed Schedule
6. Status Report

IV. 10 CFR 50.46(b) Rulemaking Activities
7. Agenda
8. Status Report
9. Letter to Dale E. Klein, Chairman U.S. Nuclear from Mario V. Bonaca, Acting Chairman, "Proposed Technical Basis for the Revision to 10 CFR 50.46 LOCA Embrittlement Criteria For Fuel Cladding Materials," Dated May 23, 2007
10. Public Comments on Technical Basis

V. Briefing on NRC Staff's White Paper on Containment Overpressure Credit
11. Table of Contents
12. Proposed Schedule
13. Status Report
14. The Use of Containment Accident Pressure In Determining the Available Net Positive Suction Head Of Emergency Core Cooling and Containment Heat Removal Pumps

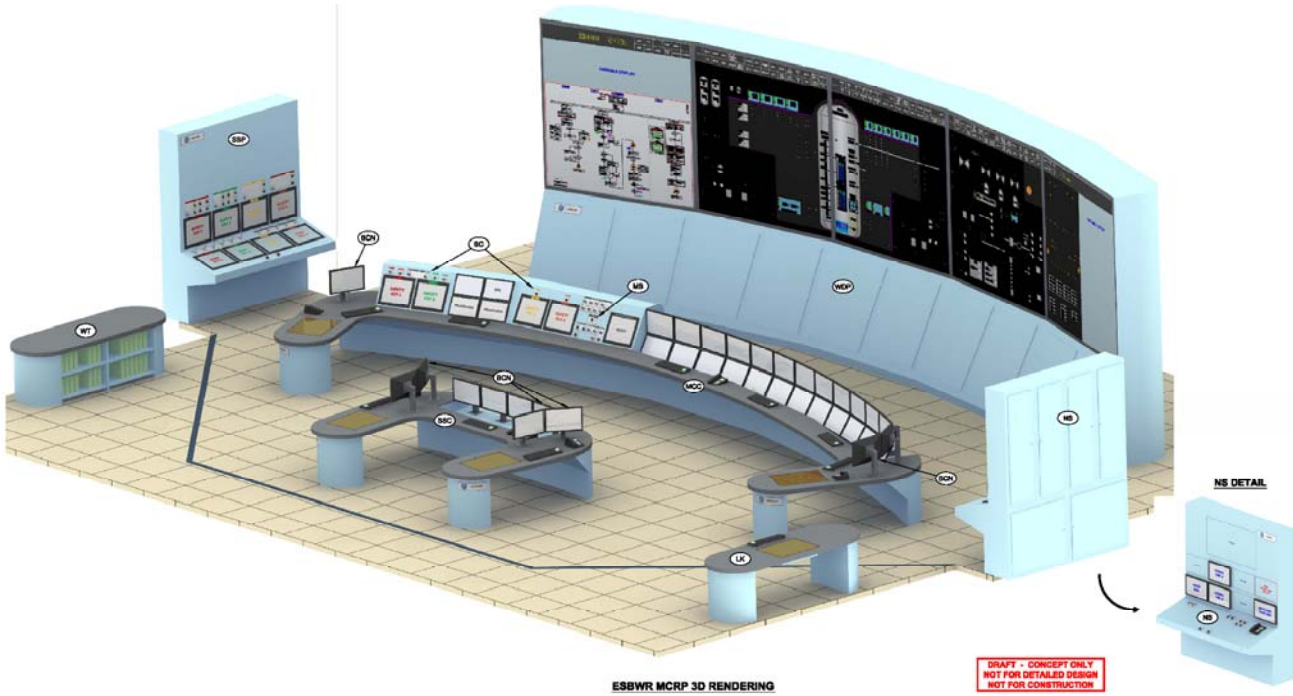
VIII. Overview of the Human Reliability Analysis (HRA) Research Activities
15. Table of Contents
16. Proposed Schedule
17. Status Report
18. Memorandum dated November 8, 2006, from Annette L. Vietti-Cook, Secretary, NRC to John T. Larkins, Executive Director, ACRS, Subject: Staff Requirements – Meeting With Advisory Committee on Reactor Safeguards, 2:30 P.M., Friday, October 20, 2006, Commissioners' Conference Room, One White Flint North, Rockville, Maryland (Open to Public Attendance)
19. Report dated April 23, 2007, from William J. Shack, Chairman, ACRS, to Dale Klein, Chairman, NRC, Subject: Human Reliability Analysis Models

IX. Draft Policy Statement on Defense-in-Depth for Future Nuclear Reactors

20. Table of Contents
21. Proposed Schedule
22. Status Report
23. Letter from D. A. Powers, Chairman, ACRS, to S. A. Jackson, Chairman, NRC, dated May 19, 1999, “The Role of Defense in Depth in a Risk-Informed Regulatory System
24. Letter dated September 26, 2007, from W. J. Shack, Chairman, ACRS, to D. E. Klein, Chairman, NRC, on Draft NUREG-1860, “Framework for Development of a Risk-Informed, Performance-Based Alternative to 10 CFR Part 50”
25. Letter dated July 24, 2007, from G.B. Wallis, ACRS, to L.A. Reyes, Executive Director for Operations, USNRC, Subject: Comments on Draft NUREG-1860, “Framework for Development of a Risk-Informed, Performance-Based Alternative to 10 CFR Part 50”

ESBWR DCD Chapter 7 DCIS Overview

Full/Subcommittee ACRS Meeting

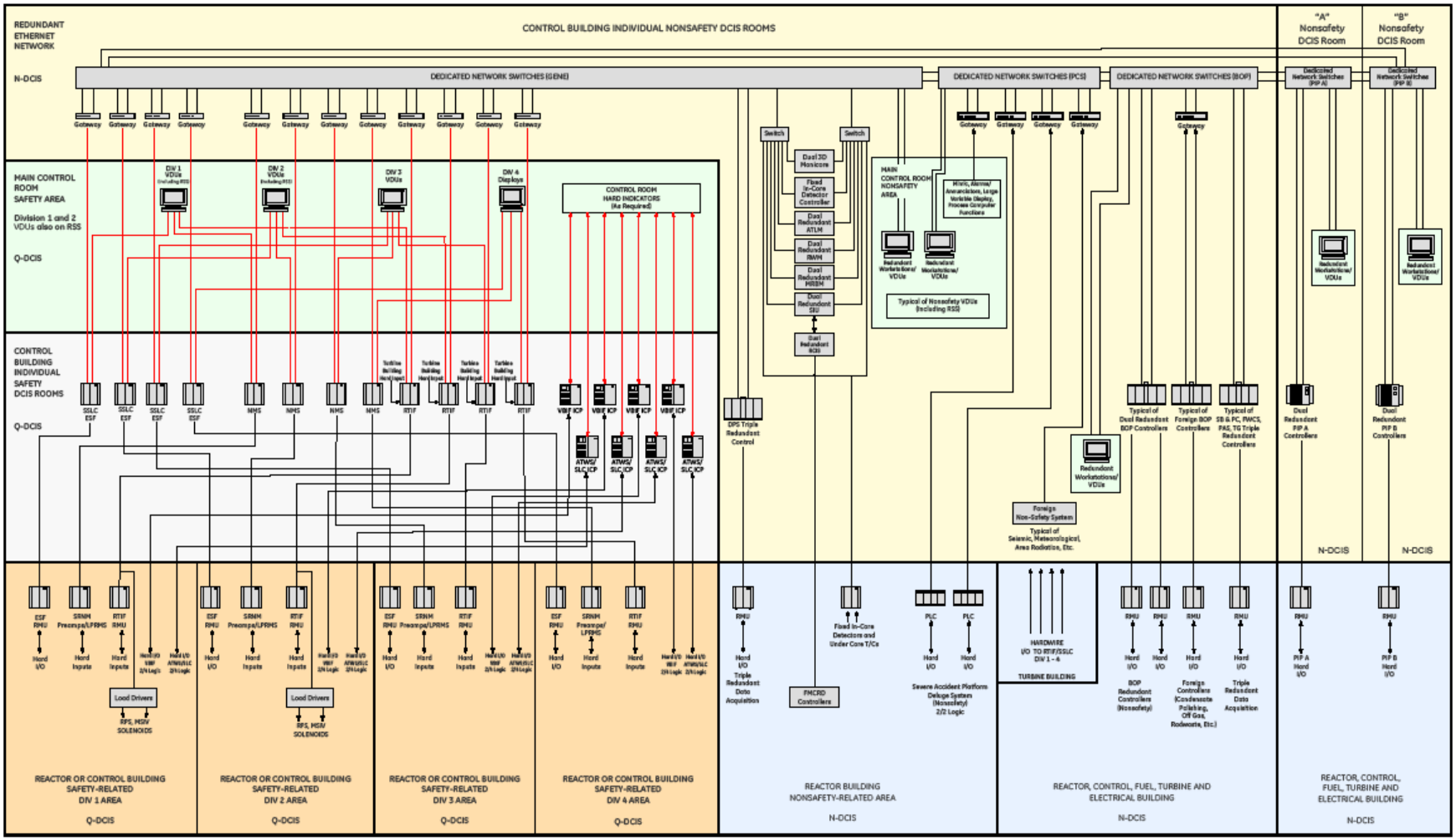


Rich Miller
Ira Poppel
Steve Kimura
Dec 3-4, 2008

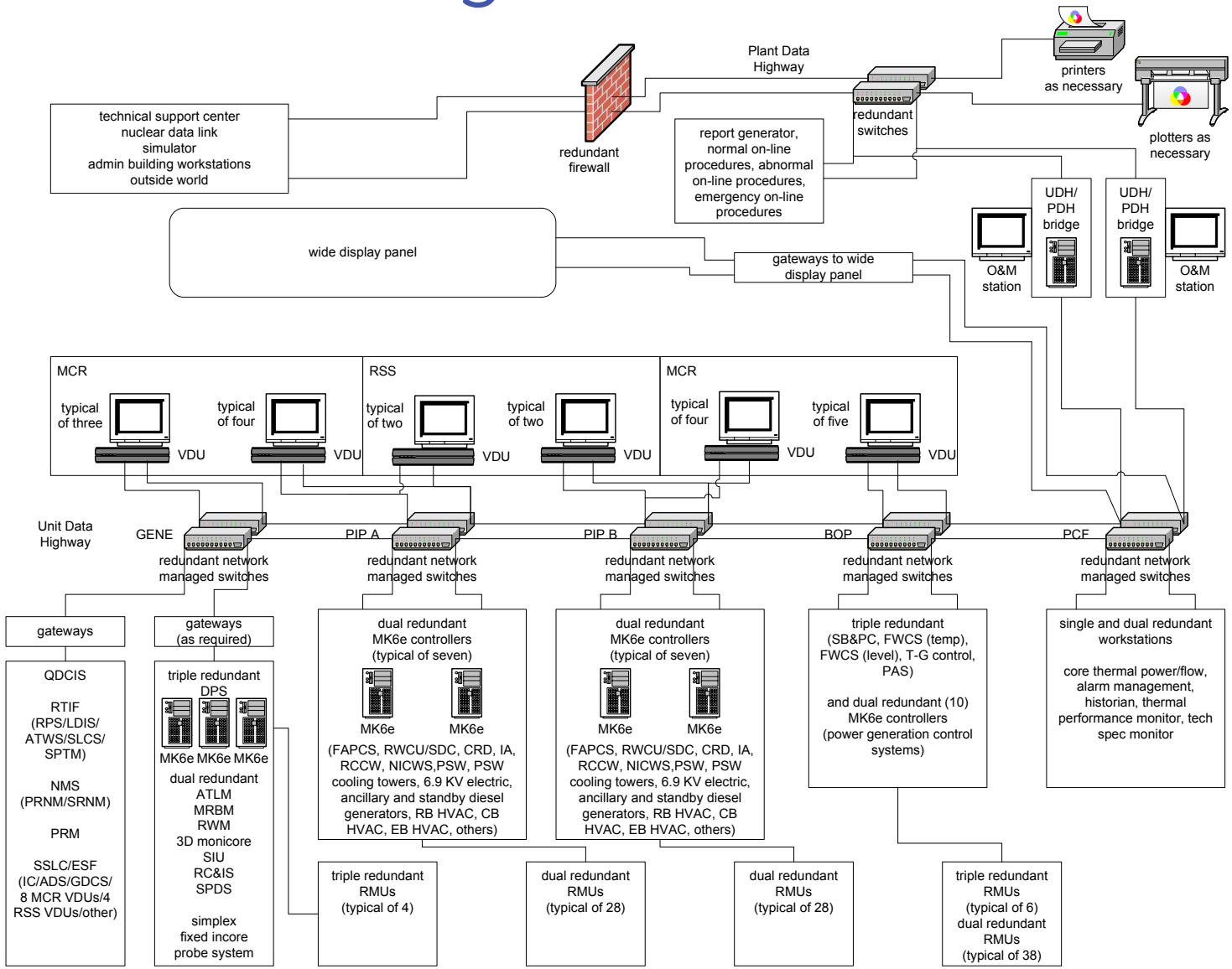
HITACHI

ESBWR DCIS

ESBWR Distributed Control and Information System (DCIS) Functional Network Diagram



ESBWR DCIS Organization - Continued



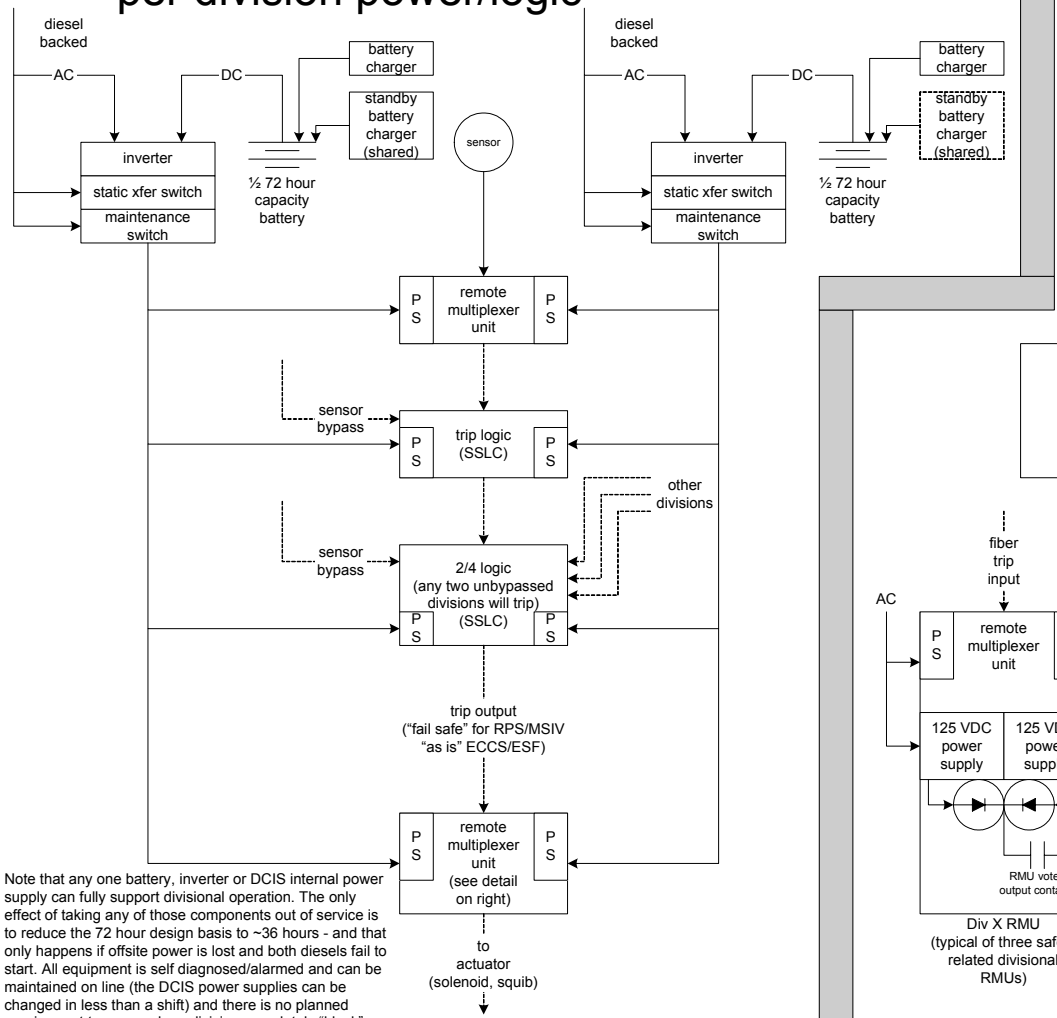
HITACHI

Q-DCIS

- Q-DCIS organized into
  - > RTIF/NMS (reactor trip system)
  - > SSLC/ESF (ECCS and information systems)
  - > ATWS/SLC and VBIF
- Q-DCIS is deterministic
- Q-DCIS has four divisions
- Q-DCIS is N-2
- RTIF/NMS and SSLC/ESF functions implemented on diverse hardware/software platforms
- Q-DCIS is physically, electrically and data isolated between divisions and between Q-DCIS and N-DCIS

Q-DCIS Power

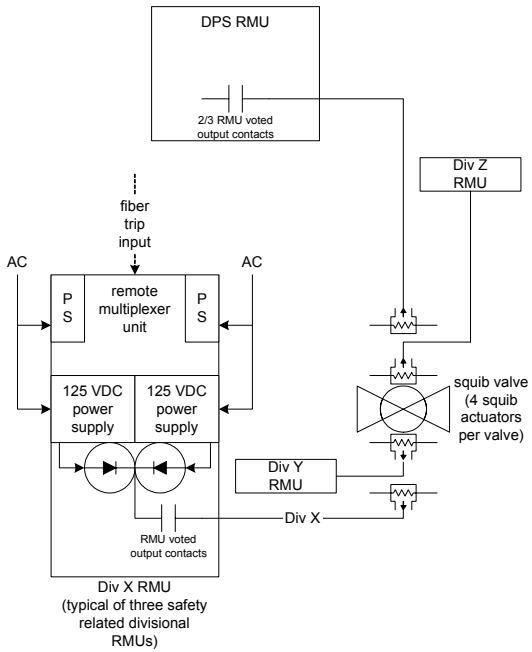
per division power/logic



Note that any one battery, inverter or DCIS internal power supply can fully support divisional operation. The only effect of taking any of those components out of service is to reduce the 72 hour design basis to ~36 hours - and that only happens if offsite power is lost and both diesels fail to start. All equipment is self diagnosed/alarmed and can be maintained on line (the DCIS power supplies can be changed in less than a shift) and there is no planned requirement to ever make a division completely "black".

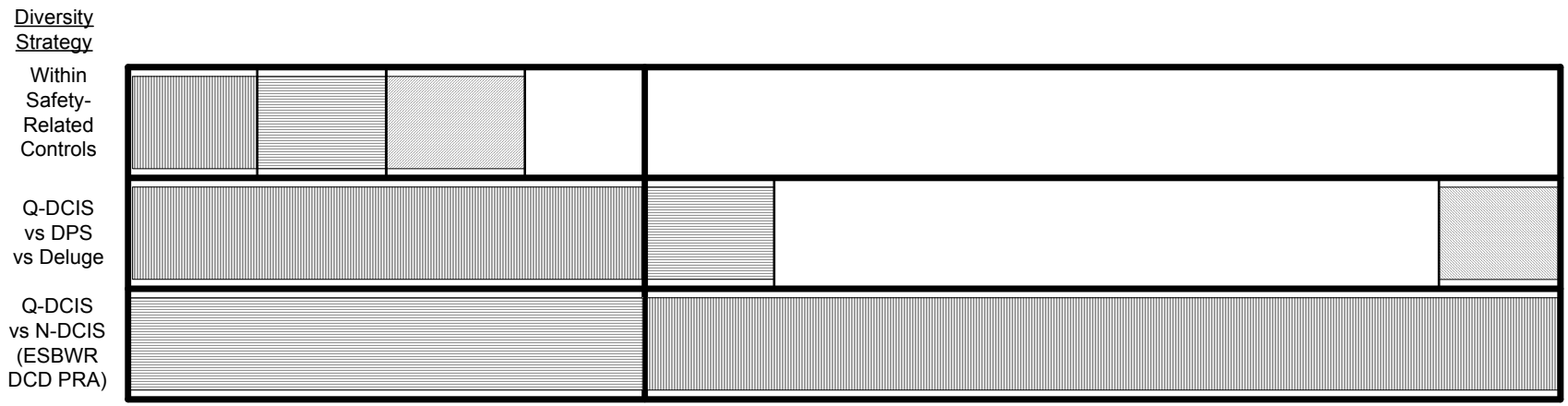
per actuator

Typical of SRV solenoids, DPV valves, GDCS valves, equalizing valves; any one (of three) squibs (divisions) or DPS will open the valve.
Lines with two isolation valves in series use two solenoids per valve; lines with one isolation valve in series with a check valve use four solenoids.
Per division, any one of two AC inputs (inverters/batteries), allows the squib to fire.
Overall scheme allows any division to be taken out of service, incur a DBA and accept a single additional failure and still open all valves.



ESBWR DCIS Overall Diversity

Safety Category	Safety-Related				Nonsafety-Related						
	Q-DCIS				N-DCIS						
Platform/ Network Segment	RTIF NMS	SSLC/ ESF	Independent Control platform	other	GENE		PIP A/B	BOP		PCF	
architecture	divisional	divisional	divisional	note 1	Triple Redundant (DPS)	Dual Redundant	Dual Redundant	Triple Redundant	Dual Redundant	Workstations	PLC (Deluge)



Note 1 – RSS provides operator workstations at appropriate diverse locations outside the main control room in accordance with GDC 19. See DCD section 7.1.3.2.3.2

Note 2 – Crosshatching denotes different platforms or networks

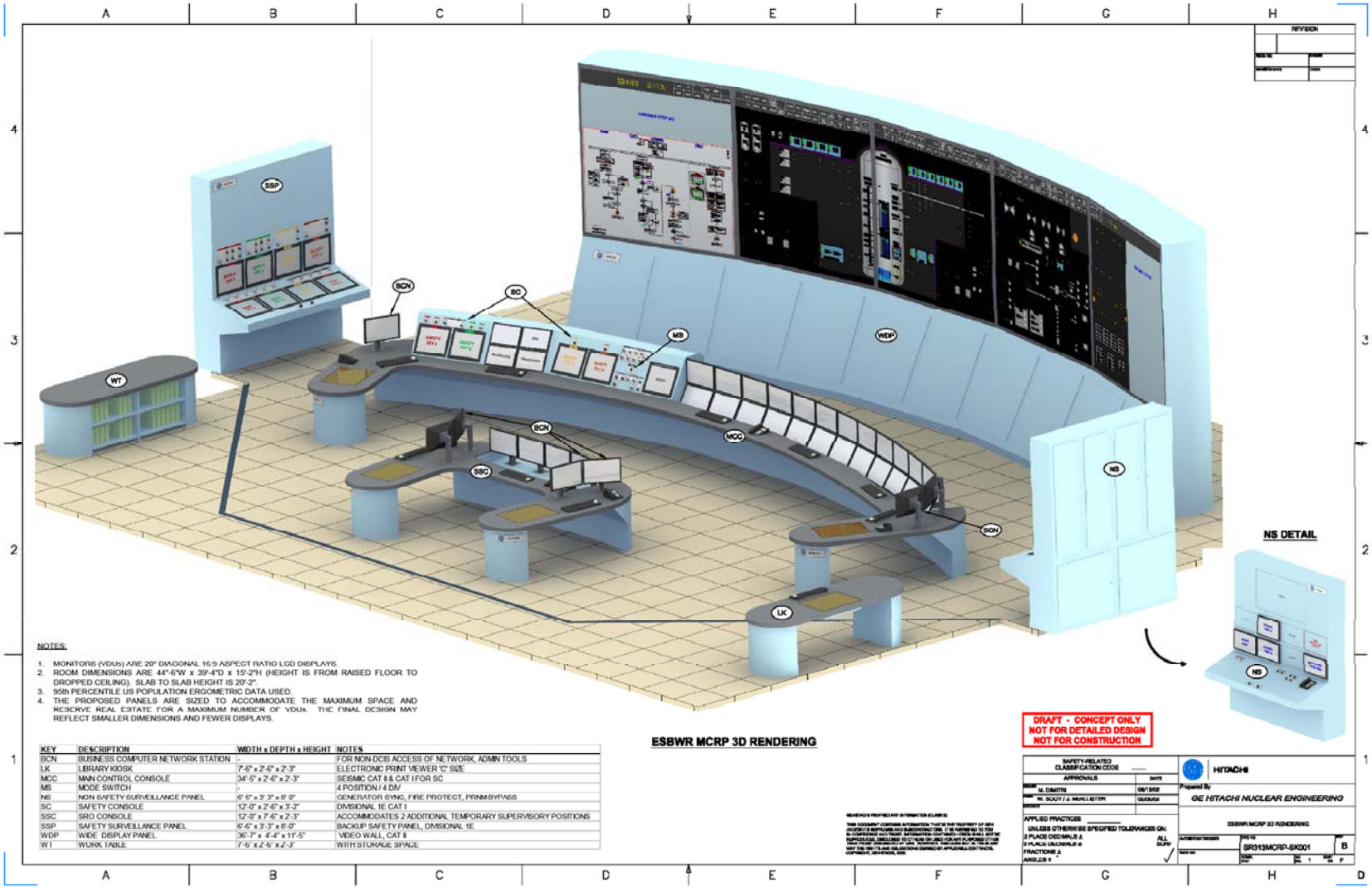
N-DCIS

- N-DCIS is organized into five independent dual redundant network segments
  - > GENE (contains DPS)
  - > PIP A (investment protection/RTNSS)
  - > PIP B (investment protection/RTNSS)
  - > BOP (power generation)
  - > Plant Computer Functions
- A/B N-DCIS components located in separate rooms/fire zones
- N-DCIS components dual or triply redundant powered by two or three uninterruptible power systems
- Important reactor control systems segmented
- Networks are not used for closed loop control
- N-DCIS components diverse from Q-DCIS components

ESBWR Diverse Protection System

- Provides manual and automatic
  - Backup scram functions
    - (Rx level, Rx pressure, pool temperature, drywell pressure)
  - Backup MSIV isolation functions
    - (Rx steam flow, Rx level)
  - backup ADS and GDCS initiation
  - Backup IC initiation
  - Backup process isolation functions
  - SLCS initiation
- Mitigates loss of feedwater heating (SRI, SCRRI)
- Initiates ARI, SRI/SCRRI, all control rod run-in
- Initiates FW runback
- Initiates level 9 FW pump trip

ESBWR Main Control Room



NOTES:

1. MONITORS (VDUs) ARE 20" DIAGONAL 16:9 ASPECT RATIO LCD DISPLAYS.
2. ROOM DIMENSIONS ARE 44'-6"W x 39'-4"D x 15'-2"H (HEIGHT IS FROM RAISED FLOOR TO DROPPED CEILING). SLAB TO SLAB HEIGHT IS 20'-2".
3. 95th PERCENTILE US POPULATION ERGOMETRIC DATA USED.
4. THE PROPOSED PANELS ARE SIZED TO ACCOMMODATE THE MAXIMUM SPACE AND RESERVE REAL ESTATE FOR A MAXIMUM NUMBER OF VDUs. THE FINAL DESIGN MAY REFLECT SMALLER DIMENSIONS AND FEWER DISPLAYS.

KEY	DESCRIPTION	WIDTH x DEPTH x HEIGHT	NOTES
BCN	BUSINESS COMPUTER NETWORK STATION	-	FOR NON-DCIS ACCESS OF NETWORK, ADMIN TOOLS
LK	LIBRARY KIOSK	7'-6" x 2'-6" x 2'-3"	ELECTRONIC PRINT VIEWER 'C' SIZE
MCC	MAIN CONTROL CONSOLE	34'-5" x 2'-6" x 2'-3"	SEISMIC CAT II & CAT I FOR SC
MS	MODE SWITCH	-	4 POSITION / 4 DIV
NS	NON SAFETY SURVEILLANCE PANEL	6' 6" x 3' 3" x 8' 0"	GENERATOR SYNC, FIRE PROTECT, PRNM BYPASS
SC	SAFETY CONSOLE	12'-0" x 2'-6" x 3'-2"	DIVISIONAL 1E CAT I
SSC	SRO CONSOLE	12'-0" x 7'-6" x 2'-3"	ACCOMMODATES 2 ADDITIONAL TEMPORARY SUPERVISORY POSITIONS
SSP	SAFETY SURVEILLANCE PANEL	6'-6" x 3'-3" x 8'-0"	BACKUP SAFETY PANEL, DIVISIONAL 1E
WDP	WIDE DISPLAY PANEL	36'-7" x 4'-4" x 11'-5"	VIDEO WALL, CAT II
WT	WORK TABLE	7'-5" x 4'-5" x 2'-3"	WITH STORAGE SPACE

ESBWR Remote Shutdown System (RSS)

- ESBWR RSS not really a “system” – instead two auxiliary control rooms with RSS panels located in Div 1 and Div 2 quadrants of the Reactor Building
- GDC 19 RSS requirements are met by the manual scram and isolation switches on the panels
- With offsite power available, either RSS panel can operate BOP normally for plant shutdown
- With only diesel power available, either RSS panel can operate PIP A or PIP B systems for plant shutdown
- With only safety-related batteries available, either RSS panel can operate division 1 or division 2 systems for plant shutdown

Presentation to the 558th ACRS Meeting

Summary of Staff Review
of
ESBWR Design Certification Document Chapter 14 and Tier 1

Presented by Eric Oesterle
Lead Project Manager (NRO/DNRL)
December 4, 2008

Purpose

- Provide an update of the status of the staff's review of ESBWR DCD Tier 2, Chapter 14, Initial Test Program and ITAAC, and Tier 1, since the 557th ACRS Full Committee meeting

Staff Review of ESBWR Chapter 14 and Tier 1 Overview of Design Certification

Regulations:

- 10 CFR 50.34(b)(6)(iii) and 10 CFR 52.79(a)(28) - Initial Test Program
- 10 CFR 53.27(b)(1) - ITAAC

Regulatory Guidance:

- Standard Review Plan 14.2, Initial Plant Test Program
- Standard Review Plan 14.3, Inspections, Tests, Analyses and Acceptance Criteria (ITAAC)
- Reg. Guide 1.68, Initial Test Programs for Water-Cooled Nuclear Power Plants
- Reg. Guide 1.20, Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing
- Reg. Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)
- Reg. Guide 1.206, Combined License (COL) Applications for Nuclear Power Plants

Staff Review of ESBWR Chapter 14 and Tier 1 Overview of Design Certification

Summary of Staff Review of ESBWR Chapter 14 and Tier 1:

- RAIs issued: 539
- RAI responses submitted: 509
- RAIs resolved: 476

Summary of Staff Review of ESBWR Section 14.2, Initial Test Program:

- RAIs issued: 99 (1 new RAI since 557th ACRS mtg)
- RAIs resolved: 93
- Unresolved RAIs associated with:
  - expansion, vibration and dynamic effects testing
  - testing of digital instrumentation and control system functions
  - safety system logic and control pre-operational testing
  - lead detection and isolation system pre-operational testing
  - reactor internals vibration testing
  - AC power distribution system pre-operational testing
  - incomplete description of pre-operational testing for DCIS

Staff Review of ESBWR Chapter 14 and Tier 1 Overview of Design Certification

Summary of Staff Review of ESBWR Tier 1 and Section 14.3, ITAAC:

- RAIs issued: 440 (3 new RAIs since 557th ACRS mtg)
- RAIs resolved: 383 (21 RAIs resolved since 557th ACRS mtg)

- Unresolved RAIs associated with:
  - tables of key aspects, analyses, and design features included in ITAAC
  - interface materials (offsite power and plant service water system)
  - digital instrumentation and control systems
  - human factors engineering
  - electrical systems
  - containment systems
  - reactor systems
  - format and consistency issues across similar ITAAC
  - security design features

Staff Review of ESBWR Chapter 14 and Tier 1 Overview of Design Certification

Summary

- NRO staff continues to engage with GEH to obtain satisfactory resolutions of open items associated with review of the Initial Test Program and ITAAC that are necessary to develop the staff's Final Safety Evaluation Report (FSER) for Tier 1 and Chapter 14 of the ESBWR Design Certification Document

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Presentation to the ACRS Full Committee

ESBWR Design Certification Review

Chapter 7,“Instrumentation and Controls”

December 4, 2008

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 7

Purpose

- Brief the Subcommittee on the staff's continuing review of the ESBWR DCD Application Sections
  - 7.1 "Introduction"
    - Software Development Activities
    - Diversity and Defense-in-Depth Assessment
    - Setpoint Methodology
    - Data Communication Systems
  - 7.2 "Reactor Trip Systems"
  - 7.3 "Engineered Safety Features Systems"
  - 7.4 "Safe Shutdown Systems"
  - 7.5 "Information Systems Important to Safety"
  - 7.6 "Interlock Systems"
  - 7.7 "Control Systems"
  - 7.8 "Diverse Instrumentation and Control Systems"
- Answer the Committee's questions

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 7 Review Team

- Project Manager
  - Dennis Galvin

- Technical Reviewers
  - Hulbert Li, Lead
  - Leroy Hardin
  - Sang Rhow
  - Royce Beacom
  - Dinesh Taneja
  - Joseph Ashcraft
  - Kimberley Corp
  - Eugene Eagle
  - Thomas Fredette
  - Jack Zhao

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 7 Presentation

<u>Outline of Presentation</u>

- Applicable Regulations
- RAI Status Summary
- SER Technical Topics of Interest
  - Key I&C DAC/ITAAC Items
  - Key SER Open Items
- Discussion / Committee Questions

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 7

Key Regulations

- 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h)(3), 10 CFR 50.34(f)(2), 10 CFR 50.62, and 10 CFR 52.47(b)(1)
- 10 CFR Part 50, Appendix A, GDC 1, 2, 4, 10, 13, 15, 16, 19, 20, 21, 22, 23, 24, 25, 28, 29, 33, 34, and 35

Principal Review Guidance

- SRP Section 7, including Branch Technical Positions
- SRP Sections 14.3 and 14.3.5
- Regulatory Guides 1.22, 1.47, 1.53, 1.62, 1.75, 1.97, 1.105, 1.118, 1.151, 1.152, 1.168, 1.169, 1.170, 1.171, 1.172, 1.173, 1.180, 1.189, 1.204, and 1.209
- SRM on SECY-93-087 and SECY-92-053

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 7

RAI Status Summary: SRP Chapter 7

- Original number of RAIs = 276
- Number of RAIs resolved = 206
- Number of Remaining Open Items = 70

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 7 Summary

The staff followed SRP Chapters 7 & 14 Guidance to review high level functional requirements and design commitments for:

- IEEE-603 criteria compliance
- Life-cycle design process
- Setpoint methodology
- Diversity & Defense-in-Depth
- Data Communication

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 7 Summary

RAI open items status

- Most of the remaining open items are clarification/consistency related issues
- No safety significant technical issues that need resolution

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Protecting People and the Environment

Presentation to the ACRS Full Committee

Safety Review of the Vogtle Electric Generating Plant Early Site Permit Application and Limited Work Authorization Request

December 4, 2008

Purpose

- To provide the ACRS an overview of the staff's safety review and conclusions on:
  - The Vogtle Electric Generating Plant (VEGP) Early Site Permit (ESP) Application
  - The VEGP Limited Work Authorization (LWA) Request
- Address the Full Committee's questions

Meeting Agenda

Early Site Permit Application Review:

- Remaining Schedule Milestones
- Key Review Areas / Resolution of Open Items
- Advanced Safety Evaluation Report (SER) Conclusions

Limited Work Authorization Review:

- VEGP LWA Request Summary
- Review of LWA Activities
- LWA Conclusion
- Discussion / Questions

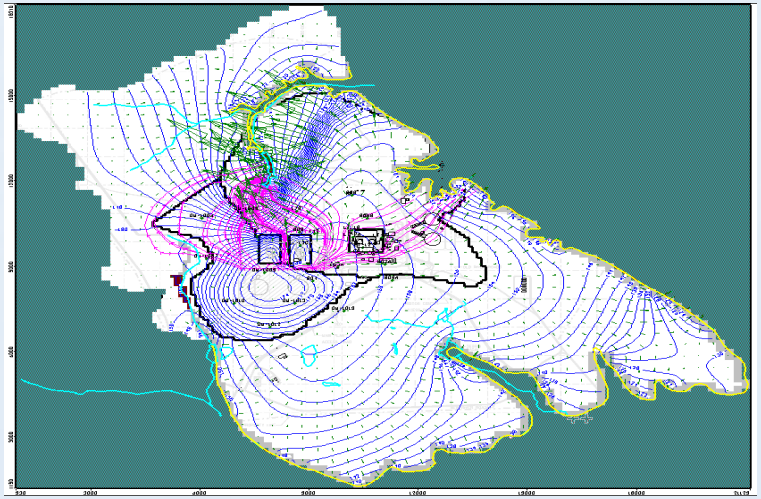
Remaining Milestones

- ACRS Final Letter Assumed – 1/2009
- Final SER Issuance – 2/5/2009
- Mandatory Hearing – 3/23/2009
- Commission Decision Assumed – Summer/Fall 2009

Key Review Areas for ESP/LWA

- The staff completed its review of the following areas for the ESP:
  - 2.1 - Geography and Demography
  - 2.2 - Nearby Industrial, Transportation, and Military Facilities
  - **2.3 - Meteorology (1)**
  - **2.4 - Hydrology (4)**
  - **2.5 - Geology, Seismology, Geotechnical Engineering (22)**
  - 3.5.1.6 - Aircraft Hazards
  - 11 - Doses from Routine Liquid and Gaseous Effluent Releases
  - **13.3 - Emergency Planning (13)**
  - 13.6 - Physical Security
  - 15 - Accident Analyses
  - 17 - Quality Assurance
- Resolution of all Open Items **(Bold)** discussed in the Advanced SER
- The staff completed its review of the following areas for the LWA:
  - 2.5.4 – Stability of Subsurface Materials and Foundations
  - 3.8.5 – Foundations
  - 13.7 – Fitness For Duty Program
  - 17 – Quality Assurance Program

Section 2.4: Hydrology



Section 2.4 Hydrologic Hazard Analyses

- Floods induced by rain, dam break, hurricane, and tsunami.
- Low water impacts
- Ice impacts
- Water use impacts
- Groundwater flow and contamination transport analyses

2.4 Hydrology

- Section 2.4.8: Cooling Water Canals and Reservoirs (**OI 2.4-1**)
  - Issue: Do canals or reservoirs are used as any external water source for safety-related cooling water?
  - Resolution: Staff confirmed that safety-related cooling water is provided not from canals and reservoirs, but from groundwater wells. Based on aquifer characteristics, staff determined that the aquifer has sufficient capacity for initial filling and occasional makeup of two proposed water storage tanks - **Closed**
- Section 2.4.12: Groundwater (**OI 2.4-2**)
  - Issue: Predict future hydrogeological conditions to determine the safety of proposed facilities from groundwater-induced loadings.
  - Resolution: The applicant provided additional field hydrogeologic data (e.g., the unconfined aquifer characters, a refined recharge and hydraulic conductivity maps). NRC staff analyzed the groundwater regime with a post-construction setting and the provided data, and confirmed that a maximum water table elevation (165 ft msl) is far below the site grade (220 ft msl) - **Closed**

2.4 Hydrology (Con't)

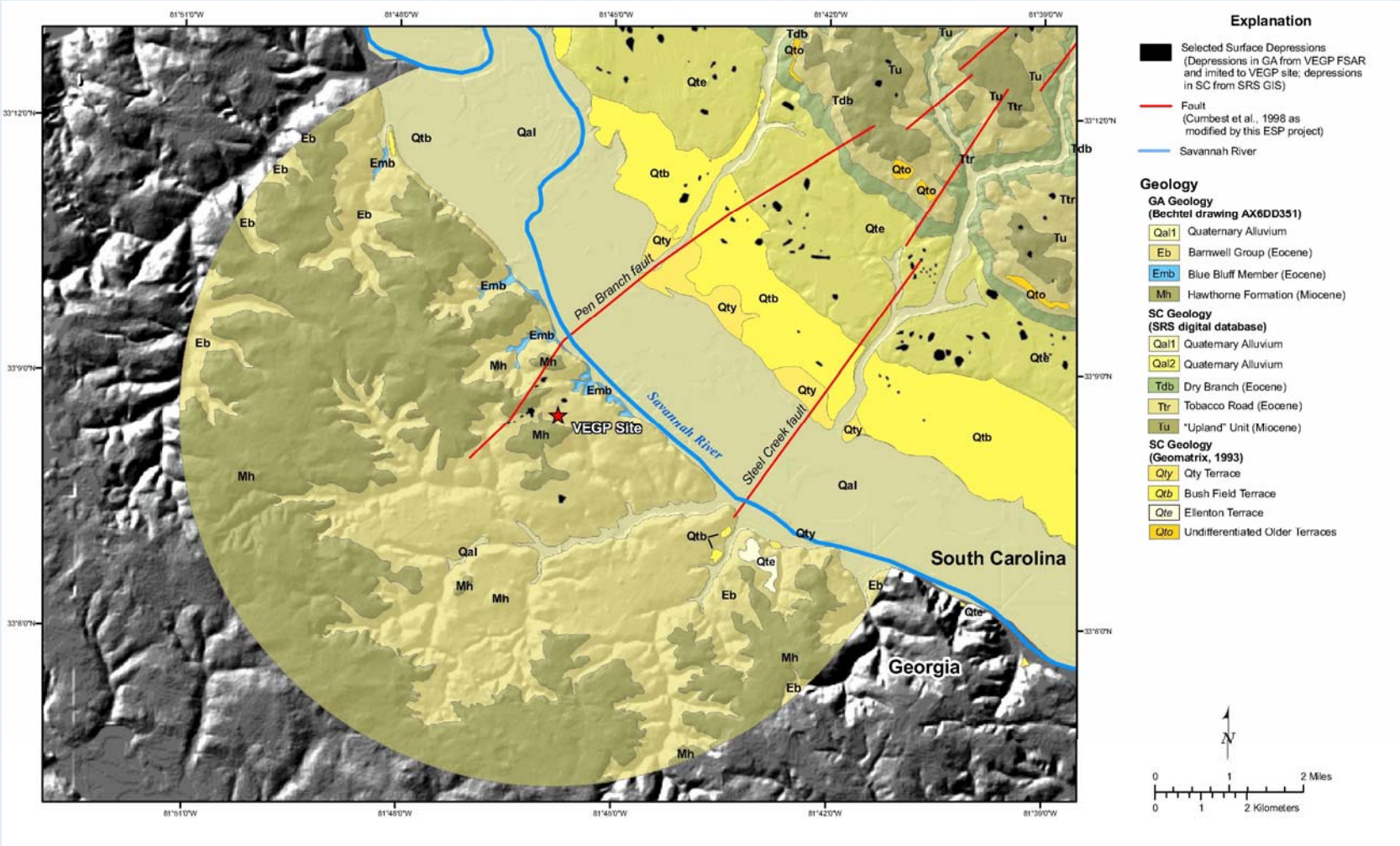
2.4.13: Accidental Releases of Radionuclides In Ground Waters

- **OI 2.4-3**
  - Issue: Consider the potential change in flow direction within the Water Table aquifer and all feasible groundwater pathways.
  - Resolution: The applicant provided additional field data; Analyses by the applicant and the NRC staff examined post-construction settings, and alternative pathways (four alternative pathways), considering an adequate number of combinations of release locations and feasible pathways - **Closed.**
- **OI 2.4-4**
  - Issue: Specify the nearest point along each potential pathway that may be accessible to the public and considered all alternative conceptual models for radionuclide transport analysis.
  - Resolution: (1) The pathways into which these releases occur leave the site boundary before entering the Savannah River; The NRC staff completed an independent analysis of the different groundwater pathways and confirmed that releases to the accessible environment met the requirement of 10 CFR Part 20, Appendix B - **Closed**.
  - COL Action Item 2.4-1: No chelating agents will be comingled with radioactive waste liquids and that such agents will not be used to mitigate an accidental release, or do the transport analysis with chelating agents.

Section 2.5: Geology, Seismology and Geotechnical Engineering

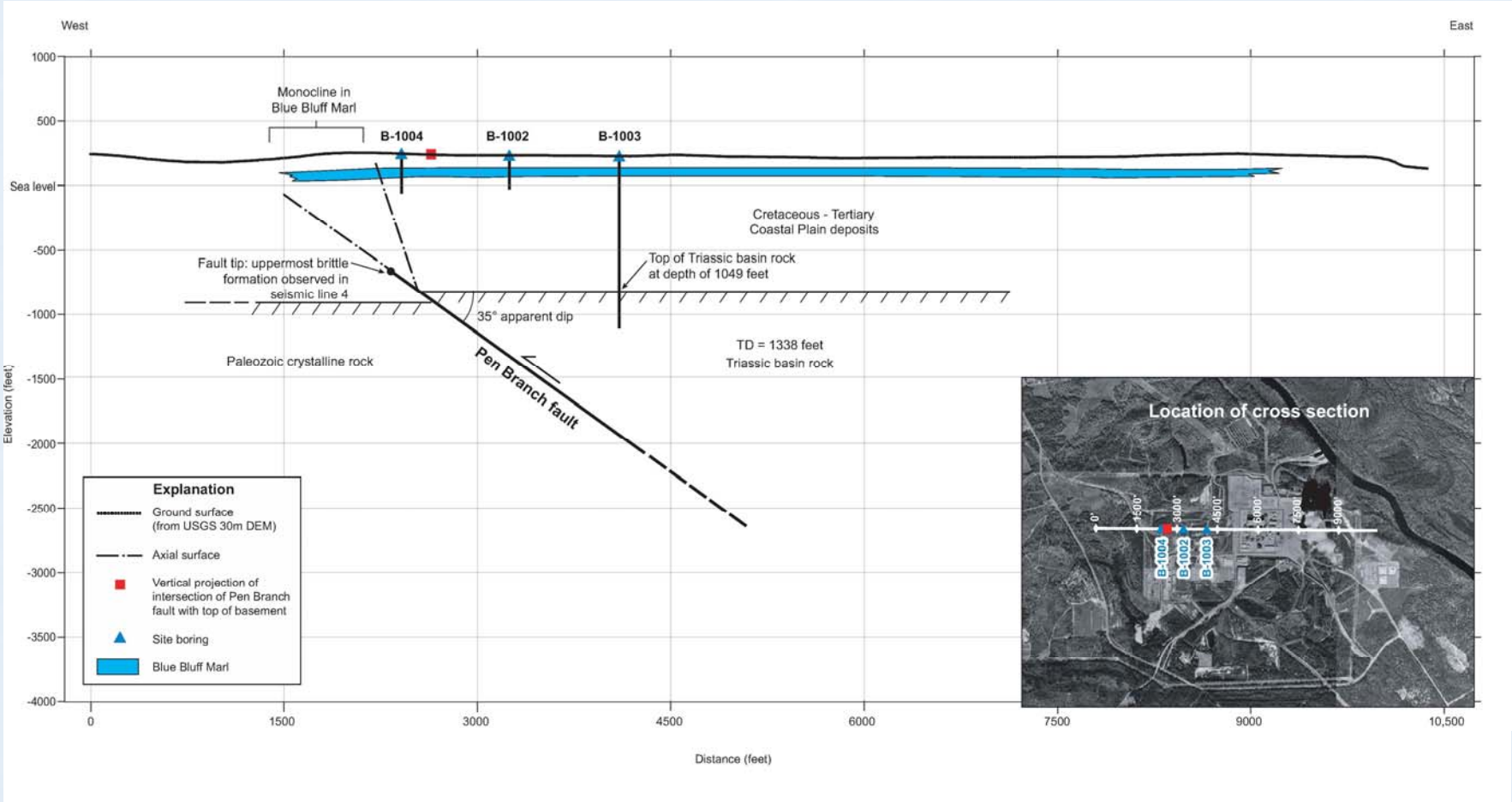
- Section 2.5.1 Site and Regional Geology
- Section 2.5.2 Vibratory Ground Motion
- Section 2.5.3 Surface Faulting
- Section 2.5.4 Stability of Subsurface Materials
- Section 2.5.5 Slope Stability

2.5.1 Basic Geologic & Seismic Information



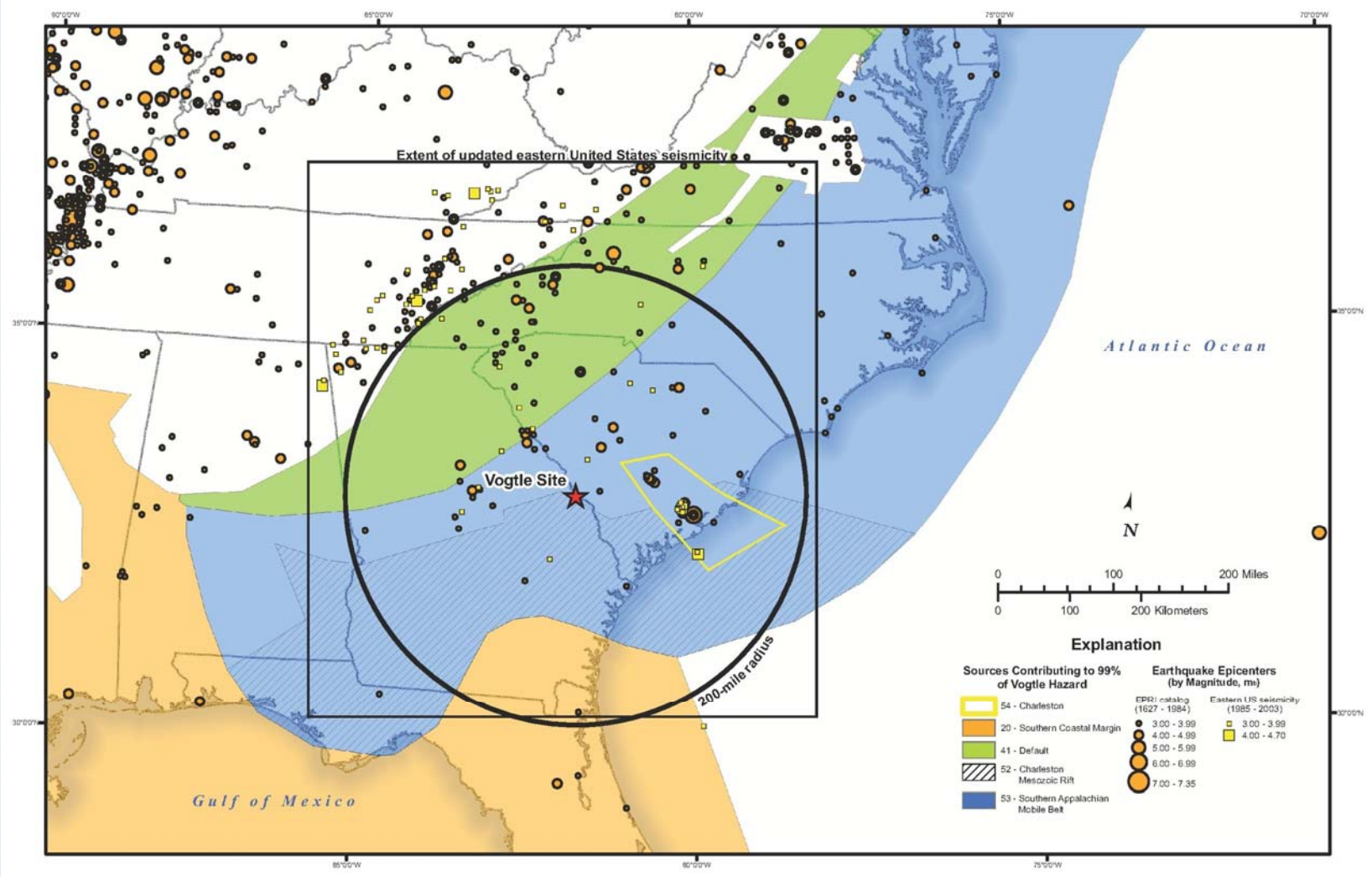
Geology in the ESP Site Vicinity

2.5.1 Basic Geologic & Seismic Information



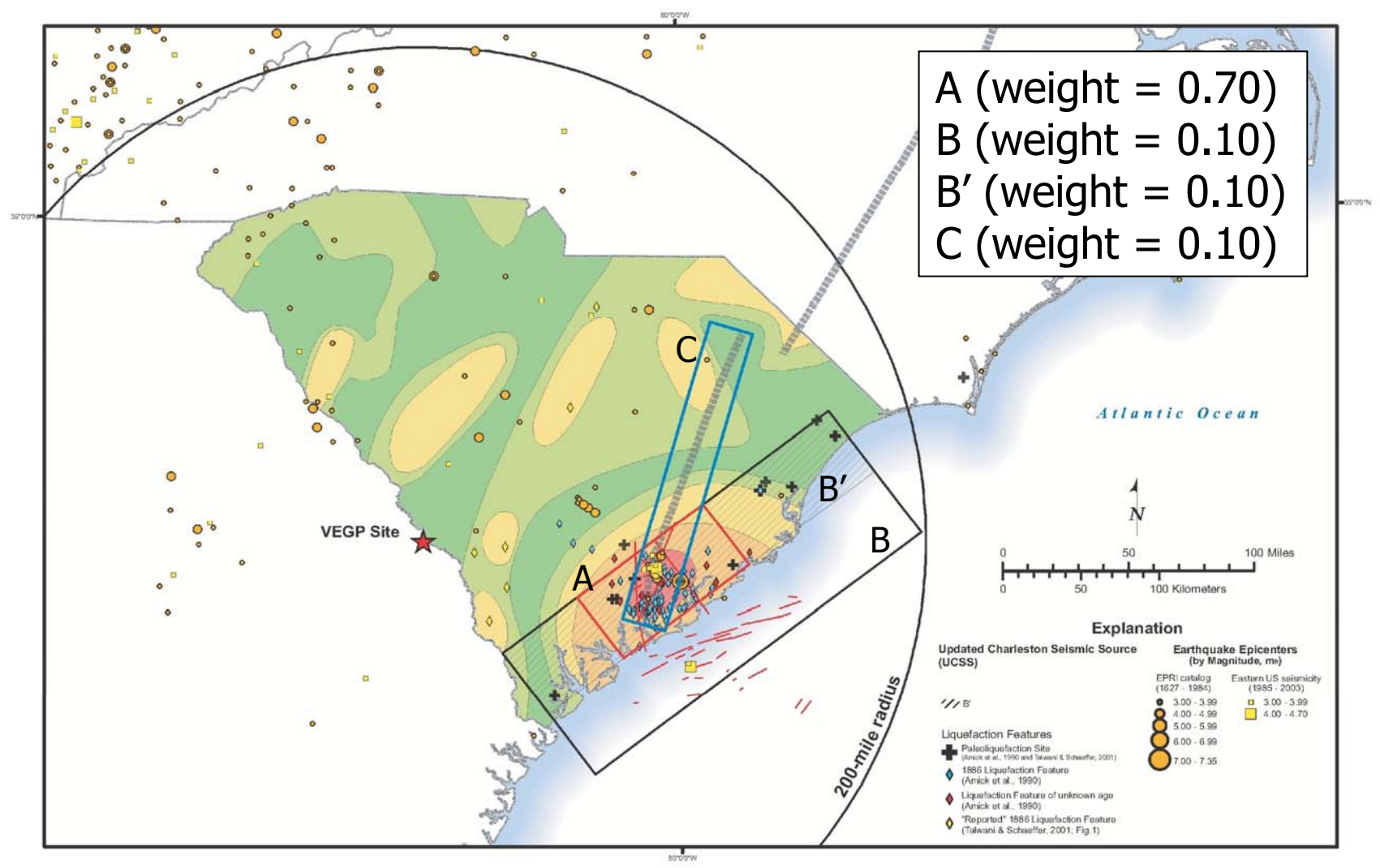
E-W Cross Section: Pen Branch Fault beneath VEGP site

2.5.2 – Vibratory Ground Motion



Example of EPRI Team Source Zones

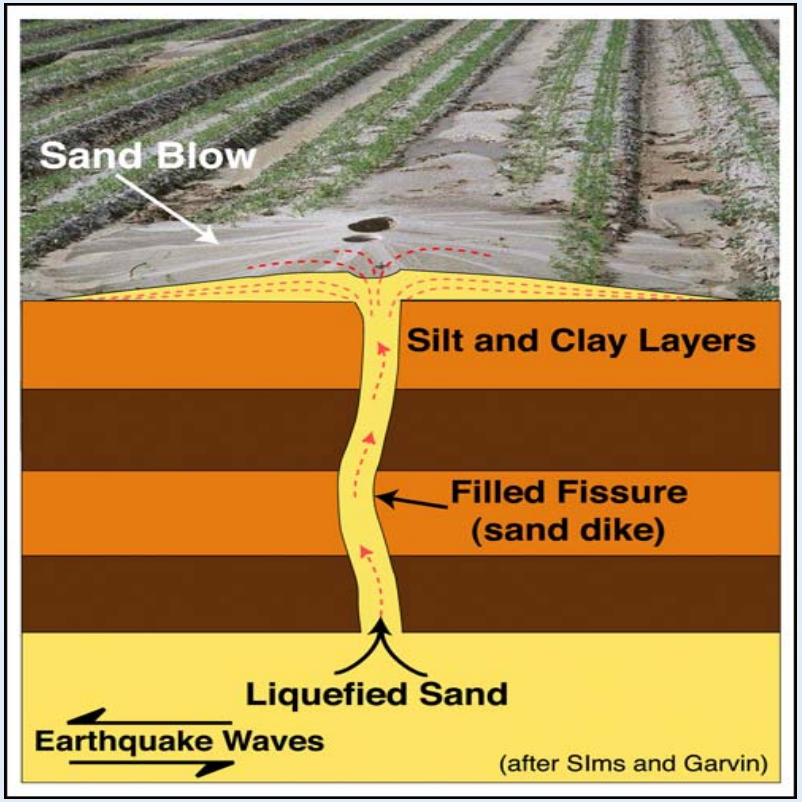
2.5.2 Vibratory Ground Motion

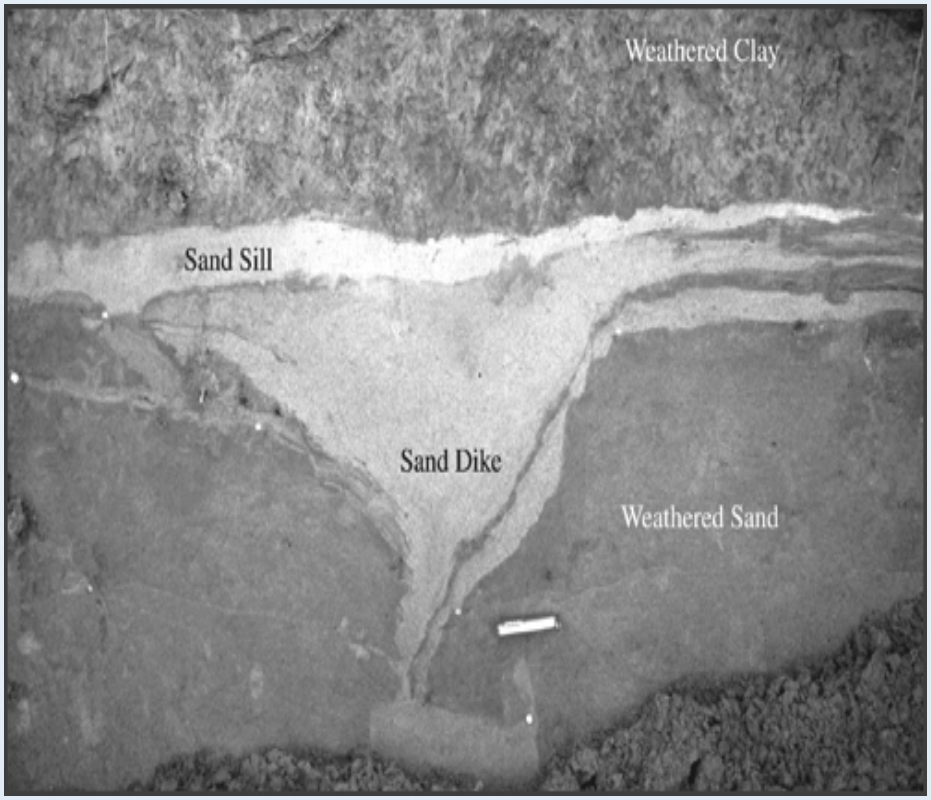


Updated Charleston Seismic Source

Charleston Update

- **Charleston update based on liquefaction features from historic and prehistoric earthquakes**
- **Liquefaction features occur in response to strong ground shaking**





Geology and Seismology

- **3 Significant Open Items addressing:**
  - **Dames and Moore EPRI-SOG Team source model**
  - **Eastern Tennessee Seismic Source Zone model**
  - **Presence of Injected Sand Dikes in site area**

2.5.4 Stability of Subsurface Material and Foundations

- **Engineering Properties of Soils and Rocks**
- **Site Explorations**
- **Geophysical Surveys**
- **Liquefaction Potential**
- **Static Stability**

2.5.4 Stability of Subsurface Material and Foundations

- **12 Open Items addressing the adequacy of:**
  - **Field and Laboratory Testing of Subsurface Materials**
  - **Measurements of Shear Wave Velocity**
  - **Development of Soil Degradation and Damping Ratio Curves**
- **Permit Condition added to require removal of Upper Sand Layer**
- **12 COL Action Items - Resolved**

2.5.4 Stability of Subsurface Material and Foundations

Site Investigations	ESP	LWA
Borings	14	174
CPTs	10	21
Test Pits	0	8
Observation Wells	15	0
P-S Velocity Logs	5	6

SER Section 13.3: Emergency Planning

- First complete EP review under 10 CFR Part 52
- Complete & Integrated Emergency Plan (ESP)
  - Included FEMA review of State/local plans
- First-of-a-kind EP Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) (30 ITAs/106 ACs)
- SER with Open Items (13 EP Open Items, 3 COL Action Items)
- Advanced SER (no EP Open Items, no EP COL Action Items, 7 EP Permit Conditions)

SER Section 13.3: Emergency Planning

SER Open Item 13.3-4 (EALs)

- NEI 07-01 EALs (AP1000 & ESBWR) (ongoing NRC endorsement review of NEI 07-01)
- AP1000 DCD EALs apply to Units 3 & 4
- Related Westinghouse amendments to AP1000 DCD (ongoing NRC AP1000 DCD review under docket 52-006)
- EAL resolution via 6 Permit Conditions (2 through 7)

SER Section 13.3: Emergency Planning

Permit Conditions:

- Emergency Action Levels (EALs)
  - 2 & 3 – NEI 07-01
  - 4 & 5 – AP1000 DCD Amendments (Units 3 & 4 TSC)
  - 6 & 7 – Full EAL set based on as-built plant, State/local agreed, & NRC approved (10 CFR Part 50, App. E.IV.B)
  - ITAAC 1.1.2 – EAL scheme consistent with RG 1.101
    - RG 1.101 is expected to endorse NEI-07-01
- Technical Support Center (TSC)
  - 8 – TSC location (AP1000 DCD, Tier 2* amendment)

SER Section 13.3: Emergency Planning

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC):

- Planning Standard (10 CFR 50.47(b)(4))
  - A standard emergency classification & action level scheme, the bases of which include facility system and effluent parameters, . . .
- EP Program Element (NUREG-0654, evaluation criterion D.1)
  - An emergency classification & EAL scheme must be established . . . The specific instruments, parameters or equipment status shall be shown for establishing each emergency class, in the in-plant emergency procedures. The plan shall identify the parameter values and equipment status for each emergency class.
- Inspections, Tests, Analysis (ITA)
  - 1.1.2 – An analysis of the EAL technical bases will be performed to verify as-built, site-specific implementation of the EAL scheme.
- Acceptance Criteria (AC)
  - 1.1.2 – The EAL scheme is consistent with Regulatory Guide 1.101 [which is expected to endorse NEI 07-01 following staff review, including AP1000-related ITAAC]

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Protecting People and the Environment

Presentation to the ACRS Full Committee

Safety Review of the Vogtle Electric Generating Plant Limited Work Authorization Request

December 4, 2008

Vogtle LWA Request

Requested Activities:

- Placement of engineered backfill
- Retaining walls
- Lean concrete backfill
- Mudmats
- Waterproof membrane

2.5.4 Stability of Subsurface Materials and Foundations

LWA Key Issues

- Adequacy of borings at the site
- Geotechnical engineering properties of the subsurface materials, especially the Blue Bluff Marl and Lower Sand Stratum
- Backfill Specifications

2.5.4 Stability of Subsurface Materials and Foundation Interfaces

LWA Key Issues – Backfill ITAAC

Design Requirement	Inspections and Tests	Acceptance Criteria
Backfill material under Seismic Category 1 structures is installed to meet a minimum of 95 percent modified Proctor compaction.	Required testing will be performed during placement of the backfill materials.	A report exists that documents that the backfill material under Seismic Category 1 structures meets the minimum 95 percent modified Proctor compaction.
Backfill shear wave velocity is greater than or equal to 1,000 fps at the depth of the nuclear island foundation and below.	Field shear wave velocity measurements will be performed when backfill placement is at the elevation of the bottom of the Nuclear Island foundation and at finish grade.	A report exists and documents that the as-built backfill shear wave velocity at the nuclear island foundation depth and below is greater than or equal to 1,000 fps.

2.5.4 Stability of Subsurface Materials and Foundations

Section 2.5.4 Conclusions

- Adequacy of borings
  - Performed substantially more borings
- Geotechnical Engineering properties of subsurface materials
  - Significant additional site investigations provided sufficiently detailed information
- Backfill Specifications
  - Test Pad measurements of backfill properties
  - ITAAC to verify compaction density and shear wave velocity

Scope of Review for Chapter 3

SRP 3.7.1-Seismic Design Parameters

- Vibratory Ground Motion
- Critical Damping
- Supporting Media (pertaining to SSI modeling)

SRP 3.7.2- Seismic Systems Analysis

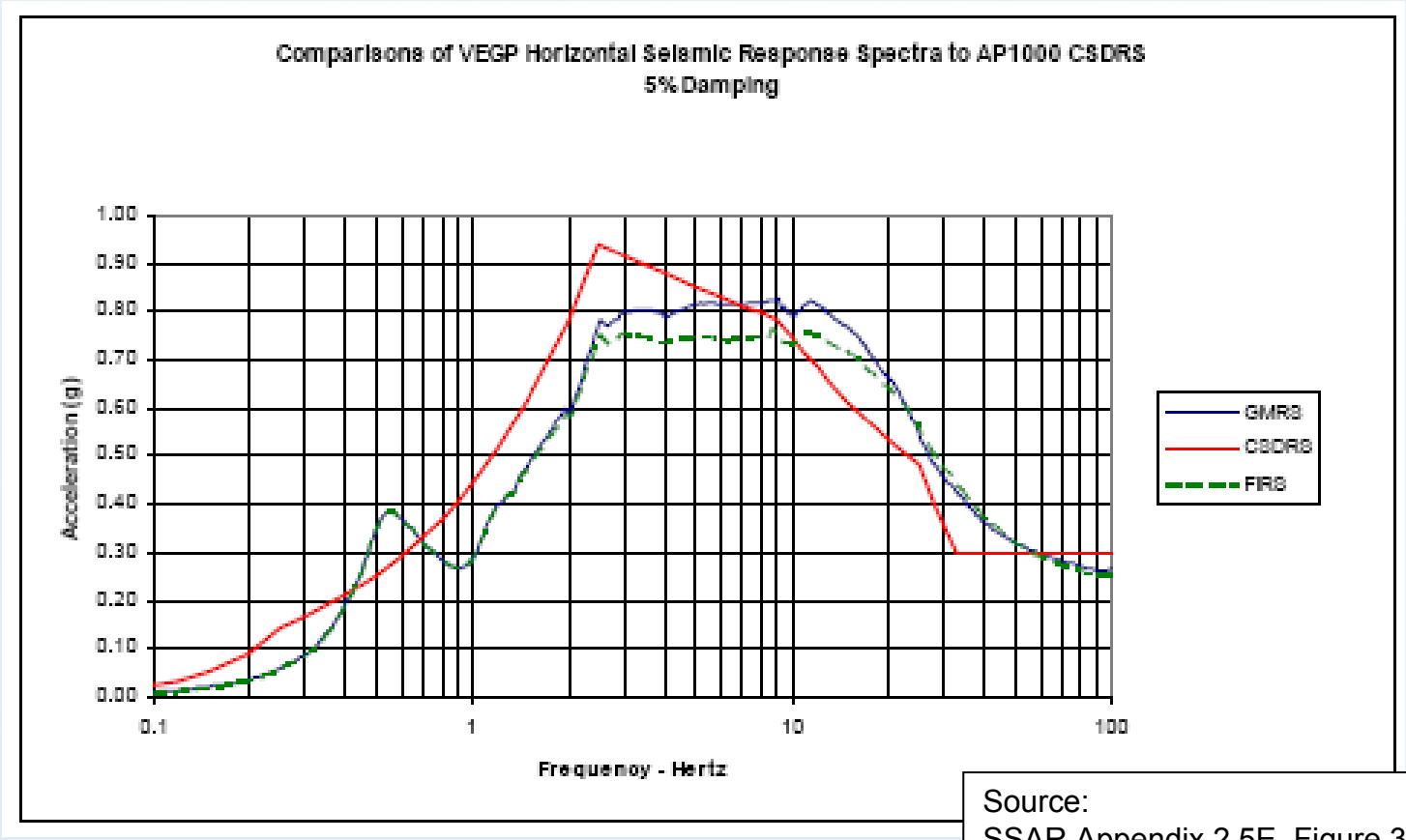
- Seismic Model Description
- Soil-Structure-Interaction Analysis

SRP 3.8.5-Foundations

- Foundation Stability
  - Sliding
  - Overturning

SER Section 3.7.1
Seismic Design Parameters

Comparison of Vogtle Horizontal GMRS and FIRS with AP1000 CSDRS



Source:
SSAR Appendix 2.5E, Figure 3-4

SER Section 3.7.1 Seismic Design Parameters

Technical Evaluation/Findings

Vibratory Ground Motion

- Approximate method was used for developing the FIRS. Review indicates that the method results in a conservative estimate of horizontal seismic demand.

- The FIRS defined as an outcrop motion in the free field satisfied the minimum PGA value of 0.10g (10 CFR Part 50, Appendix S)

Critical Damping

- The critical structural damping values used in SSI analysis were consistent with damping values provided in RG 1.61.

Supporting Media

- SSI modeling assumptions properly account for site characteristics such as depth of soil over bedrock, soil properties, soil layering characteristics and groundwater elevation.

SER Section 3.7.2
Seismic Systems Analysis

Technical Evaluation/Findings

Seismic Model

- The use of 2D SASSI models is acceptable for the evaluation of sliding stability and bearing pressure demands.

Soil-Structure-Interaction Analysis

- Staff compared the analysis results (e.g., ZPA values near the NI center-of-gravity) with the AP1000 DCD soft soil case and found them to be similar.

- Maximum seismic base shear forces are acceptable based on staff simplified independent calculations.

SER Section 3.8.5 Foundations

Summary of Application

- Test data of waterproofing membrane indicate a coefficient of friction of 0.7 between the membrane and the concrete mudmat.
- Test data indicate a coefficient of friction of 0.45 for soil immediately below mudmat.
- Soil test data indicate a bearing capacity of 42 ksf.

SER Section 3.8.5 Foundations

Technical Evaluation/Findings

NI Structure Stability Analysis

- Staff reviewed the maximum horizontal seismic forces and maximum friction forces below the basemat.

Maximum NI Seismic Forces

Reaction	Vogtle Lower Bound	Vogtle Best Estimate	Vogtle Upper Bound
Seismic Shear NS	78.3 E3 kips	82.5 E3 kips	89.0 E3 kips
Seismic Shear EW	88.9 E3 kips	89.8 E3 kips	95.8 E3 kips
Friction Force	117.3 E3 kips	116.7 E3 kips	116.4 E3 kips

- The NI structure will not slide during the SSE, because the frictional force is greater than the inertial force.

SER Section 3.8.5 Foundations

Technical Evaluation/Findings (Continued)

Bearing Capacity

- The maximum dynamic bearing pressure on soils for the NI, radwaste, annex, and turbine buildings are 17.95 ksf, 1.68 ksf, 7.20 ksf, and 2.54 ksf, respectively, during the SSE.

- The minimum factor of safety with respect to a failure of the dynamic soil bearing capacity during the SSE is 2.34 (42 ksf divided by 17.95).

Summary Findings

SRP Section 3.7.1 Seismic Design Parameters

- Adequately developed seismic design parameters.
- Met the applicable regulatory requirements.

SRP Section 3.7.2 Seismic Systems Analysis

- Adequately performed site-specific 2D SSI analysis for the purpose of determining the maximum seismic demands for use in the NI structure stability and maximum dynamic soil bearing evaluations.
- Staff's evaluation of in-structure response will be done as part of the SCOL review.
- Met the applicable regulatory requirements.

SRP Section 3.8.5 Foundations

- Demonstrated that the mudmat and the waterproofing membrane are adequate and that the NI foundation is stable during an SSE.
- Met the applicable regulatory requirements.

Advanced SER/LWA Conclusions

- The VEGP ESP application meets the applicable standards and requirements of the Act and the Commission's regulations.
- Site Characteristics, Design Parameters, and Terms and Conditions proposed to be included in the Permit meet the applicable requirements of Part 52.
- There is reasonable assurance that the site is in conformity with the provisions of the Act, and the Commission's regulations.
- The proposed ITAAC are necessary and sufficient, within the scope of the ESP, to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the emergency plans, the provisions of the Act, and the Commission's regulations.
- Issuance of the permit will not be inimical to the common defense and security or to the health and safety of the public

BACKUP SLIDES for ESP

2.5.4 Stability of Subsurface Materials and Foundations

- **Insufficient Laboratory Testing (Open Item 2.5-11)**
  - Issue: Conduct sufficient field & laboratory tests to reliably determine subsurface soil static & dynamic properties at the ESP site
  - Resolution: In support of the LWA request, the applicant performed additional field and laboratory investigations which were used to determine the static and dynamic properties of the subsurface materials

2.5.4 Stability of Subsurface Materials and Foundations

- **Blue Bluff Marl Load-bearing Properties (Open Item 2.5-12)**
  - Issue: Provide sufficient data to derive reliable site-specific engineering parameters for the Blue Bluff Marl
  - Resolution: The applicant performed SPT and split-spoon sampling in almost all ESP borings and conducted additional laboratory tests such as grain size distribution, Atterberg Limits, and carbonate content

2.5.4 Stability of Subsurface Materials and Foundations

- **Undrained Shear Strength (Open Item 2.5-13)**
  - Issue: Provide sufficient sampling and testing results to reliably derive the undrained shear strength and other related engineering parameters
  - Resolution: The applicant revised the SSAR using the additional field and laboratory investigations to provide the preconsolidation pressure calculations and overconsolidation ratios

2.5.4 Stability of Subsurface Materials and Foundations

- **Angles of Friction (Open Item 2.5-14)**
  - Issue: Provide reliable effective angles of internal friction for the subsurface soils
  - Resolution: The applicant revised the SSAR to include a description of the empirical correlation of average effective angles of internal friction which were used.

2.5.4 Stability of Subsurface Materials and Foundations

- **Blue Bluff Marl Behavior (Open Item 2.5-15)**
  - Issue: Provide information to demonstrate that the Blue Bluff Marl will behave as a hard clay or soft rock material
  - Resolution: Additional borings in support of the LWA request were used to demonstrate the behavior of the BBM

2.5.4 Stability of Subsurface Materials and Foundations

- **Elastic Modulus (Open Item 2.5-16)**
  - Issue: Provide sufficient site-specific data to justify the determination of the design parameter elastic modulus “E” for the Upper and Lower Sand Strata
  - Resolution: The applicant used representative data from the SPTs performed in support of the LWA request to determine E

2.5.4 Stability of Subsurface Materials and Foundations

- **Unit Weight Values (Open Item 2.5-17)**
  - Issue: Develop sufficient data (vs. values from previous investigations) to calculate the unit weight values for the ESP subsurface soils
  - Resolution: Additional data were included in support of the LWA request and were used to calculate the unit weight of the subsurface materials

2.5.4 Stability of Subsurface Materials and Foundations

- **SSAR Degradation Curve Revision (Open Item 2.5-20)**
  - Issue: Revise SSAR Sections 2.5.2.5.1.5, 2.5.4.7.2.1, and 2.5.4.7.2.2, along with associated tables and figures, to show the degradation curves only at a ≤1% cyclic shear strain
  - Resolution: The SSAR was revised accordingly.

2.5.4 Stability of Subsurface Materials and Foundations

- **Liquefaction Potential of Blue Bluff Marl (Open Item 2.5-21)**
  - Issue: Provide sufficient ESP soil property data to confirm that the Blue Bluff Marl is non-liquefiable
  - Resolution: Additional borings completed in support of the LWA were used to confirm the negligible liquefaction potential of the BBM

2.5.4 Stability of Subsurface Materials and Foundations

- **Bearing Capacity (Open Item 2.5-22)**
  - Issue: Provide appropriate bearing capacity estimates
  - Resolution: Later revisions to the SSAR in support of the LWA request included the bearing capacity calculations and settlement estimates

2.5.4 Stability of Subsurface Materials and Foundations

- **Previous COL Action items**
  - 2.5-1 A COL or CP applicant will need to confirm the absence of soft materials in the load bearing layers.
  - 2.5-2 A COL or CP applicant will need to confirm the locations of the soft zones and evaluate the potential impact of the soft zones on the foundation and structures.
  - 2.5-3 A COL or CP applicant will need to provide chemical test results on the backfill.

2.5.4 Stability of Subsurface Materials and Foundations

- **Previous COL Action items**
  - 2.5-4 A COL or CP applicant will need to submit plot plans and profiles of all seismic Category I facilities for comparison with the subsurface profile and material properties.
  - 2.5-5 A COL or CP applicant will need to provide detailed excavation and backfill plans during the COL stage.
  - 2.5-6 A COL or CP applicant will need to provide sufficient information to show the backfills meet the minimum shear wave requirement.

2.5.4 Stability of Subsurface Materials and Foundations

- **Previous COL Action items**
  - 2.5-7 A COL or CP applicant will need to submit ground water condition evaluations and a detailed dewatering plan during the COL stage.
  - 2.5-8 A COL or CP applicant will need to demonstrate quantitatively whether the observed large settlement that occurred at the existing VEGP units will occur at the ESP site and have no impact on the new units.
  - 2.5-9 A COL or CP applicant will need to provide more details regarding the bearing capacity during the COL stage.

2.5.4 Stability of Subsurface Materials and Foundations

- **Previous COL Action items**
  - 2.5-10 A COL or CP applicant will need to describe the design criteria and design methods, including the factor of safety for slope stability at the COL stage.
  - 2.5-11 A COL or CP applicant will need to provide information regarding ground improvement after removal of Upper Sand Stratum for the ESP site.

Key Review Areas

Chapter 15 – Radiological Consequences of Design Basis Accidents

- Permit condition 9:
  - The permit will include the time-dependent isotropic release (source term) for each DBA
  - COL applicant referring to certified design only required to demonstrate site-specific atmospheric dispersion factor values less than used in DCD to show compliance with Part 100, 10 CFR 52.79 and GDC-19
  - Permit condition to not require holder of Vogtle ESP do anything more than any other COL applicant referring to a certified design. If ESP holder does not refer to a certified design, COLA would demonstrate that plant source term is bounded by the source term in ESP

Key Review Areas

- Applicant used AP1000, DCD Rev. 15
  - Calculated site-specific short term atmospheric dispersion factors (χ/Qs)
  - Ratio of site-specific to design reference χ/Qs applied to DCD calculated DBA dose to give estimate of site-specific DBA dose for each DBA in AP1000 DCD
  - Since each site-specific χ/Q was less than comparable design reference χ/Q, then site-specific DBA doses are less than AP1000 DCD DBA doses and therefore meet regulatory criteria
    - Can confirm by taking AP1000, Rev. 15 source term release rates for each DBA and calculating site-specific DBA dose using site-specific χ/Qs

Early Site Permit

Jim Davis
ESP Project Engineer
Southern Nuclear

Agenda

- Introduction
- Schedule
- Early Site Permit (ESP) Overview
- Limited Work Authorization (LWA) Overview

Introduction

- Southern Nuclear is pursuing an Early Site Permit (ESP) in accordance with 10 CFR 52 Subpart A-Early Site Permits
- In addition Southern Nuclear is seeking a Limited Work Authorization (LWA) in accordance with 10 CFR 50.10

Introduction

- An ESP grants approval of a site for one or more nuclear power facilities separate from the filing of an application for a construction permit or combined license for the facility
- The requested LWA will allow a limited scope of safety-related construction activities to proceed at applicants risk as long as a site redress plan is included.

U.S.NRC

UNITED STATES NUCLEAR REGULATORY COMMISSION

Protecting People and the Environment

Strategy for Revising 50.46(b) Fuel Performance Criteria

ACRS Full Committee Meeting
December 4, 2008

Paul M. Clifford
Division of Safety Systems
Nuclear Reactor Regulation

Rulemaking Objectives

- Following Commission directive, develop a performance-based rule which enables licensees to use advanced cladding materials without needing an exemption.
  - Replace prescriptive criteria with performance-based regulatory requirements.
  - Expand applicability beyond “zircaloy or ZIRLO”.

- Capture results of High Burnup LOCA Research Program.
  - Research identified new embrittlement mechanisms which necessitate rule changes.

Applicability of Rule

Current Regulation:

- Paragraph (a)(1)(i) limits applicability to “zircaloy or ZIRLO”.

Research Finding:

- Empirical database includes wide range of zirconium alloys.

Plant Safety:

- No impact.

Strategy for Revising Regulation:

- Replace “zircaloy or ZIRLO” with less specific terminology (e.g., approved zirconium-alloy).
- Applicability to new alloys will need to be demonstrated by testing.

Peak Cladding Temperature

Current Regulation:

- Paragraph (b)(1) limits PCT to 2200☜F.

Research Finding:

- Post quench ductility (PQD) decreases dramatically in samples oxidized beyond 2200☜F.
- Confirms current regulatory criterion.

Plant Safety:

- No impact.

Strategy for Revising Regulation:

- No change.

Local Oxidation

Current Regulation:

- Paragraph (b)(2) limits local oxidation to 17% ECR.

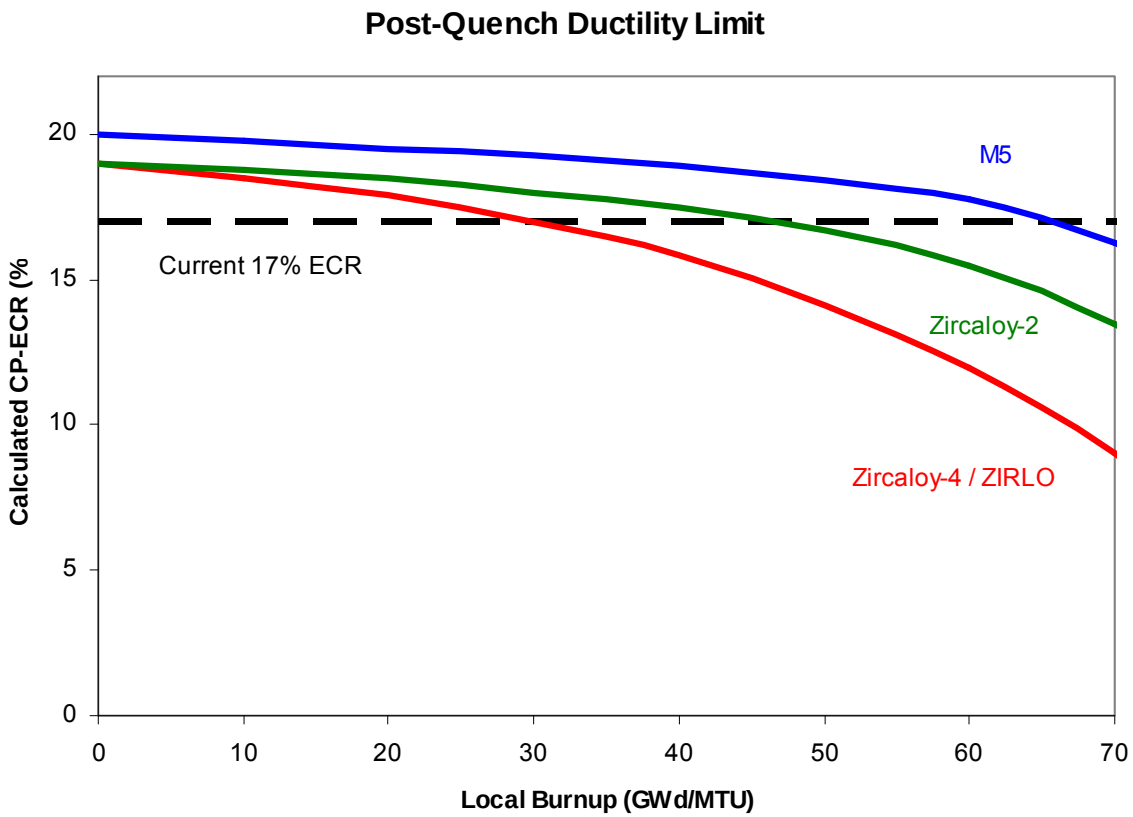
Research Finding:

- New cladding embrittlement mechanism identified.
  - PQD sensitive to pre-transient cladding hydrogen concentration.
- A constant 17% ECR limit does not always ensure PQD.
- Information Notice 98-29 adjustment (subtract initial oxide layer from 17% ECR limit) may not always ensure PQD.

Local Oxidation (cont.)

Plant Safety:

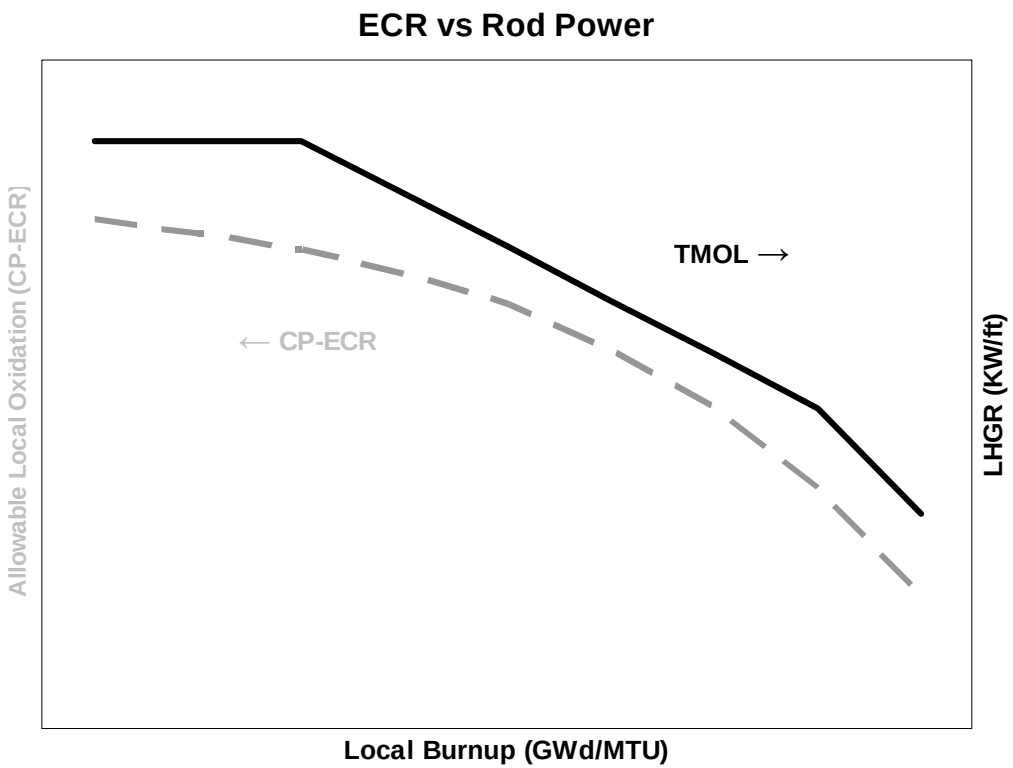
- Modern alloys exhibit unirradiated brittle transition at or above 17% ECR.



Local Oxidation (cont.)

Plant Safety:

- Highest power fuel rods challenge 2200☜F and 17% ECR limits.
- Corrosion build-up coincident with U^{235} depletion (diminishing rod power).
- Lower power fuel rods experience more benign transient.

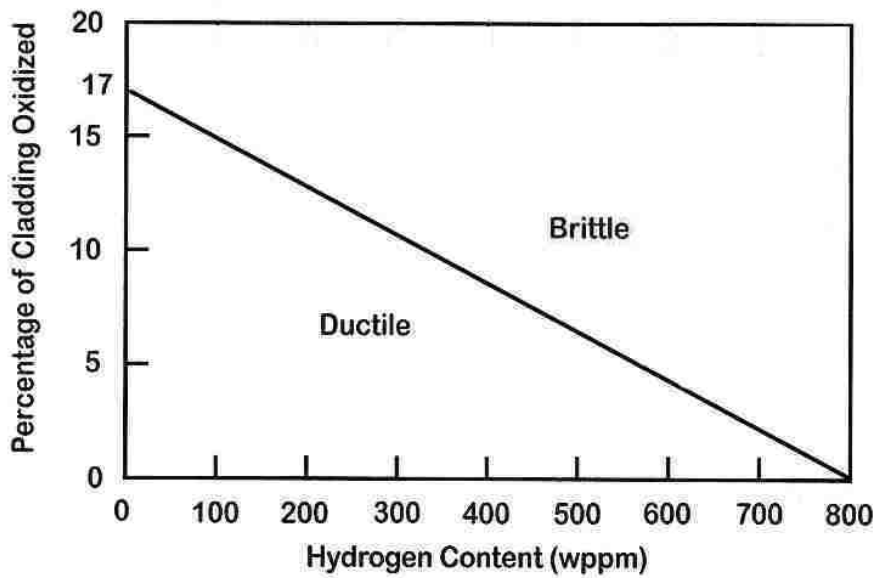


Local Oxidation (cont.)

Strategy for Revising Regulation:

Alternative Regulations:

1. Generic PQD criteria specified within rule.



2. Optional test program for defining alloy-specific or temperature-specific PQD criteria.

ID Oxygen Diffusion

Current Regulation:

- None.

Research Finding:

- Oxygen from fuel bonding layer (on cladding ID) diffuses into the base metal and exacerbates cladding embrittlement.

Plant Safety:

- Current methods require double sided oxidation within the balloon region.
- Higher burnup fuel rods operating at lower power will experience more benign transient.

Strategy for Revising Regulation:

- New requirement within rule.

Breakaway Oxidation

Current Regulation:

- None.

Research Finding:

- New cladding embrittlement mechanism identified.
  - Protective tetragonal oxide transforms to monoclinic structure.
  - Hydrogen uptake promotes cladding embrittlement.
- Timing of transformation sensitive to manufacturing process.

Plant Safety:

- Measured breakaway time for domestic alloys exceed 3000 seconds.
- SBLOCA analysis coupled with reasonable operator actions show that the duration at elevated temperatures remains below breakaway time.

Breakaway Oxidation (cont)

Strategy for Revising Regulation:

- New performance requirement within rule.
  - Required testing to establish measured break-away time.
- Required periodic testing.

Regulatory Challenge

- Developing a performance-based rule which meets the objectives of the rulemaking plan (e.g., optional testing program) while satisfying legal requirements (e.g., specific enforceable requirements).
  - Performance-based rule more difficult to script.
  - Specifying optional test protocols within rule versus regulatory guidance document.

Optional Test Program

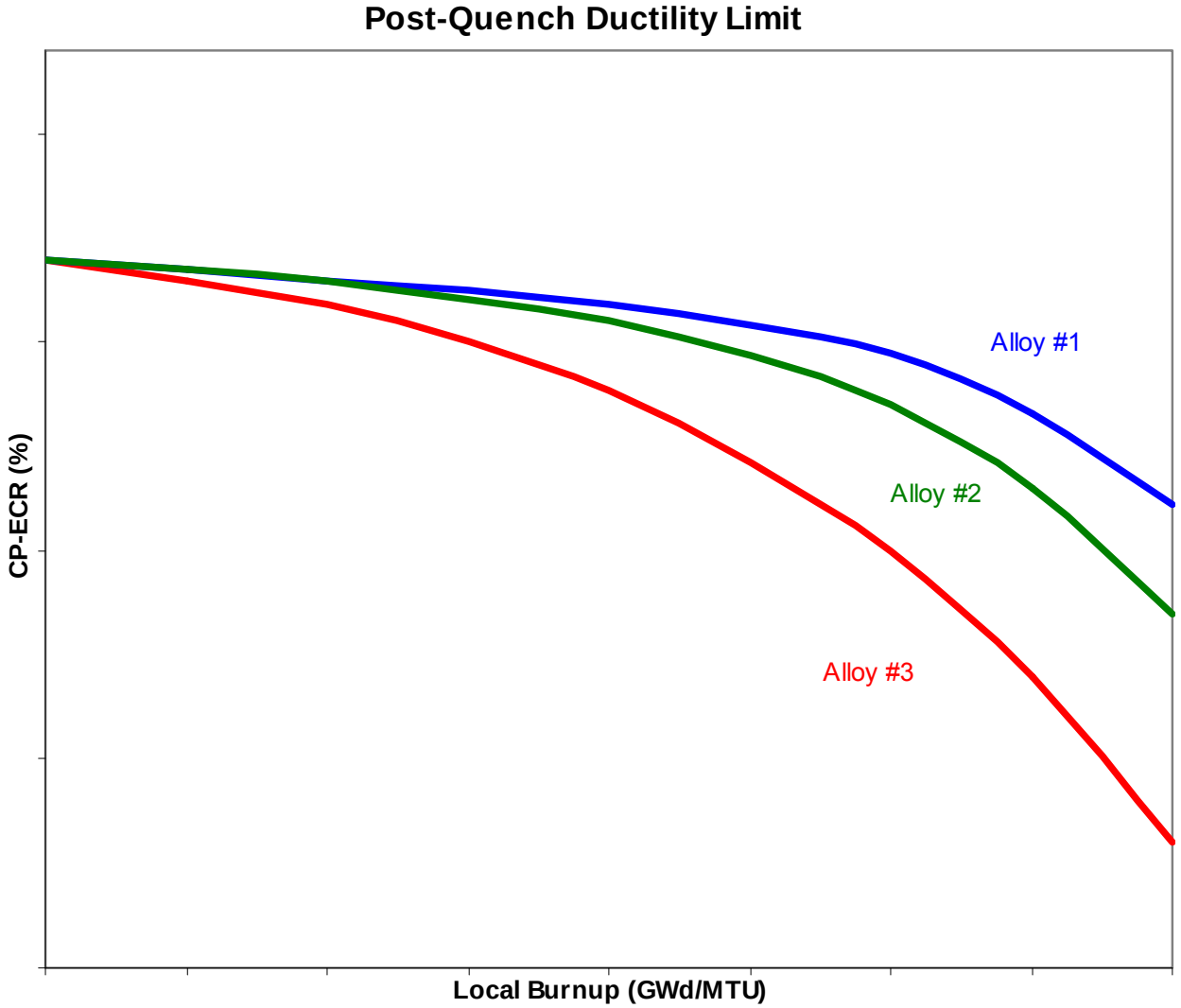
- Regulations within 50.46(b)(2) specify general requirements for optional testing:
  - Criterion for the ductility test would be 1% plastic strain using ring-compression tests.
  - Criterion for the breakaway oxidation test would be 200 wppm hydrogen uptake.

- Acceptable experimental protocols for establishing cladding ductility criteria and breakaway oxidation limits would be provided within a comprehensive test procedure.

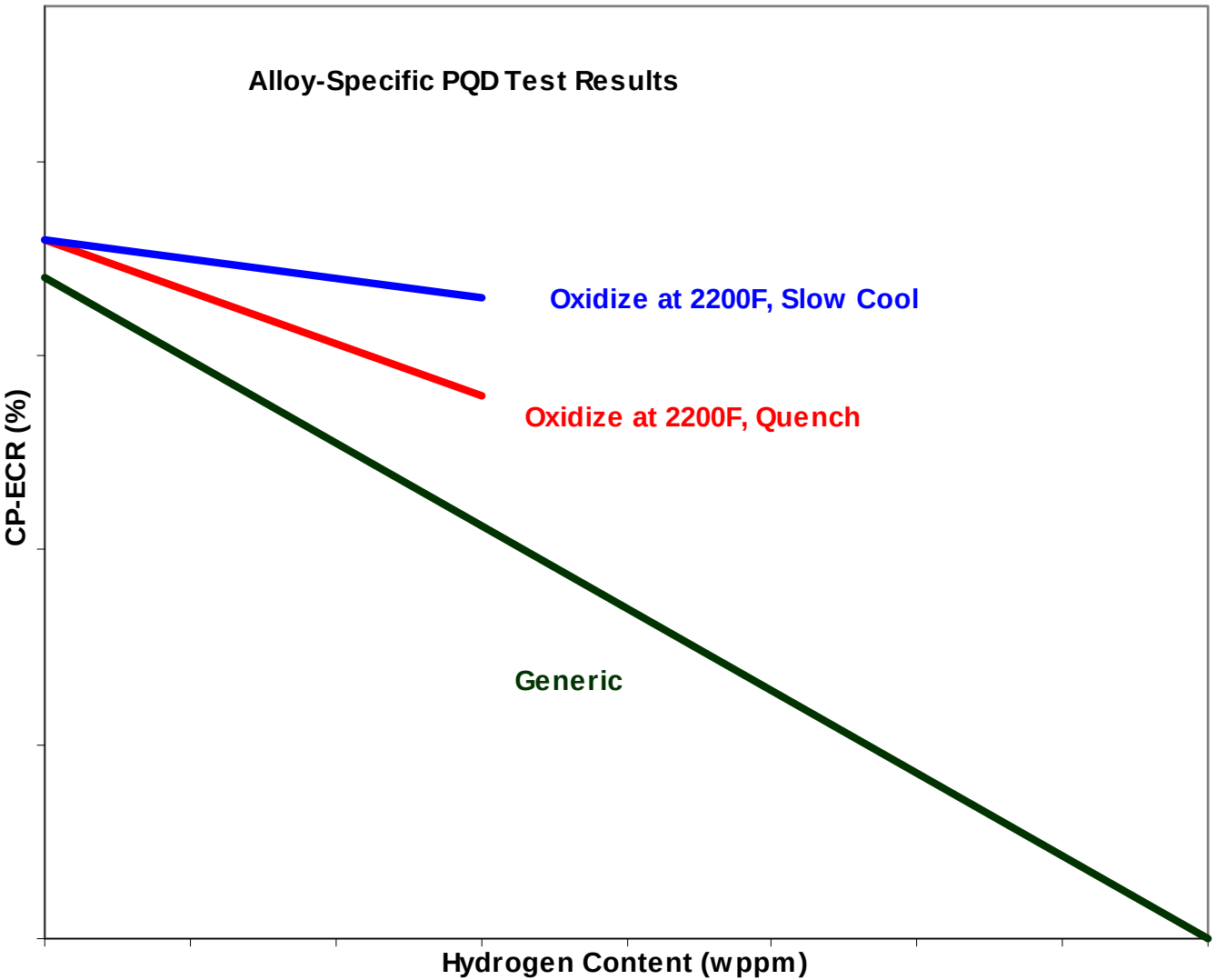
Implementing Alternative PQD Criteria

Implementing PQD Curve

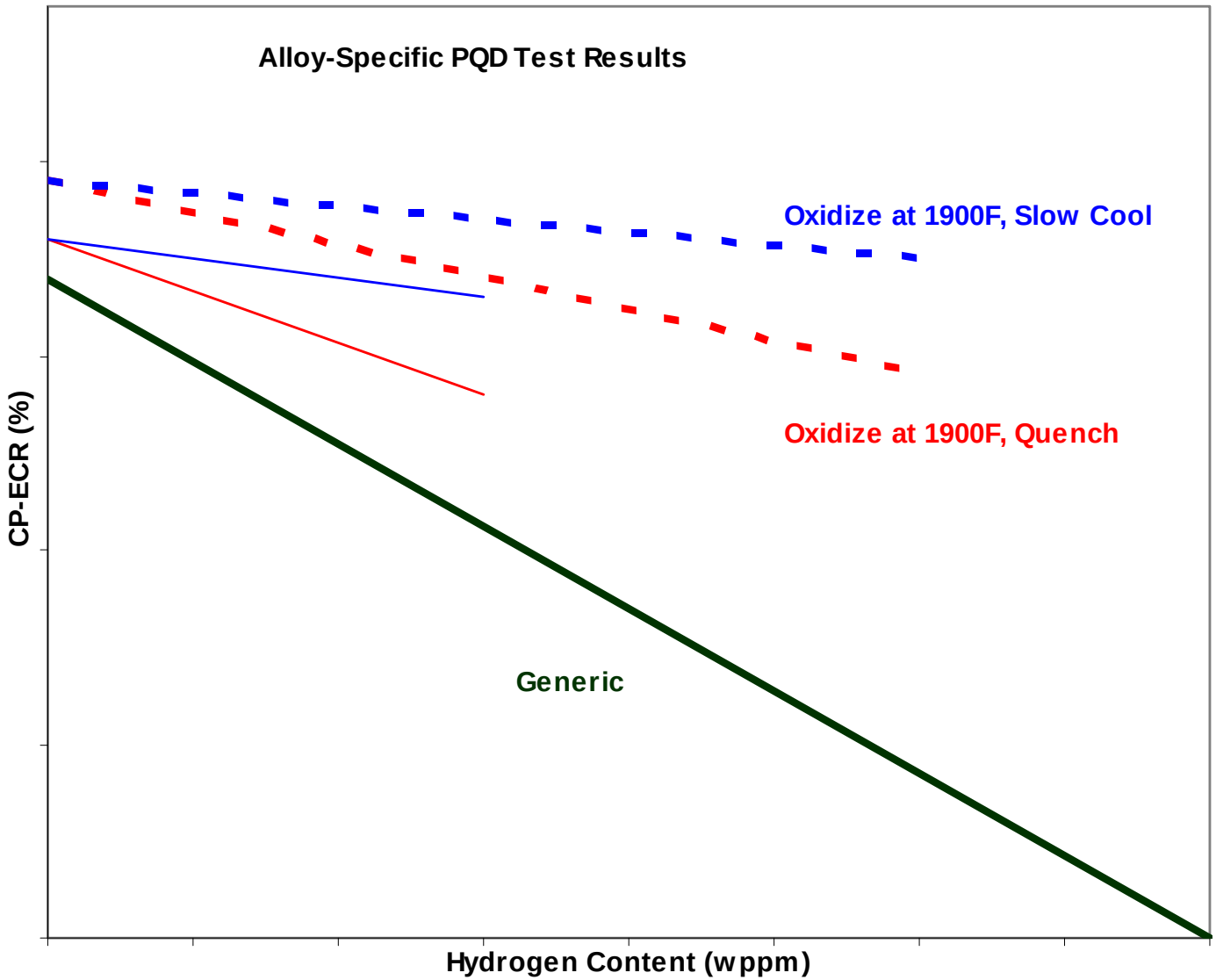
(initial hydrogen content converted to burnup)



Added Flexibility



Added Flexibility (cont.)



Path Forward

Ongoing Research Activities

1. Development and validation of a comprehensive, performance-based test procedure.
2. Additional PQD tests at intermediate hydrogen levels.
3. Additional breakaway tests to investigate whether the timing of breakaway oxidation is sensitive to variations in temperature profile or thermal cycling.

Advance Notice of Proposed Rulemaking

- ANPR process designed to enhance public participation during significant rulemaking campaigns. Benefits include:
  - Public response to rule concept and/or staff requests for additional information factored into the rulemaking proceeding and language of proposed rule language
  - Facilitates formal stakeholder interaction on the rulemaking while further research is acquired.

Staff White Paper Concerning Containment Overpressure Credits: Risk Considerations

Marty Stutzke, RES/DRA

ACRS Presentation
December 4, 2008

Staff Position

- The staff will continue to consider risk insights in its reviews of license amendment request (LARs) that contain requests for containment overpressure (COP) credits in accordance with its existing processes, which implement Commission-approved guidance.

Use of Risk Insights to Support Regulatory Decisionmaking

- NRR Office Instruction LIC-101 describes the staff's process for reviewing LARs.
- Risk-informed LARs
  - Guidance: RG 1.174 and SRP Section 19.2.
  - Risk insights provide one of the primary justifications for acceptability of the LAR.
  - All five key principles of risk-informed decisionmaking stated in RG 1.174 should be met.
  - Licensees voluntarily submit risk informed LARs.
- Non-risk-informed LARs
  - Guidance: SRP Section 19.2, Appendix D.
  - Risk insights may be used to determine whether or not a proposed plant change rebuts the presumption of adequate protection despite the fact that the proposed change meets currently specified regulatory requirements.
    - If one or more of the five key principles are not met, then a more complete assessment (deterministic and/or probabilistic) should be performed.
    - The fact that one or more of the five key principles is not met does not automatically imply a lack of adequate protection (i.e., the five key principles do <u>not</u> define "adequate protection").
  - Staff assumes the burden of demonstrating that the presumption of adequate protection is not supported.

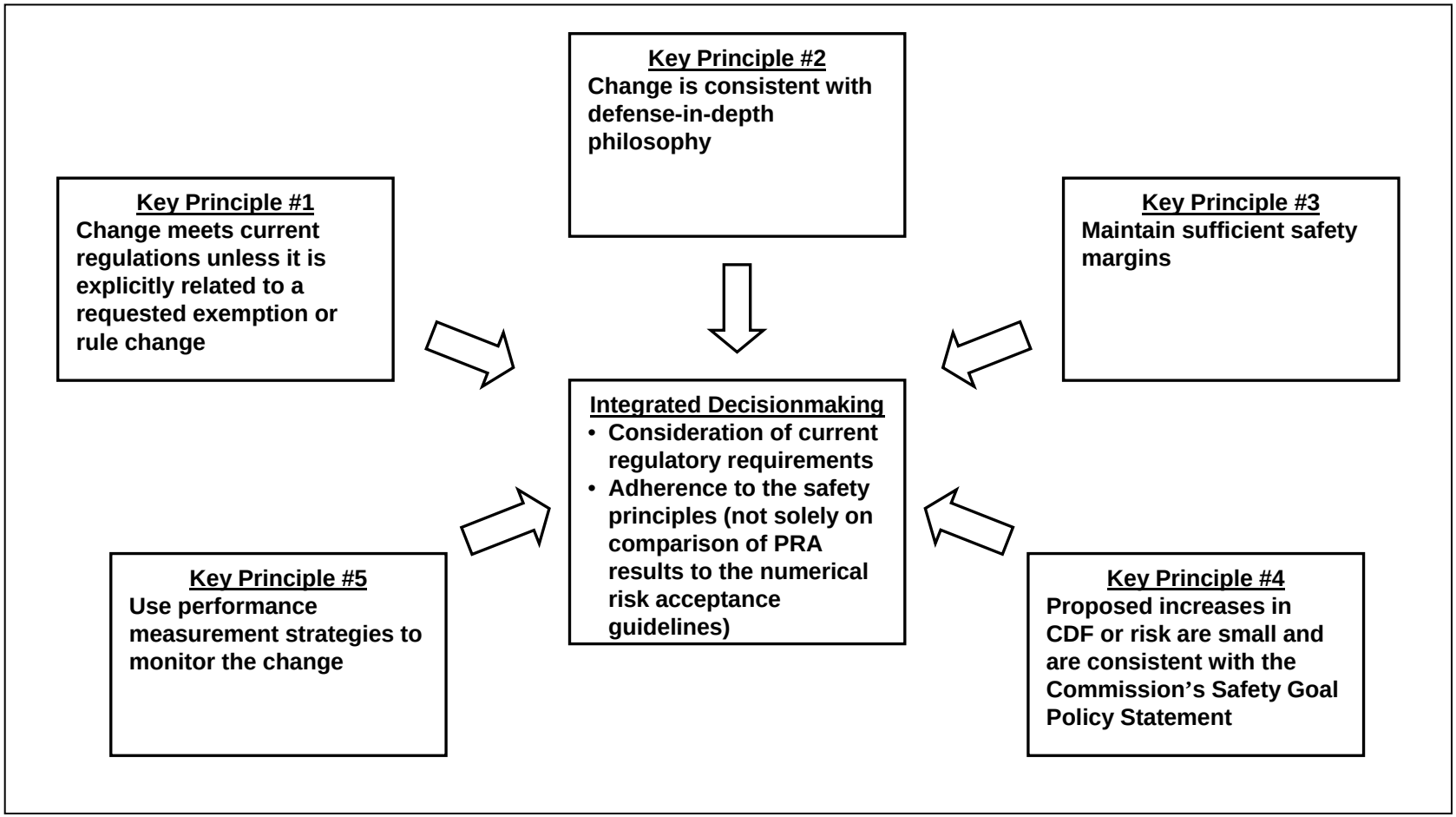
Invoking SRP Section 19.2, Appendix D

- SRP Section 19.2, Appendix D is invoked when the staff believes that a non-risk-informed LAR:
  - Significantly changes allowed outage time, initiator probability, mitigation probability, recovery time, or operator action,
  - Significantly changes functional requirements or redundancy,
  - Significantly affects the basis for successful safety function, or
  - Creates “special circumstances:”
    - Substantially increases the likelihood or consequences of accidents that are risk significant, but beyond the design and licensing basis of the plant,
    - Degrades multiple levels of defense or Reactor Oversight Process cornerstones,
    - Significantly reduces availability/reliability of systems, structures and components that are risk significant, but not required by regulations, or
    - Synergistic or cumulative effects that significantly impact risk.

Using SRP Section 19.2, Appendix D

- The numerical risk acceptance guidelines and safety principles in RG 1.174 are intended to provide a basis for finding that there is reasonable assurance of adequate protection.
  - The guidelines and safety principles serve as a point of reference for gauging risk impact, but are not legally binding requirements.
  - SRP Section 19.2, Appendix D emphasizes the need to differentiate between the concept of adequate protection and the numerical risk acceptance guidelines.
- The staff must notify the Commission whenever “special circumstances” are identified.
- The decision to reject a non-risk-informed LAR on the basis of risk will be made by the Director, NRR.

Five Key Principles of Risk-Informed Decisionmaking



Application of the Five Key Principles to COP Credits

- Principle #1: Compliance with regulation
  - There is no regulation that prohibits use of a COP credit.
- Principle #2: Defense-in-depth
  - RG 1.174 focuses on understanding how a proposed change affects the physical barriers that provide defense-in-depth.
    - A COP credit reduces defense-in-depth because it introduces a dependency between the containment and fuel cladding barriers.
  - SRP Section 19.2 also discusses the need to consider programmatic elements that provide defense-in-depth.
    - In and of itself, a COP credit does not eliminate or alter any programmatic element (i.e., containment leakage testing) that provides defense-in-depth.
    - Licensees and staff should consider possible synergistic effects that may arise when various programmatic elements are modified (perhaps through a series of LARs).

Application of the Five Key Principles to COP Credits (Con't.)

- Principle #3: Safety margins
  - Discussed elsewhere in the staff's presentation.
- Principle #4: Small changes in risk
  - Current estimates indicate that the change in internal events CDF due to a COP credit is less than 10^{-6}/y.
  - The staff's white paper describes how PRA elements should be modified to reflect a COP credit.
  - RG 1.174 allows the use of qualitative risk evaluations (e.g., seismic margins analysis).
  - The final acceptability of a proposed COP credit is based on consideration of current regulatory requirements and adherence to the five key principles, and <u>not solely</u> on a comparison of quantitative PRA results to the numerical risk acceptance guidelines.
- Principle #5: Performance measurement
  - The staff's white paper lists many performance measurement strategies relevant to COP credits.

Backup Viewgraphs

Acronyms and Initialisms

COP	containment overpressure
LAR	license amendment request
NRR	The Office of Nuclear Reactor Regulation
RG	Regulatory Guide
SRP	Standard Review Plan (NUREG-0800)

The Evolution of SRP Section 19.2, Appendix D

- 8/25/1997, COMSAJ-97-008: Discussion of compliance and safety; staff has the responsibility to consider risk during review of LARs.
- 4/12/1998: Union Electric submitted LAR to electrosleeve SG tubes at Calloway (not risk-informed); staff concerned about behavior of electrosleeve material during severe accidents.
- 12/23/1998, SECY-98-300: Options to risk-inform 10 CFR 50; staff identified policy issue to get clarification of its authority to apply risk-informed decisionmaking in areas beyond those associated with licensee-initiated risk-informed LARs.
- 5/24/1999: Staff approved Calloway electrosleeve LAR.
- 6/8/1999, SRM on SECY-98-300: Commission agreed that additional guidance was needed.
- 10/12/1999, SECY-99-246: Transmitted interim guidance on applying risk-informed decisionmaking in LARs.
- 1/5/2000, SRM on SECY-99-246: Commission approved interim guidance.
- 3/28/2000, RIS 00-007: Advised licensees about interim guidance on the use of risk information by the staff during its reviews of LARs.

The Evolution of SRP Section 19.2, Appendix D (Con't.)

- 4/10/2000: Draft appendix to SRP Chapter 19 published in the Federal Register.
- 5/11/2000: ACRS meeting on draft appendix to SRP Chapter 19.
- 5/16/2000: Public workshop on draft appendix to SRP Chapter 19.
- 5/30/2000: CRGR meeting on draft appendix to SRP Chapter 19.
- 9/26/2000: Staff forwarded the final appendix to SRP Chapter 19 to the Commission.
- 11/12/2000, COMSECY-00-0038: Commission approved final appendix to SRP, Chapter 19; directed the staff to notify the Commission of the first few LARs that create special circumstances.
- 1/18/2001, RIS 01-002: Advised licensees of final guidance on the use of risk information by the staff during its reviews of LARs.
- June 2007: Former SRP Chapter 19 redesignated as SRP Section 19.2; Appendix D retained without modification.

Use of Containment Accident Pressure in Determining Available NPSH of ECCS and Containment Heat Removal Pumps

R. Lobel, NRR/DSS

M. Stutzke, RES/DRA

December 4, 2008

PURPOSE

- To discuss the NRC staff position on the use of containment accident pressure in determining the available NPSH of ECCS and containment heat removal pumps

- Staff position and discussion provided to ACRS in a memorandum to the ACRS Executive Director, dated November 4, 2008

TOPICS

- INTRODUCTION
- REGULATORY BACKGROUND
- REGULATORY BASIS
- TECHNICAL BASIS
- RISK CONSIDERATIONS
- FUTURE ACTIONS
- CONCLUSIONS

INTRODUCTION-1

- Changes to November 4, 2008 position paper:
  - Position paper states that RG 1.1 will be withdrawn. It will not. (Executive Summary and Page 11)
  - For non-EPU submittals, risk procedure will follow SRP 19.2 Appendix D (Executive Summary and Page 28)
  - Discussion of uncertainty in NPSHR will be revised (Page 4)

INTRODUCTION-2

- ECCS AND CONTAINMENT HEAT REMOVAL PUMPS IN BWRs AND PWRs ARE CENTRIFUGAL PUMPS
  - Capable of operation over a wide range of flow rates and pressures
  - Operation well understood
  - Used in wide variety of applications
  - Subject to cavitation

REGULATORY BACKGROUND-1

- Regulations allow use of containment accident pressure in determining the available NPSH of safety related pumps

REGULATORY BACKGROUND-2

- RG 1.1 November 1970
- RG 1.82 Rev 0
  - 50% Blockage
- RG 1.82 Rev 1 November 1985
  - Incorporates findings of USI A-43
  - Uniform coverage of sump screens by loca generated debris
  - RG 1.1 cited as guidance for use of containment accident pressure

REGULATORY BACKGROUND-3

- RG 1.82 Revision 2 May 1996
  - Incorporates guidance supporting NRC Bulletin 96-03
- GL 97-04 October 1997
- RG 1.82 Revsion 3 November 2003
  - Incorporates guidance supporting NRC Bulletin 2003-01
- DRAFT RG 1.82 Revision 4
  - Revises guidance on calculating available NPSH
  - September 20, 2005 ACRS letter recommended revisions and further restrictions on use of containment accident pressure prior to issuing

NRC POSITION

- The NRC allows use of containment accident pressure in determining available NPSH in the following cases:
  - Analyses using conservative assumptions have demonstrated that this pressure will be available for postulated design basis accidents
  - When examined from a broader perspective (i.e., beyond design basis accidents), an acceptable level of safety is maintained

NRC POSITION-2

- Duration of use of containment accident pressure is not risk significant.
- Significant contributors to loss of containment integrity occur at start of postulated accident:
  - Pre-existing leak
  - Failure of containment isolation
  - Possible exception for App R fire (associated circits). Examined during staff reviews.

NRC POSITION-3

- The magnitude of pressure needed is not risk significant.
- A calculation of peak LOCA containment pressure demonstrates that the pressure is less than the design pressure.
- Pressure at the time of peak sump or suppression pool temperature is much less than containment design pressure.

NPSH MARGIN

- Some authorities specify a margin between NPSHR and NPSHA of 30% or more
- Nuclear industry practice is NPSHA = NPSHR
- This is acceptable because:
  - LOCA pressure is conservatively calculated
  - Margin is important to ensure continuous long-term pump operation. Not one time operation for period of hours.
  - Tests have shown that damage rate is highest at some point between 3% and incipient cavitation. (Pump dependent.)

TECHNICAL BASIS

- Considerations for acceptability of using containment accident pressure:
  - High confidence in containment integrity
  - Conservative calculations
  - Pump design
  - Emergency operating procedures
  - Minimal impact on plant risk

CONTAINMENT-1

- RG 1.1: One rationale for not using containment accident pressure is the possibility of “impaired containment integrity”
  - Structural integrity test prior to licensing
  - 10 CFR 50.54(o) and Appendix J require leak testing of containment and individual penetrations
- 10 CFR 50.55a requires periodic inspections of the containment.
- TS control containment integrity.
- Stringent plant procedures.
- Good experience

CONTAINMENT-2

- Majority of plants using containment accident pressure to determine available NPSH are BWRs with Mark I containments
  - inerted
  - O_2 monitors
  - Drywell-wetwell ΔP restricted by technical specifications

CONTAINMENT-3

- 4 plants subatmospheric. 3 more operate as sub- atmospheric
- 4 PWRs with large dry containments

CONTAINMENT-4

- Other safety analyses assume containment integrity:
  - Containment integrity is assumed in calculating offsite dose (10 CFR 50.67 or 10 CFR Part 100)
  - Accident pressure is assumed in calculating peak cladding temperature (10 CFR Appendix K)

EQ CONSIDERATIONS

- SRP 3.11 covers all items of equipment important to safety (mechanical, electrical, I&C)

- SRP 3.11:
  - *For mechanical equipment located in a harsh environment, compliance with the environmental design provisions of GDC 4 are generally achieved by demonstrating that the nonmetallic parts/components are suitable for the postulated design basis environment conditions.*

  - *For mechanical equipment, the staff concentrates its review on materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems and diaphragms*

CONSERVATISM-1

- Calculation for LOCA underestimates containment pressure and overestimates suppression pool or sump temperature
- Calculations for ATWS, station blackout and Appendix R fire are realistic
  - some conservatism is typically present
- NRC staff November 4, 2008, white paper provides lists of typical conservative assumptions used in BWR and PWR LOCA calculations.

PUMP DESIGN-1

- All pumps of interest share certain characteristics with respect to cavitation:
  - robust construction
  - mechanical seals
  - stainless steel (cavitation-resistant) impellers
- ECCS pumps of later plants have lower required NPSH than those used in earlier plants

PUMP DESIGN-2

- NRC staff has approved pump operation in cavitation below the required NPSH
- Based on testing and subsequent inspection of pumps
- These tests of prototypical pumps in cavitation have not shown damage or more than very minor wear (scratches)

PUMP DESIGN-3

SUMMMARY OF NUCLEAR POWER PLANT SAFETY RELATED PUMP CAVITATION TESTING

PLANT	PUMP	COMMENTS
Browns Ferry	RHR	●Tests performed at 8000 and 10000 gpm ● Severe audible cavitation but acceptable motor vibration ● Tests terminated before "breakout" point (complete loss of head) ● Discharge head drop 10-12% ● Manufacturer's NPSHR curves may be reduced an additional 9 ft ● Operated for 10 minutes below manufacturer's recommended design NPSH conditions
Browns Ferry	RHR and Core Spray	● Pump vendor provided curves showing acceptable operation for limited times at up to 6% head loss ● Based on total operation time of 8000 hrs at various NPSHR values.
Dresden	Core Spray	● Witness and NPSH testing. Pump disassembled and examined. All parts in excellent condition. ● Cavitation tested 4000 to 6000 gpm. Time not specified. Pump disassembled and examined. No damage or wear. ● Pump again tested below previous cavitation point for one hour. No damage or wear. ● Pump cavitation tested again for one hour. Suction pressure lowered and tested further for 30 minutes. Pump again disassembled and examined. No damage or wear. Several scratches.
Vermont Yankee		● Pump vendor provided curves showing acceptable operation for limited times at up to 6% head loss ● Based on total operation time of 8000 hrs at various NPSHR values.
Monticello	Core Spray	Cavitation test performed by pump vendor. Pump went through "extensive cavitation" for several hours "without visible damage to the impeller."
Beaver Valley (North Anna Unit 2 pump)	Recirculation Spray	Closed loop test. NPSHA lowered by water temperature increase and tank level decrease Initial NPSHA = 15.1 ft. NPSHA lowered to 5 ft (well into the breakdown region) for ½ hour. After testing pump Total Dynamic Head/Capacity curve regenerated. No degradation noted.
Crystal River	Building Spray	Pump vendor provided justification for a required NPSH based on a 5% head drop

OPERATIONAL CONSIDERATIONS-1

- BWR EOPs consider containment pressure in assessing adequate available NPSH
- BWR NPSH analyses consider operation of containment spray for the duration of the event

OPERATIONAL CONSIDERATIONS-3

- Operator indications of cavitation (from control room)
  - Erratic or decreasing pump motor current
  - Erratic flow or flow less than expected
  - Frequent adjustments to ECCS pump discharge valves to maintain constant flow rate (BWRs)
- Operator response to cavitation
  - throttle pump
  - remove pump from service
  - consider other water sources

EFFECT OF THROTTLING

Dresden 2/3 calculation DRE97-0002 Rev 0 Attachment A

RHR/CS	RHR Pump Flow/Pump (gpm)	Suction Loss (ft)	NPSHR (ft)	NPSH margin (ft)
4/2	5000	10.7	30	-11.1
4/2	3750	6.5	25.5	-0.7
4/2	2500	3.4	25	4.3

FUTURE ACTIONS

- Revise RG 1.82 Revision 3
  - Clarify and add more detail to NPSH discussion
  - Revise positions
  - Remove material not relevant to current status of the issue (e.g., sump design descriptions)
  - Update references
  - Revise RG 1.1 to state that RG 1.82 provides the current guidance.
- Revise white paper
- Make white paper publicly available.

CONCLUSIONS

- High confidence in containment integrity
- Prototypical pumps have been cavitation tested for periods up to several hours with no damage
- Need for credit for containment accident pressure for BWRs is limited to older plants
- Where examined, the risk of using containment accident pressure in determining available NPSH is negligible
- For some plants, reliance on containment accident pressure is a result of conservative analysis

BACKUP SLIDES

STATISTICAL APPROACH

- A statistical estimate of the uncertainty in the pressure needed for adequate NPSH is added to a realistic value
- BWROG has submitted NEDC-33347 for review and approval.
- Approach used in several other areas of reactor safety analysis:
  - Realistic LOCA
  - Departure from Nucleate Boiling Ratio (DNBR)
  - BWR Anticipated operational Occurrences (AOO's)

PENETRATION SEALS

- “Both Viton and EPR/EPDM O-rings appear undamaged when exposed directly to a steam environment with temperatures up to about 600 F at a pressure of 155 psia for 4 to 6 hours…
- “Silicon rubber O-rings appear undamaged up to 500 F at a pressure of 155 psia when exposed directly in a steam environment for about 4 hours…”

  8th SMIRT Conference 1985

  “Integrity of Containment Penetrations under Severe Accident Conditions,” C.V. Subramanian