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

August 25, 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 566th FULL COMMITTEE MEETING HELD ON OCTOBER 8-10, 2009 IN ROCKVILLE, MARYLAND

The minutes of the subject meeting were certified on November 17, 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

November 17, 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 566th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), October 8-10, 2009

I certify that based on my review of the minutes from the 566th 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	11/17/09	11/17/09

OFFICIAL RECORD COPY

CERTIFIED Date Certified: 11/17/2009

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- Closure of Steam Generator Action Plan Items 3.1k, 3.4, 3.5, 3.10, 3.11, and 3.12, dated October 22, 2009

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LETTER

Letter to R. W. Borchardt, Executive Director for Operations, NRC, from Mario V. Bonaca, Chairman, ACRS:

- NRC Staff's Safety Evaluation Report with Open Items Regarding the North Anna, Unit 3 Combined License Application, dated October 23, 2009

MEMORANDUM

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

- Withdrawal of Regulatory Guide 1.56, dated October 14, 2009

MINUTES OF THE 566th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

ROCKVILLE, MARYLAND

The 566th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on October 8-10, 2009. Notice of this meeting was published in the *Federal Register* on August 25, 2009 (72 FR42946-42947) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting agenda (Appendix II). The meeting was open to public attendance. See Appendix III for a listing of attendees.

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. Mario Bonaca (Chairman), Dr. Said Abdel-Khalik (Vice-Chairman), Mr. J. Sam Armijo (Member-at-Large), Dr. George E. Apostolakis, 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, Dr. William Shack, 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. Mario Bonaca, 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. Bonaca 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.

II. Combined License (COL) Application for North Anna, Unit 3, Economic Simplified Boiling Water Reactor (ESBWR), and the Draft Safety Evaluation Report (SER) with Open Items

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

The Committee met with representatives of the NRC staff and the applicant, Dominion Virginia Power (Dominion), to discuss the COL application for North Anna, Unit 3, and the associated NRC staff's draft Safety Evaluation Report (SER) with Open Items. Some of the topics discussed were: hydrologic engineering; geology; qualification and In-Service Testing (IST) Programs; plant water system, and Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC).

Dominion representatives provided an overview of the North Anna, Unit 3, COL activities. Dominion discussed the responses to selected requests for additional information (RAIs) issued by the staff. In particular, Dominion discussed RAIs related to the function of the makeup water system; during shutdown/refueling mode, the plant service water system, and the circulating water system.

The staff provided the status of the resolution of the open items in the draft SER, especially the open issues related to the buried fiberglass piping used in the plant service water system and the proposed resolution to this issue. The staff also discussed the remaining open items on stability of subsurface materials and foundations, and safety-related mechanical equipment qualification. In closing, the staff stated that there were a few open items related to ITAAC and the Design Acceptance Criteria (DAC) that will be treated in a generic manner in the future.

The Committee issued a letter to the Executive Director for Operations on this matter, dated October 23, 2009, recommending that the staff proceed with the development of the final SER after resolving all open items. The Committee stated that at this time it has not identified any significant issues regarding the North Anna COL application.

III. License Renewal Application and Final SER for the Susquehanna Steam Electric Station, Units 1 and 2

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

The Committee met with representatives of the NRC staff and PPL Susquehanna, LLC (PPL or the applicant) to discuss the final SER related to the license renewal application for the Susquehanna Steam Electric Station, Units 1 and 2.

PPL discussed its general site information, operating experience, corrective actions, and commitment tracking process. PPL made 60 regulatory commitments for its license renewal program. The applicant also discussed four follow-up items resulting from the ACRS interim review: condition of underground medium voltage cables, station blackout recovery scoping, condition of containment, and aging of main steam line flow restrictors. PPL stated that it has

established adequate aging management programs for the period of extended operation. The NRC staff provided an overview of its review results, documented in the SER, and regional inspection results. The staff provided additional information regarding the applicant's plant-specific Boral operating experience, the Boral Coupon Testing Program, and the associated aging management program. The staff concluded that the applicant's Water Chemistry Program and Boral Coupon Testing Program are adequate to manage the aging effects of Boral used in the spent fuel pool racks. The staff also provided a brief discussion on the applicant's Metal Fatigue Monitoring Program and the effects of reactor coolant environment on the fatigue life of components and piping. The Susquehanna final SER contained no open items. Based on its review, the staff concluded that the requirements of 10 CFR 54.29(a) have been met.

The Committee issued a report to the NRC Chairman on this matter, dated October 23, 2009, recommending that the PPL application for renewal of the operating licenses of Susquehanna Steam Electric Station, Units 1 and 2, be approved.

IV. Resolution of Several Steam Generator Action Plan (SGAP) Items

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

The Committee met with representatives of the NRC staff to discuss the resolution of the remaining task items in the SGAP. In particular, the staff discussed the closure of task items 3.1k, 3.4, 3.5, 3.10, and 3.12. The SGAP task items were intended to develop a better understanding of reactor coolant system conditions and corresponding component behavior under severe accidents. In addition, the items were expected to define the risk associated with severe accident induced steam generator tube ruptures leading to containment bypass. The presentation topics were from the following technical areas of research: thermal-hydraulics (computational fluid dynamics methods), steam generator tube material failures, reactor coolant system material failures, component behavior studies, and probabilistic risk assessment (PRA). The thermal-hydraulics analysis takes the PRA sequence being evaluated and determines the fluid temperatures and pressures as a function of time. These conditions are then used as inputs to the reactor coolant system material failure and component behavior models. The thermal-hydraulic results and material failure information are combined into a PRA model to determine the risk associated with the consequential steam generator tube rupture. The methods and models for evaluating reactor components under severe accident conditions were discussed. The staff stated that the remaining SGAP items can be closed and that the remaining issues associated with consequential steam generator tube failure risk no longer require the level of coordination and agency focus required to implement the action plan process.

The Committee issued a letter to the Chairman on this matter, dated October 22, 2009, recommending that the staff proceed with closure of the remaining SGAP items.

V. 3-Dimensional Finite Element Analysis of the Oyster Creek Drywell Shell

[NOTE: Mr. Peter Wen was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff and Exelon Nuclear Generation Company (Exelon) to discuss the 3-dimensional (3-D) finite element analysis (FEA) of the

Oyster Creek Nuclear Generating Station (Oyster Creek) drywell shell. During the Oyster Creek license renewal process, Exelon committed to perform a 3-D FEA of its drywell shell prior to entering the period of extended operation. By letter dated January 22, 2009, Exelon submitted the results of the Oyster Creek drywell shell analysis.

Exelon provided background information, described the drywell, discussed drywell thickness measurements, and summarized the analyses. Exelon explained why the ultrasonic thickness internal grid locations are representative of the general area average thickness. Exelon discussed specifically the modeling of the vent header boundary conditions, and explained the reasons why the vent pipes/header will not buckle before the shell. Exelon also discussed its finite element model, sensitivity studies, and overall results.

The staff presented its overview of the 3-D FEA and regional inspection results. The staff concluded that the Oyster Creek 3-D FEA was performed utilizing widely accepted engineering practices consistent with ASME Code, good engineering judgment, and applied conservatively biased realistic assumptions. The staff also concluded that the evaluations in all cases (baseline and sensitivity cases) confirm the Oyster Creek drywell shell complies with the ASME Code limits, and provide reasonable and realistic quantification of the available safety margin of the drywell shell for the postulated loading conditions.

The Committee issued a report to the NRC Chairman on this matter, dated October 16, 2009, stating that the analysis presented by Exelon fulfills its commitment to provide a modern, realistic, 3-D FEA that better quantifies the available safety margin for the current Oyster Creek drywell shell configuration. The Committee agreed with the staff's conclusion that this analysis was performed using good engineering practices and judgment and used conservatively biased realistic assumptions.

VI. Draft Final Revision 2 to Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants"

[NOTE: Mrs. Kathy Weaver was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff to review draft final Regulatory Guide (RG) 1.189, Revision 2, and "Fire Protection for Nuclear Power Plants." The staff described the changes in RG 1.189 including discussions of safe shutdown success path components and components important to safety, the use of operator manual actions and fire modeling for assessing components important to safe shutdown, and examples of safe shutdown success path components and components important to safe shutdown.

The staff also discussed the previously unresolved issues in relation to the Nuclear Energy Institute document (NEI), NEI 00-01, Revision 2, "Guidance for Post Fire Safe Shutdown Circuit Analysis," and subsequent staff resolution of these issues. In addition, the staff discussed the industry comment that NEI 00-01 should be referenced in the Guide.

The Committee issued a report to the NRC Chairman on this matter, dated October 20, 2009, recommending that Regulatory Guide 1.189, Revision 2 be issued as final.

VII. 10 CFR Part 52 Regulatory Process

[NOTE: Mr. Girija Shukla was the Designated Federal Official for this portion of the meeting.]

Representatives of the NRC staff provided an overview of the new reactor licensing process under 10 CFR Part 52, "Early Site Permits, Standard Design Certifications, and Combined Licenses for Nuclear Power Plants." The staff discussed requirements for licensing such as applicant qualifications, design acceptability, environmental impacts, operational programs, site safety, and verification of the closure of ITAAC and DAC. The staff also discussed the regulatory process related to Early Site Permit (ESP), Design Certification, COL, Standard Design Approval, Manufacturing License, and Limited Work Authorization (LWA). This was an information briefing. No Committee action was necessary.

VIII. Subcommittee Report

The Chairman of the AP1000 Subcommittee provided a report regarding the matters discussed at the October 6-7, 2009, Subcommittee meeting. In that meeting, the Subcommittee was briefed by representatives of the NRC staff and the Westinghouse Electrical Company on three Chapters of the amended AP1000 Design Control Document (DCD) as well as the corresponding Chapters of the draft SER with open items. The specific Chapters were Chapter 3 (Design of Structures, Components, Equipment, and Systems), Chapter 8 (Electric Power), and Chapter 18 (Human Factors Engineering). The Standard Review Plan Sections 3.7 and 3.8 pertaining to the treatment of seismic issues and the design of containment structures were not discussed, and have been deferred to a later Subcommittee meeting. It was noted that these Chapters contain about 30 open items pertaining to the staff's AP1000 design review. The Subcommittee plans to continue its review of the AP1000 DCD amendments during future meetings.

IX. Draft ACRS Report on the NRC Safety Research Program

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

The Committee discussed the draft 2010 ACRS biennial report to the Commission on its review and evaluation of the NRC Safety Research Program. The Committee plans to continue the discussions on its draft report on review and evaluation of the NRC Safety Research Program during its November 5-7, 2009 meeting.

X. Executive Session

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

A. Reconciliation of ACRS Comments and Recommendations/EDO Commitment

- The Committee considered the EDO's response of September 22, 2009, to comments and recommendations included in the July 24, 2009 ACRS letter concerning draft final Regulatory Guide 1.215, "Guidance for ITAAC Closure under 10 CFR Part 52." The Committee was partially satisfied with the EDO's response.

  The staff has argued that using the design certification process, and post-licensing inspections and reviews, will be adequate to support DAC closure. Since detailed design information may not be available during the design certification process, the ACRS remains concerned that the DAC closure process is not satisfactorily defined to ensure adequacy of the design. ACRS and NRO have initiated a dialogue focused on this subject. The ACRS is looking forward to ongoing engagement on this subject with NRO as the staff continues to work out the details for ITAAC/DAC closure.

- The Committee considered the EDO's response of August 25, 2009, to comments and recommendations included in the July 27, 2009 ACRS letter concerning draft Final Revision 3 to Regulatory Guide (RG) 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants." The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of August 27, 2009, to comments and recommendations included in the July 22, 2009 ACRS letter concerning draft Template NEI-08-08, "Generic FSAR Template Guidance for Life Cycle Minimization of Contamination," and Draft DC/COL-Interim Staff Guidance -06. 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 October 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 February 2010 was 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

Webstreaming of the ACRS Meetings

During its April and May 2009 meetings, the Committee discussed the March 6, 2009 Staff Requirements Memorandum (SRM) in which the Commission stated that:

If the ACRS decides to pursue Webstreaming of the ACRS Meetings, the ACRS should prepare a proposed plan reflecting their interest, in coordination with the Office of Administration.

During the May meeting, the Committee established a Panel to assess the pros and cons of participating in the Webstreaming Program and to provide recommendations for use by the Committee in making a decision. The Panel consists of:

Dr. Corradini, Chairman
Dr. Armijo
Dr. Banerjee
Mr. Ray
Mr. Stetkar.

ACRS Meeting With the Commission

At the September meeting, we informed the Committee of our intent to request the Commission to postpone the ACRS meeting with the Commission scheduled for December 4, 2009 to April 9, 2010. Accordingly, subsequent to the September ACRS meeting, we requested that the Commission postpone the meeting with the ACRS to April 2010. This request was made in view of the heavy workload at the December meeting, and postponing some items to the February 2010 meeting to accommodate the Commission meeting may have significant impact on the staff schedule. However, the Commission decided to meet with the ACRS between 9:30 and 11:30 a.m. on Friday, December 4, 2009. We plan to request that the meeting be held between 1:30 and 3:30 p.m. so as to accommodate scheduling some presentations in the morning of December 4, 2009.

A proposed list of topics for meeting with the Commission is as follows:

- Overview (Bonaca)

Major Accomplishments
Future Plan Design Activities
Containment Accident Pressure Issue
Major Areas of Ongoing and Future ACRS Activities

- Inspection, Tests, Analyses, and Acceptance Criteria (ITAAC)/Design Acceptance Criteria (DAC) Closure Process (Bley)

- Amendment to the AP1000 Design Control Document (Ray)

- Three-Dimensional Analysis of the Oyster Creek Drywell Shell (Shack)

- Beaver Valley Containment Liner Corrosion (Armijo)

Dr. Armijo commented that since the Beaver Valley containment liner corrosion is one of many issues reviewed by the Committee prior to recommending approval of the license extension, there is no need for a separate presentation on this issue, and it could be covered under the Overview.

As far as Item 3, AP1000, is concerned, the Committee has not written a letter on the amendment to the AP1000 Design Control Document. A letter is scheduled to be completed at the November ACRS meeting. Without a documented Committee position on this matter, this should not be included as a line item in the Agenda.

Mini Retreat

A mini retreat is scheduled for Saturday, November 7, 2009 to discuss the ACRS process for reviewing amendments to the Design Control Documents (DCD) related to the previously certified designs. Some of the issues for discussion include:

- Is it effective to apply the same process (chapter-by-chapter review) being used in reviewing the new design certification applications (e.g., ESBWR, EPR, US APWR) to the review of amendments to the DCDs associated with previously certified designs (e.g., AP1000, ABWR)?

- Does the Committee add value by reviewing all changes and is it the efficient use of Committee's time?

- Should the Committee review only the major design changes and other safety-significant changes? Should the Committee ask the staff to provide a list of such changes along with the staff's evaluation?

Travel Issue

The ACRS Travel team does its best to ensure all paper work is completed in a timely manner and airline tickets are issued by Carlson Travel. Under unusual circumstances, if the members do not have their tickets issued by Carlson, they are requested to purchase the ticket with the Government TRAVEL CARD.

Interim Staff Guidance and Regulatory Guide

a) Interim Staff Guidance

The staff issued the following Interim Staff Guidance (ISG) and would like to know whether the Committee wants to review this ISG.

ISG 09-016, "Compliance with 10 CFR 50.54(hh)(2) and 10 CFR 52.80(d) Loss of Large Areas of the Plant Due to Explosions or Fires from a Beyond-Design Basis Event."

The staff prepared ISG09-016 to provide guidance to new reactors regarding implementation of the new rule in 50.54(hh)(2) that implements Section B.5.b of the post 9/11 order to existing reactors.

Based on his review of ISG09-016, Dr. Bonaca recommends that the Committee review this ISG.

b) Draft Regulatory Guide

The staff plans to issue the following Draft Final Regulatory Guide and would like to know whether the Committee wants to review this Guide prior to being issued final.

Proposed Revision 1 to Regulatory Guide 1.62 (DG-1190), “Manual Initiation of Protective Actions”

Regulatory Guide 1.62 is a proposed Revision 1 that was issued as draft Regulatory Guide (DG-1190) on December 23, 2008 for public comment. The comment period ended February 20, 2009. The major changes to this draft final guide are: (1) to update the IEEE Standard 279-1971 to the IEEE Standard 603-1991, and (2) to expand the scope of manual actions covered by this Guide.

Based on his review of Regulatory Guide 1.62, Mr. Brown recommends that the Committee not review this Guide.

c) Withdrawal of Regulatory Guide

Regulatory Guide 1.56, “Maintenance of Water Purity in Boiling Water Reactors”

Regulatory Guide (RG) 1.56 was issued for comment in July 1978 and never finalized. RG 1.56 was intended to support Title 10, Part 50, of the *Code of Federal Regulations*, “Domestic Licensing of Production and Utilization Facilities,” Appendix A, “General Design Criteria for Nuclear Power Plants,” General Design Criterion (GDC) 14, “Reactor Coolant Pressure Boundary,” and GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary.” RG 1.56 describes an acceptable method for maintaining water purity levels in the reactor coolant in order to ensure that degradation of the reactor coolant pressure boundary is not exacerbated by poor chemistry conditions. The staff considers water chemistry to be an operational issue for plants. It is in the licensee’s best interest to operate the plant with a chemistry regime that optimizes component performance. There is adequate industry-generated guidance available for licensees to develop a plant-specific water chemistry program. The industry routinely updates this guidance to incorporate the latest knowledge and lessons learned in the area of water chemistry.

Based on his review of the proposed withdrawal of this Guide, Dr. Armijo recommends the Committee to agree with the staff’s proposal to withdraw this Guide.

Christmas Party

Each year the Committee sponsors a Christmas Party for the ACRS Office staff during the December meeting. The Committee should decide whatever it wants to keep up with the tradition this year.

The meeting was adjourned at 7:00 p.m. on October 9, 2009.

members who collect information from the public requesting reproduction of publicly available documents in NRC Headquarters' Public Document Room. Copies of the form are utilized by the reproduction contractor to accompany the orders. One copy of the form is kept by the contractor for their records, one copy is sent to the public requesting the documents, and the third copy (with no credit card data) is kept by the PDR staff for 90 calendar days, and then securely discarded.

Submit, by November 24, 2009, comments that address the following questions:

1. Is the proposed collection of information necessary for the NRC to properly perform its functions? Does the information have practical utility?
2. Is the burden estimate accurate?
3. Is there a way to enhance the quality, utility, and clarity of the information to be collected?
4. How can the burden of the information collection be minimized, including the use of automated collection techniques or other forms of information technology?

A copy of the draft supporting statement may be viewed free of charge at the NRC Public Document Room, One White Flint North, 11555 Rockville Pike, Room O-1 F21, Rockville, MD 20852. OMB clearance requests are available at the NRC worldwide Web site: *http://www.nrc.gov/public-involve/doc-comment/omb/index.html.* The document will be available on the NRC home page site for 60 days after the signature date of this notice. Comments submitted in writing or in electronic form will be made available for public inspection. Because your comments will not be edited to remove any identifying or contact information, the NRC cautions you against including any information in your submission that you do not want to be publicly disclosed. Comments submitted should reference Docket No. NRC–2009–0395. You may submit your comments by any of the following methods. Electronic comments: Go to *http://www.regulations.gov* and search for Docket No. NRC–2009–0395. Mail comments to Acting NRC Clearance Officer, Tremaine Donnell (T–5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Questions about the information collection requirements may be directed to the Acting NRC Clearance Officer, Tremaine Donnell (T–5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, by telephone at 301–415–6258, or by e-mail to *INFOCOLLECTS.Resource@NRC.GOV.*

Dated at Rockville, Maryland, this 17th day of September 2009.

For the Nuclear Regulatory Commission,

Tremaine Donnell,

NRC Clearance Officer, Office of Information Services.

[FR Doc. E9–23203 Filed 9–24–09; 8:45 am]

BILLING CODE 7590–01–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 October 8–10, 2009, 11555 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Monday, October 6, 2008, (73 FR 58268–58269).

Thursday, October 8, 2009, Commissioners' Conference Room O–1f16, One White Flint North, Rockville, Maryland

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

8:35 a.m.–10:30 a.m.: Combined License Application for North Anna, Unit 3, Economic Simplified Boiling Water Reactor (ESBWR) and the Draft Safety Evaluation Report (SER) with Open Items (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Dominion Virginia Power regarding the North Anna ESBWR Combined License Application, the draft SER with Open Items, and related matters.

10:45 a.m.–12:15 p.m.: License Renewal Application and Final SER for the Susquehanna Steam Electric Station, Units 1 and 2 (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and PPL Susquehanna, LLC regarding the license renewal application for the Susquehanna Steam Electric Station, Units 1 and 2, the associated NRC staff's final SER, and related matters.

1:15 p.m.–3:15 p.m.: Steam Generator Action Plan (SGAP) Task 3.5, "A Risk Assessment of Consequential Steam Generator Tube Ruptures," and Other SGAP Items (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding SGAP Task 3.5, other SGAP items, and related matters.

3:30 p.m.–5 p.m.: Oyster Creek 3-Dimensional Structural Analysis of the Drywell Shell (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Exelon Generation Company, LLC, regarding the Oyster Creek 3-dimensional structural analysis of the drywell shell and related matters.

Conference Room T8–A1, Two White Flint North, Rockville, Maryland

5:15 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters discussed during this meeting.

Friday, October 9, 2009, Commissioners' Conference Room O–1F16, One White Flint North, Rockville, Maryland

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

8:35 a.m.–9:45 a.m.: Draft Final Revision 2 to Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft final Revision 2 to Regulatory Guide 1.189, NRC staff's resolution of public comments, and related matters.

10 a.m.–11 a.m.: 10 CFR Part 52 Regulatory Process (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the 10 CFR part 52 regulatory process and related matters.

11 a.m.–11:15 a.m.: Subcommittee Reports (Open)—The Committee will hear a report by and hold discussions with the Chairman of the AP1000 Subcommittee regarding selected Chapters of the draft SER with Open Items related to changes to the AP1000 Design Control Document that were discussed during the meeting on October 6–7, 2009.

12:15 p.m.–1 p.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed)—The Committee will discuss the recommendations of the Planning and Procedures

Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings, and 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.]

1 p.m.–1:15 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.

1:30 p.m.–3:30 p.m.: Draft ACRS Report on the NRC Safety Research Program (Open)—The Committee will discuss a draft ACRS report on the NRC Safety Research Program.

Conference Room T8–A1, Two White Flint North, Rockville, Maryland

3:45 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters discussed during this meeting.

Saturday, October 10, 2009, Conference Room T8–A1 Two White Flint North, Rockville, Maryland

8:30 a.m.–1:30 p.m.: Preparation of ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

1:30 p.m.–2 p.m.: Miscellaneous (Open)—The Committee will continue its discussion related to the conduct of Committee activities and specific issues that were not completed during previous meetings.

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. Thirty-five hard copies of each presentation or handout should be provided to the Designated Federal Official 30 minutes before the meeting. In addition, one electronic copy of each presentation should be emailed to the Designated Federal Official one day before meeting. If an electronic copy cannot be provided within this timeframe, presenters should provide the Designated Federal Official with a CD containing each presentation at least 30 minutes before the meeting. 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 a portion of this meeting noted above to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which constitute a clearly unwarranted invasion of personal privacy pursuant to 5 U.S.C. 552b(c)(2) and (6).

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.resource@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: September 21, 2009.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E9–23194 Filed 9–24–09; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[NRC–2009–0417]

Solicitation of Public Comments on the Implementation of the Reactor Oversight Process

AGENCY: Nuclear Regulatory Commission.

ACTION: Request for public comment.

SUMMARY: The NRC is soliciting comments from members of the public, licensees, and interest groups related to the implementation of the Reactor Oversight Process (ROP). An electronic version of the survey questions and additional information about the ROP are available at *http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/index.html*. This solicitation will provide insights into the self-assessment process and a summary of the feedback will be included in the annual ROP self-assessment report to the Commission.

DATES: The comment period expires on November 6, 2009. The NRC will consider comments received after this date if it is practical to do so, but is able to ensure consideration of only those comments received on or before this date.

ADDRESSES: You may submit completed questionnaires and/or comments by any one of the following methods. Please include Docket ID NRC–2009–0417 in the subject line of your comments. Comments submitted in writing or in electronic form will be posted on the NRC Web site and on the Federal rulemaking Web site Regulations.gov. Because your comments will not be edited to remove any identifying or contact information, the NRC cautions you against including any information in your submission that you do not want to be publicly disclosed.

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

September 21, 2009

AGENDA
566th ACRS MEETING
OCTOBER 8-10, 2009

THURSDAY, OCTOBER 8, 2009, COMMISSIONERS' CONFERENCE ROOM O-1F16, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 – 10:30 A.M. Combined License Application for North Anna, Unit 3, Economic Simplified Boiling Water Reactor (ESBWR) and the Draft Safety Evaluation Report (SER) with Open Items (Open) (MLC/CLB)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and Dominion Virginia Power regarding the North Anna ESBWR Combined License Application, the draft SER with Open Items, and related matters.

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

10:30 – 10:45 A.M. ***** BREAK *****

3) 10:45 – 12:15 P.M. License Renewal Application and Final SER for the Susquehanna Steam Electric Station, Units 1 and 2 (Open) (WJS/PW)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff and PPL Susquehanna, LLC regarding the license renewal application for the Susquehanna Steam Electric Station, Units 1 and 2, the associated NRC staff's final SER, and related matters.

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

12:15 – 1:15 P.M. ***** LUNCH *****

4) 1:15 – 3:15 P.M. Steam Generator Action Plan (SGAP) Task 3.5, "A Risk Assessment of Consequential Steam Generator Tube Ruptures," and other SGAP items (Open) (DAP/CLB)

4.1) Remarks by the Subcommittee Chairman

4.2) Briefing by and discussions with representatives of the NRC staff regarding SGAP Task 3.5, other SGAP items, and related matters.

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

3:15 – 3:30 P.M. ***** BREAK *****

5) 3:30 – 5:00 P.M. Oyster Creek 3-Dimensional Structural Analysis of the Drywell Shell (Open) (WJS/PW)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff and Exelon Generation Company, LLC, regarding the Oyster Creek 3-dimensional structural analysis of the drywell shell and related matters.

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

5:00 – 5:15 P.M. ***** BREAK *****

CONFERENCE ROOM T-8A1, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

6) 5:15 – 7:00 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

6.1) Combined License Application for North Anna, Unit 3, ESBWR and the Draft SER with Open Items (MLC/CLB)

6.2) License Renewal Application and Final SER for the Susquehanna Steam Electric Station, Units 1 and 2 (WJS/PW)

6.3) Steam Generator Action Plan (SGAP) Task 3.5, "A Risk Assessment of Consequential Steam Generator Tube Ruptures," and other SGAP items (DAP/CLB)

6.4) Oyster Creek 3-Dimensional Structural Analysis of the Drywell Shell (WJS/PW)

FRIDAY, OCTOBER 9, 2009, COMMISSIONERS' CONFERENCE ROOM O-1F16, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

8) 8:35 – 9:45 A.M. Draft Final Revision 2 to Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants" (Open) (JDS/KDW)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final Revision 2 to Regulatory Guide 1.189, NRC staff's resolution of public comments, and related matters.

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

9:45 – 10:00 A.M. ***** BREAK *****

9) 10:00 – 11:00 A.M. 10 CFR Part 52 Regulatory Process (Open) (DCB/GSS)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussions with representatives of the NRC staff regarding the 10 CFR Part 52 regulatory process and related matters.

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

10) 11:00 – 11:15 A.M. Subcommittee Report (Open)

Report by and discussions with the Chairman of the AP1000 Subcommittee regarding selected Chapters of the draft SER with open items related to changes to the AP1000 Design Control Document that were discussed during the meeting on October 6-7, 2009. (HBR/MPL)

11:15 – 12:15 P.M. ***** LUNCH *****

11) 12:15 – 1:00 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed) (MVB/EMH)

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

11.2) 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]

12) 1:00 – 1:15 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.

1:15 – 1:30 P.M. ***** BREAK *****

13) 1:30 – 3:30 P.M Draft ACRS Report on the NRC Safety Research Program (Open) (DAP, et al/HPN, et al)
13.1) Remarks by the Subcommittee Chairman
13.2) Discussion of a draft ACRS report on the NRC Safety Research Program

3:30 – 3:45 P.M. ***** BREAK *****

CONFERENCE ROOM T-8A1, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 3:45 – 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
14.1) Combined License Application for North Anna, Unit 3, ESBWR and the Draft SER with Open Items (MLC/CLB)
14.2) License Renewal Application and Final SER for the Susquehanna Steam Electric Station Units 1 and 2 (WJS/PW)
14.3) Steam Generator Action Plan (SGAP) Task 3.5, "A Risk Assessment of Consequential Steam Generator Tube Ruptures," and other SGAP items (DAP/CLB)
14.4) Oyster Creek 3-Dimensional Structural Analysis of the Drywell Shell (WJS/PW)
14.5) Draft Final Revision 2 to Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants" (JDS/KDW)

SATURDAY, OCTOBER 10, 2009, CONFERENCE ROOM T-8A1, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

16) 1:30 – 2:00 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.

- One (1) electronic copy of each presentation should be emailed to the Designated Federal Official 1 day before the meeting. If an electronic copy cannot be provided within this timeframe, presenters should provide the Designated Federal Official with a CD containing each presentation at least 30 minutes before the meeting.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
565th FULL COMMITTEE MEETING

October 8-10, 2009

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TODAY'S DATE: October 8, 2009

	NAME	NRC ORGANIZATION
1	Janelle Jessie	NRO/DNRL
2	Mark McBride	NRO/DSER/RHEB
3	JERRY WILSON	NRO / NRGA
4	John Lai	ACRS/SPBA
5	FRANCIS AKSTULEWICZ	NRO/DNRL
6	Tom Kevern	NRO/DNRL
7	Brad Harvey	NRO/DSER/RSAC
8	Marcia Carpenter	OGC
9	Seshagiri Rao Tammara	NRO/DSER/RSAC
10	Mark Tonacci	NRO/DNRL/NGEI
11	Bruce Musico	NRC/NSIR
12	WEIJUN WANG	NRO/DSER
13	Thomas Scarbrough	NRO/DE/CIB2
14	LARRY WHITNEY	NRO/DSRA/SBPB
15	John Segala	NRO/DSRA/SBPA
16	Aida Rivera-Varona	NRO/DUP/CQVB
17	A.H. HSIA	NRO/DSER
18	David Terao	NRO/DE/CIB1
19	Neil Ray	NRO/DE/CIB2
20	ERIC OESTERLE	NRO/DNRL/NRGA
21	George GEORGIEV	NRO/DE/NRO
22	Eduardo Sastre	NRO/DE/CIB2
23	Matthew McConnell	NRO/DE/EEEB
24	CHRISTOPHER COOK	NRO/DSER/RHEB
25	DUC NGUYEN	NRR/DLR
26	E. Gettys	NRR/DLR
27	CLIFF DOVIT	NRR/DLR
28	M. Banic	NRR/DLR

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
565th FULL COMMITTEE MEETING

October 8-10, 2009

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TODAY'S DATE: October 8, 2009

	NAME	NRC ORGANIZATION
1	Bill Rogers	NRR/DLR
2	Seung Min	NRR/DLR
3	David Alley	NRR/DLR
4	JAY ROBINSON	NRR/DLR
5	Glenn Meyer	Region I/DRS
6	JAMES MEDOFF	NRR/DLR
7	Brian Harris	NRR/DLR
8	SAMSON LEE	NRR/DLR
9	Robert Sun	NRR/DLR
10	Emma Wong	NRR/DCI
11	Bart Fu	/DLR
12	Bryce Lehman	NRR/DLR
13	Xi Gan	NRR/DLR
14	Andrew Primaris	NRR/DLR
15	Albert Wong	NRR/DLR
16	Chuang-yeh Yang	NRR/DLR
17	BARRY ELLIOT	NRR/DLR
18	Rachel Vaughn	NRR/DLR
19	Zuhan Xi	NRO/DSER
20	Simon Sheng	NRR/DCI
21	R. Auluck	NRR/DLR
22	C. Boyd	RES/DSA
23	Selim Sancaktar	RES/DRA
24	BOB PALLA	NRR/DRA
25	Jason Schaperow	RES/DSA
26	Tim McGinty	NRR/DPR
27	Chaks Tapp	RES/DE
28	STUART RICHARDS	RES/DE

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
565th FULL COMMITTEE MEETING

October 8-10, 2009

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TODAY'S DATE: October 8, 2008

NAME	NRC ORGANIZATION
Ken Karwoski	NRR/DCI
Hans Asher	NRR/DLR
Allen Hiser	NRR/DLR
SAMSON LEE	NRR/DLR
John Munninger	NRC
TIM OHARA	NRC REGION I
KAMAL MANOLY	NRR/DE
Louise Lund	NRR/DLR

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
566th FULL COMMITTEE MEETING

October 8-10, 2009

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TODAY'S DATE: October 9, 2009

	NAME	NRC ORGANIZATION
1	Jaclyn Storch	OIG
2	Charles Moulton	NRR/DRA/AFPB
3	JERRY WILSON	NRO / NRGA
4	ERIC OESTERLE	NRO/DNRL / NRGA
5	Alex Klein	NRR/DRA/AFPB
6	Daniel Frumkin	NRR/DRA/AFPB
7	John Tappert	NRO/DCIP
8	Phil Reed	RES/DRA
9	Nonette Gilles	NRO/DNRL/NRGA
10	LAURA DUDES	NRO/DE
11	Andy Campbell	NRO/DCIP
12	TRAVIS CHAPMAN	NRO/DCIP/CFSB
13	DENISE McGOVERN	NRO/DCIP/CFSB-ITAAC
14	MARK KOWAL	NRO/DCIP/CFSB
15	Mike Junge	NRO/DCIP/COLA
16	DON HABIB	NRO/DNRL/NWE1
17	Eileen McKenna	NRO/DNRL/NWE2
18	SALMAN HAQ	HR/HRTD/RFT
19	PATRICK SEKERAK	NRO/DE/EMB
20	K Robert HSU	NRO/DE/EMB1
21	Ann Hodgdon	OGC
22	EARL R. LIBBY	NRO/DNRL/NRGA
23	Jocelyn Mitchell	RES
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
566th FULL COMMITTEE MEETING

October 8-10, 2009

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TODAY'S DATE: October 9, 2009

	NAME	AFFILIATION
1	Richard Baker	Bechtel
2	Lynn Van-Derpool	Bechtel
3	Douglas Kemp	Bechtel
4	Rick Kingston	GE-Hitachi
5	MARVIN SMITH	DOMINION
6	Patricia Campbell	GEH
7	GEOFFREY QUINN	BECHTEL
8	JOE HEGNER	DOMINION
9	Tony Banks	Dominion
10	JIN CHUNG	MNES
11	JOHN KRAUS	SSES
12	ROSS GARDNER	SSES PPL
13	James K. Williams	PPL SSES
14	Christopher Hoffman	PPL SSES
15	Bruce Swoyer	PPL SSES
16	Dale Roth	PPL Susquehanna
17	Duane Filchner	PPL Susquehanna
18	Dave Flyte	PPL Susquehanna
19	Jeffrey Weik	PPL Susquehanna
20	Susan Miller	PPL Susquehanna
21	Philip Brady	PPL SUSQUEHANNA
22	Daniel Miller	PPL Susquehanna
23	Richard D Pagodin	PPL Susquehanna
24	Gary L. Stevens	Structural Integrity Associates
25	JAMES STAVELY	PSEG
26	MASSOUD TAFAZZOLI	AREVA
27	Mitch McFarland	AREVA
28	Jim DAVIS	Self

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
566th FULL COMMITTEE MEETING

October 8-10, 2009

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TODAY'S DATE: October 9, 2009

	NAME	AFFILIATION
1	Erach Patel	ATL
2	Wilbur (Bob) Jackson	ATL
3	MIKE Gallagher	EXELON
4	David Benson	Exelon
5	S.S. (Stan) Tang	Structural Integrity
6	JOHN O'ROURKE	EXELON
7	PETER TAMBURRO	EXELON
8	JOHN HUFNAGEL	EXELON
9	Howie RAY	Exelon
10	CLARENCE MILLER	EXELON
11	Don FONTUN	Exelon
12	Ray Kuyler	Morgan Lewis
13	A.E KADAK	MIT.
14	Tim Lupold	RES/DE/CMB
15	Symm Trajkovska	SAT Consultants
16	Brian Harris	NRL/OGC
17	Nancy Chapman	SERCH/Bechtel
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
566th FULL COMMITTEE MEETING

October 8-10, 2009

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TODAY'S DATE: October 9, 2009

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~~DENISE McGOVERN~~	
~~TRAVIS CHAPMAN~~	
G BORSH	DOMINION
JIN CHUNG	MNES

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

October 15, 2009

AGENDA
567th ACRS MEETING
NOVEMBER 5-7, 2009

THURSDAY, NOVEMBER 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/EMH/MPL)
   1.1) Opening statement
   1.2) Items of current interest

2) 8:35 – 10:30 A.M. Amendments to the AP1000 Design Control Document (DCD) (Open) (HBR/MPL)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and Westinghouse Electric Company regarding amendments to the AP1000 DCD and related matters.

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

10:30 – 10:45 A.M. ***** BREAK *****

3) 10:45 – 12:15 P.M. Draft Final Regulatory Guide 5.71, "Cyber Security Programs for Nuclear Facilities" (Open/Closed) (GEA/CEA)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding draft final Regulatory Guide 5.71, "Cyber Security Programs for Nuclear Facilities," NRC staff's resolution of public comments, and related matters.

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

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

12:15 – 1:15 P.M. ***** LUNCH *****

4) 1:15 – 3:15 P.M. Overview of the Advanced Boiling Water Reactor (ABWR) Design as Applied to the South Texas Project (STP) Combined License Application (COLA) (Open/Closed) (SAK/MB)

4.1) Remarks by the Subcommittee Chairman

4.2) Briefing by and discussions with representatives of the NRC staff and the STP Nuclear Operating Company regarding an overview of the ABWR design as it applies to the STP COLA and related matters.

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

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

3:15 – 3:30 P.M. ***** BREAK *****

5) 3:30 – 5:30 P.M. NRC Staff's Plan for the STP COLA Review (Open) (SAK/MB)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding staff's Plan for reviewing the STP COLA and related matters.

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

5:30 – 5:45 P.M. ***** BREAK *****

6) 5:45 – 7:00 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

6.1) Amendments to the AP1000 DCD (HBR/MPL)

6.2) Draft Final Regulatory Guide 5.71, "Cyber Security Programs for Nuclear Facilities" (GEA/CEA)

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

7) 8:30 – 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (MVB/EMH)

8) 8:35 – 10:00 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed) (MVB/EMH)

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

8.2) 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.]

9) 10:00 – 10:15 A.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.

10:15 – 10:30 A.M. * BREAK ***

10) 10:30 – 12:00 P.M. Preparation for Meeting with the Commission on December 4, 2009 (Open) (MVB, et al. /EMH, et al.)
Discussion of the topics for meeting with the Commission on December 4, 2009.

12:00 – 1:00 P.M. * LUNCH ***

11) 1:00 – 3:30 P.M. Draft ACRS Report on the NRC Safety Research Program (Open) (DAP, et al./HPN, et al.)
11.1) Remarks by the Subcommittee Chairman
11.2) Discussion of a draft ACRS report on the NRC Safety Research Program

3:30 – 3:45 P.M. * BREAK ***

12) 3:45 – 4:45 P.M. Significant Operating Experience (Open) (JDS/KDW)
12.1) Remarks by the Subcommittee Chairman
12.2) Briefing by and discussion with the Chairman of the ACRS Subcommittee on Plant Operations and Fire Protection regarding significant operating events, insights gained from these events, and any follow-up actions by the Subcommittee and/or the Full Committee.

13) 4:45 – 5:15 P.M. Subcommittee Reports (Open)
13.1) Report by and discussions with the Chairman of the ESBWR Subcommittee regarding Resolution of Open Items Associated with the Review of the ESBWR Design Certification that were discussed during the meeting on October 20-22, 2009. (MLC/CLB/KDW)

13.2) Report by and discussions with the Chairman of the EPR Subcommittee regarding Evolutionary Power Reactor (EPR) Design Certification Application Review that was discussed during the meeting on November 3, 2009. (DAP/DAW)

13.3) Report by and discussions with the Chairman of the Radiation Protection and Nuclear Materials Subcommittee regarding proposed changes to NUREG-1520, "Standard Review Plan for Review of a License Application for a Fuel Cycle Facility," that was discussed during the meeting on November 4, 2009. (MTR/JHF)

5:15 – 5:30 P.M. ***** BREAK *****

14) 5:30 – 7:00 P.M Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
14.1) Amendments to the AP1000 DCD (HBR/MPL)
14.2) Draft Final Regulatory Guide 5.71, "Cyber Security Programs for Nuclear Facilities" (GEA/CEA)

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

15) 8:30 – 10:00 A.M. Preparation of ACRS Reports (Open)
Continue discussion of the proposed ACRS reports listed under Item 14.

10:00 – 10:15 A.M. ***** BREAK *****

16) 10:15 – 1:00 P.M. Process for ACRS Review of Amendments to the DCD of Previously Certified Reactor Designs (Open) (MVB/CS)
16.1) Remarks by the ACRS Chairman
16.2) Discussion of potential enhancements to the current ACRS process for reviewing amendments to the DCDs related to previously certified reactor designs.

17) 1:00 – 1:30 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.

- One (1) electronic copy of each presentation should be emailed to the Designated Federal Official 1 day before the meeting. If an electronic copy cannot be provided within this timeframe, presenters should provide the Designated Federal Official with a CD containing each presentation at least 30 minutes before the meeting.

APPENDIX V

LIST OF HANDOUTS
566th ACRS MEETING
OCTOBER 8-10, 2009

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

II Combined License Application for North Anna, Unit 3, Economic Simplified Boiling Water Reactor (ESBWR) and the Draft Safety Evaluation Report (SER) with Open Items
3. Agenda
4. Status Report

III License Renewal Application and Final SER for the Susquehanna Steam Electric Station, Units 1 and 2
5. Proposed Schedule
6. Status Report
7. Subcommittee meeting minutes dated June 1, 2009
8. ACRS Consultant Report, dated March 21, 2009 from John Barton to Dr. William Shack, "Susquehanna Steam Electric Station, Units 1 and 2, License Renew Application"

IV Steam Generator Action Plan (SGAP) Task 3.5, "A Risk Assessment of Consequential Steam Generator Tube Ruptures," and other SGAP items
9. Proposed Schedule
10. Status Report

V Oyster Creek 3-Dimensional Structural Analysis of the Drywell Shell
11. Proposed Schedule
12. Status Report
13. Structural Integrity Associates, Inc., "Structural Evaluation of the Oyster Creek Drywell Summary Report," January 9, 2009
14. Structural Integrity Associates, Inc., "Oyster Creek Drywell Sandbed Region Wall Thinning Sensitivity Analyses Summary Report," January 9, 2009
15. Memo from Meena Khanna (NRR) to Richard Conte (Region I), "Assessment of the Oyster Creek 3-D Finite Element Analysis of the Drywell Shell," dated May 12, 2009
16. Letter from Jill Lipoti (State of NJ) to US NRC transmitting, "New Jersey Department of Environmental Protection Oyster Creek Drywell Review," prepared by Becht Nuclear Services, dated April 7, 2009
17. Clarence D. Miller, "Applicability of ASME Code Case N-284-1 to Buckling Analysis of Drywell Shell," June 15, 2006

APPENDIX V

LIST OF HANDOUTS
566th ACRS MEETING
OCTOBER 8-10, 2009

V Oyster Creek 3-Dimensional Structural Analysis of the Drywell Shell (continued)

18. Letter from William J. Shack, Chairman, ACRS, to Dale E. Klein, Chairman, US NRC, "Report on the Safety Aspects of the License Renewal Application for the Oyster Creek Generating Station," dated February 8, 2007
19. Letter from Michael P. Gallagher to US NRC, "Updated Information Regarding the Results of the Structural Analysis of the Oyster Creek Drywell Shell, Performed in Support of License Renewal"
20. ASLBP Memo, dated October 29, 2009, "Addressing the Issue Referred by the Commission Regarding the Adequacy of AmerGen's Proposed 3-D Finite Element Structural Analysis Studies"
21. The Commission Order, CLI-09-07, dated April 1, 2009,"License Renewal for Oyster Creek Nuclear Generating Station
22. Exelon's presentation slides used during 9/23/2009 Subcommittee meeting
23. NRC Staff's presentation slides used during 9/23/2009 Subcommittee meeting
24. Rich Webster's presentation slides used during 9/23/2009 Subcommittee meeting

VI Draft Final Revision 2 to Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants"

13. Proposed Schedule
14. Status Report
15. Draft Regulatory Guide DE-1214 (Proposed Revision 2 of Regulatory Guide 1.189 dated March 2007 (issued for public comment)
16. Draft Regulatory Guide DE-1214 (Proposed Revision 2 of Regulatory Guide 1.189 dated March 2007 (with Industry Comments incorporated - strike out version)
17. Draft Regulatory Guide DE-1214 (Proposed Revision 2 of Regulatory Guide 1.189 dated March 2007 (with Industry Comments incorporated - clean version)
18. NRC Responses to Comments – Comments on Draft RG 1.189, Revision 2 (DG-1214)
19. SECY-08-0093, For the Commissioners, from R. W. Borchardt, Executive Director for Operations, "Resolution of issues related of Fire-Induced circuit Failures"
20. NEI 00-01, Revision 2, "Guidance for Post Fire Safe Shutdown Circuit Analysis"

VII 10 CFR Part 52 Regulatory Process

Presentation information could not be located at the time the minutes were prepared.

North Anna 3

North Anna Unit 3 Presentation to Advisory Committee on Reactor Safeguards

Dominion

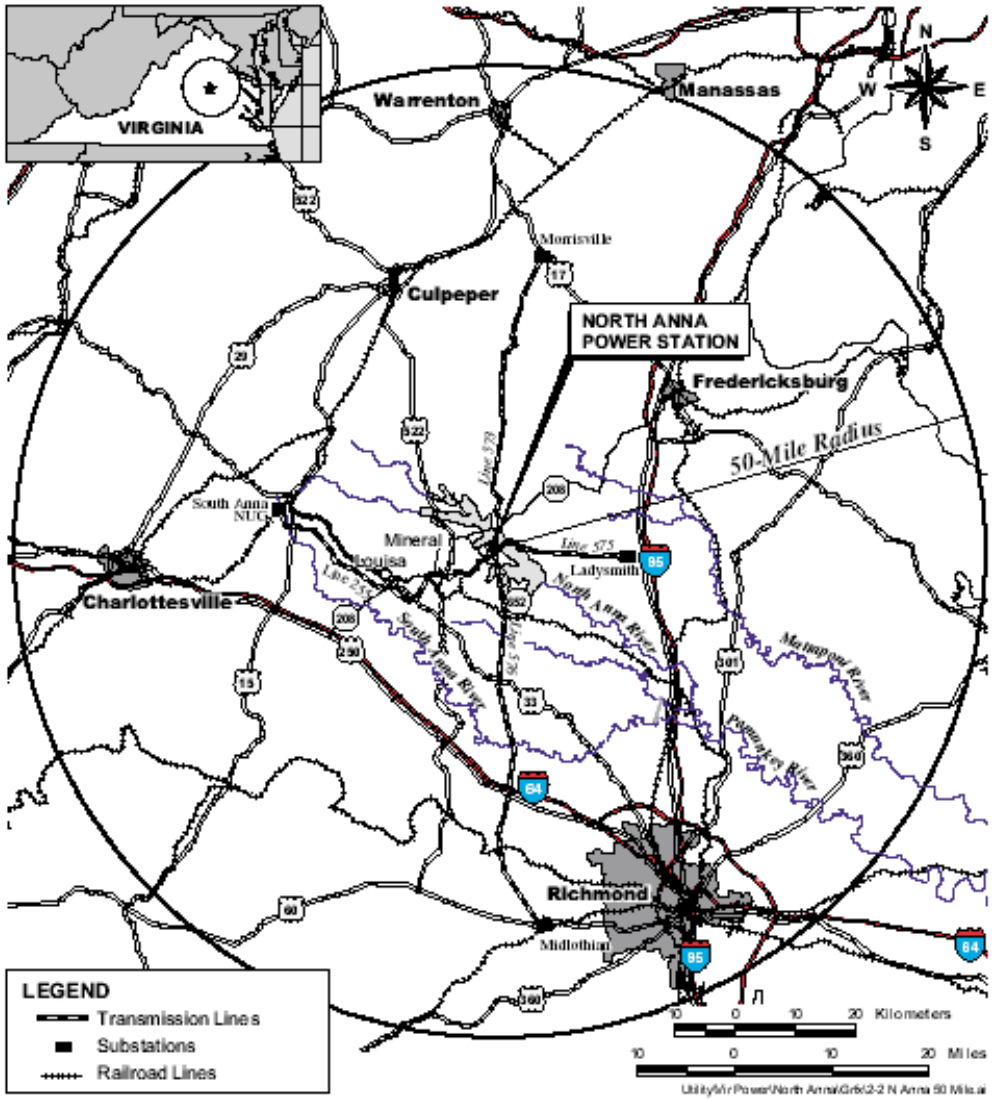
Dominion's North Anna 3 Licensing Activities

• ESP Application submitted	9/25/03
• ESP issued	11/27/07
• COLA submitted	11/27/07
• Draft EIS issued	12/17/08
• COLA Revision 1 submitted	12/20/08

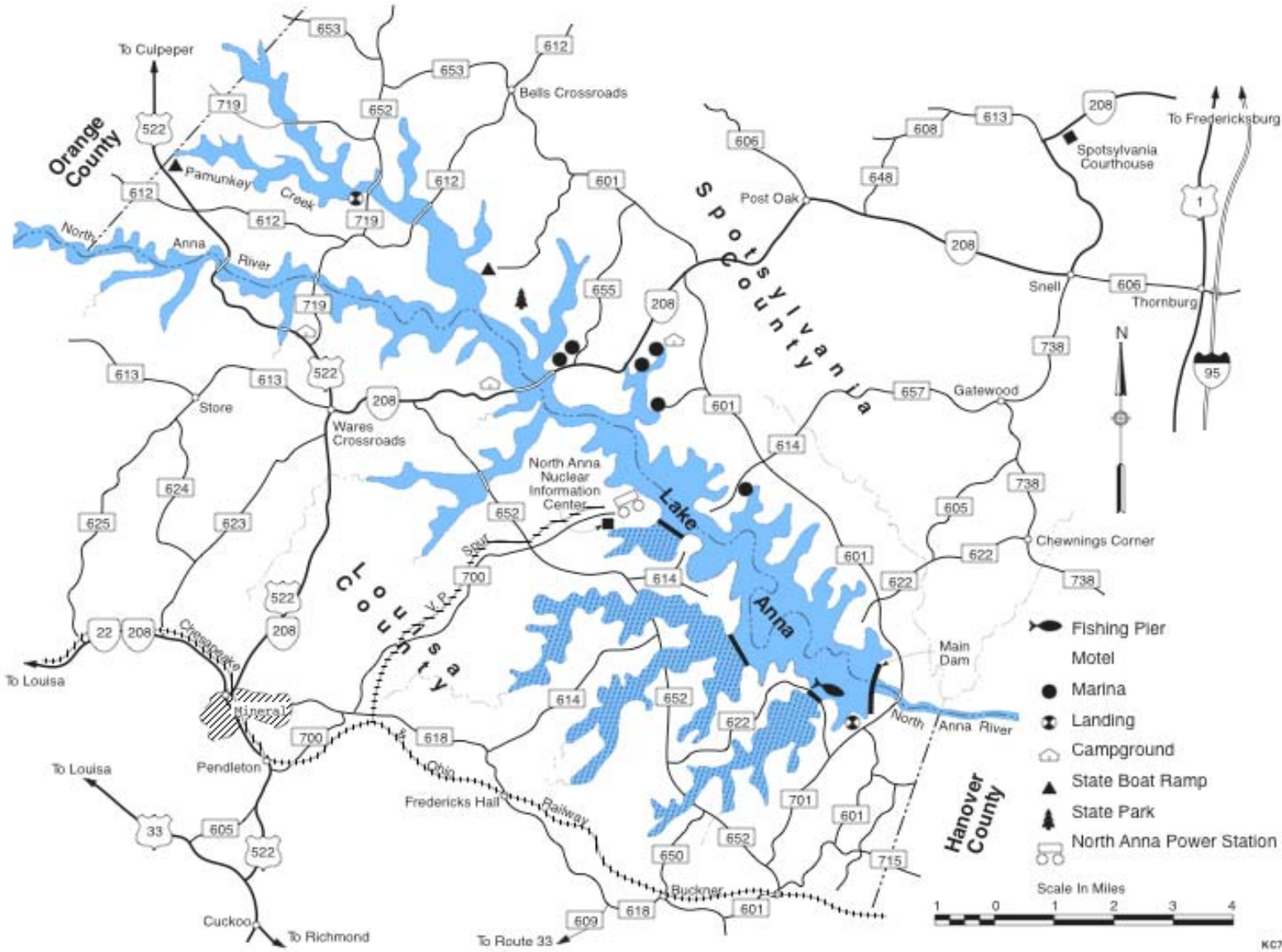
Dominion North Anna 3 Presentations to ACRS Subcommittee

- June 11, 2009
  - Chapters 1, 4, 6, 7, 8, 15, 17, 18, 19
- July 21 and 22, 2009
  - Chapter 5, 9, 10, 11, 12, 13, 16
- August 21, 2009
  - Chapters 2, 3, 14

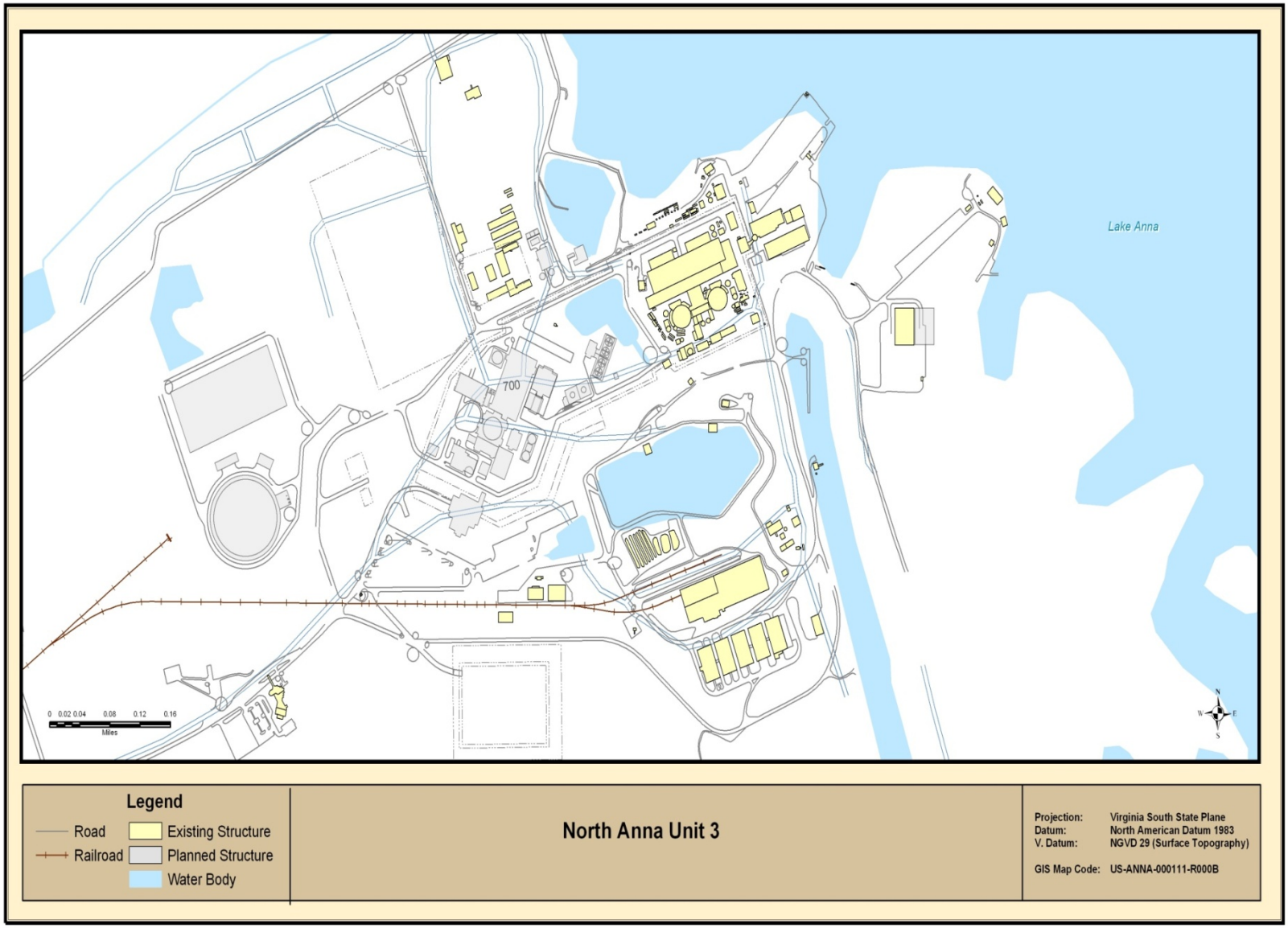
Location of North Anna Power Station



Location of North Anna Power Station



North Anna Power Station



Introduction of COLA Team's Meeting Participants

Dominion

- Joe Hegner
- Gina Borsh

GEH

- Rick Kingston
- Patricia Campbell

Bechtel

- Doug Kemp
- Geoff Quinn

Subject Matter Experts by phone

North Anna 3

ACRS COLA Presentation: Site-Specific Topics and ACRS Inquiries

Introduction

- NAPS U3 FSAR Content
  - Incorporated by reference (IBR):
    - ESBWR DCD as permitted by 10CFR52.55(c)
    - ESPA SSAR
  - Supplemental information added:
    - COL Item
    - ESP COL Action Item
    - ESP Permit Condition
    - Conceptual Design Information
    - Supplemental Information (e.g., Reg Guide 1.206)

Discussion Topics

- Chapter 2 – Site Characteristics
  - 2.4, Hydrology
  - 2.5, Geology, Seismology, and Geotechnical Engineering
- 9.2.3 Makeup Water System
- 10.4.5 Circulating Water System
- 9.2.1 Plant Service Water System

Chapter 2, Site Characteristics: Background

- Compares Unit 3 FSAR site characteristics and facility design values with corresponding DCD, ESP, or ESP Application SSAR values to determine if:
  - Unit 3 site characteristics fall within DCD's site parameters
  - Unit 3 site characteristics or facility design fall within ESP's site characteristics or design parameters [ESP used plant parameter envelope (PPE)]
  - Unit 3 site characteristics and design values fall within SSAR site characteristic and design parameter values

North Anna 3

Chapter 2:
Site Characteristics

Section 2.4, Hydrology

Chapter 2, Site Characteristics: 2.4 Hydrology

- SSAR Section 2.4 is incorporated by reference and supplemented with:
  - Impact on wetlands and two intermittent streams that flow into Lake Anna [NAPS COL]
  - Design plant grade is above design basis flooding level [NAPS COL]
  - Location of safety-related SSCs is above maximum water surface elevation produced by local intense precipitation [NAPS ESP COL]

Chapter 2, Site Characteristics: 2.4 Hydrology (cont.)

- Supplemental Information (cont):
  - Water supply to the UHS is above design plant grade elevation [NAPS COL]
  - Engineered underground reservoirs or storage basins not used for UHS [NAPS ESP COL]
  - Emergency cooling water for Unit 3 is provided from UHS, which is not affected by ice conditions [NAPS COL]
  - Lake Anna is not used for safety-related water withdrawals for Unit 3 [NAPS ESP COL]

Chapter 2, Site Characteristics: 2.4 Hydrology (cont.)

- Supplemental Information (cont):
  - Embankment for water intake structure is protected by rip-rap [NAPS ESP COL]
  - Maximum PMP water level in power block area is 2.8 ft below design plant grade elevation for safety-related facilities [NAPS COL]
  - Circulating water system operating modes [NAPS ESP COL]

Chapter 2, Site Characteristics: 2.4 Hydrology (cont.)

- Supplemental Information (cont):
  - Additional borings, groundwater level measurements, hydraulic conductivity testing [NAPS COL, NAPS ESP VAR]
  - Groundwater supply wells, groundwater use, and groundwater level monitoring program [NAPS COL, NAPS ESP VAR]
  - Estimated maximum groundwater level in power block area is 7 ft below design plant grade elevation of 290 ft; therefore, permanent dewatering system is not required for safe operation of Unit 3 [NAPS COL]

Chapter 2, Site Characteristics: 2.4 Hydrology (cont.)

- Supplemental Information (cont):
  - Mitigating design features are incorporated into design of Unit 3 to preclude an accidental release of liquid effluents [NAPS COL, NAPS ESP PC]
  - Accidental release of radioactive liquid effluent to either groundwater or surface water complies with 10 CFR 20 limits for release to unrestricted areas [NAPS COL]
  - Unit 3 will shut down when water level in Lake Anna drops below Elevation 242 ft msl [NAPS ESP COL]

North Anna 3

Chapter 2: Site Characteristics

Section 2.5, Geology, Seismology, and Geotechnical Engineering

Chapter 2, Site Characteristics: 2.5 Geology, Seismology, and Geotechnical Engineering

- SSAR Section 2.5 is incorporated by reference and supplemented with:
  - Additional Unit 3 borings to further describe the site stratigraphy [NAPS COL]
  - Weathered or fractured rock at foundation level for safety-related structures will be excavated and replaced with lean concrete before foundation construction [NAPS ESP PC]
  - Future excavations for safety-related structures will be geologically mapped and unforeseen geologic features will be evaluated [NAPS ESP PC]

Chapter 2, Site Characteristics: 2.5 Geology, Seismology, and Geotechnical Engineering

- Supplemental Information (cont):
  - Unit 3 OBE ground motion is one-third of FIRS and is bounded by DCD OBE [NAPS COL]
  - Description of excavations, fills, and slopes, including discussion of excavation and fill methods and stability [NAPS ESP COL, NAPS ESP PC]
  - SWV profiles for soil used for liquefaction analysis and slope stability analysis; backfill for FWSC [NAPS COL, NAPS ESP COL]

Chapter 2, Site Characteristics: 2.5 Geology, Seismology, and Geotechnical Engineering (cont)

- Supplemental Information (cont):
  - Liquefaction Potential [NAPS COL, NAPS ESP PC]
    - Chances of liquefaction occurring in Zone IIA saprolite are extremely low; will not impact stability of any Seismic Category I or II structure
  - Foundation and Slope Stability
    - Allowable bearing capacity values are adequate for Seismic Category I and II structures, and Radwaste Building [NAPS COL]
    - Total and differential settlement values are within limits for Seismic Category I structures [NAPS ESP COL]
    - Static and seismic lateral earth pressures are provided [NAPS ESP COL]
    - Slope stability analysis [NAPS ESP COL]

North Anna 3

Water Systems

Make-up Water System

Circulating Water System

Plant Service Water System

Chapter 9, Auxiliary Systems: Section 9.2.3 Makeup Water System

- Function: Supplies demineralized water to equipment listed in DCD
  - Nonsafety-related system
- During shutdown/refueling/startup mode, increases in plant water consumption may require use of temporary demineralization subsystem
  - Per DCD, design capacity is for normal power requirements
  - Increase is due to draining of non-contaminated systems that can't be refilled with condensate-quality water
  - Ensures adequate supply to supplement CST volume
  - Effective outage Makeup Water usage plan will minimize need for temporary subsystem

Chapter 10, Steam and Power Conversion System: Section 10.4.5: Circulating Water System

- PRA model for system is described in DCD PRA
- Plant-specific design of circulating water system (e.g., cooling towers) is enveloped by DCD PRA analysis
  - Condenser only credited in PRA for events with low significance – where offsite power is available
  - DCD PRA includes analysis of CIRC pump failures due to power loss and trash rack plugging
  - Cooling tower fans not specifically modeled in DCD PRA
    - Fan power supply is same as CIRC water pumps

Chapter 9, Auxiliary Systems: Section 9.2.1 Plant Service Water System (PSWS)

- Function: PSWS rejects heat from nonsafety-related Reactor Component Cooling Water System (RCCWS) and Turbine Component Cooling Water System (TCCWS)
  - No safety-related function
  - Categorized in DCD as RTNSS C

Chapter 9, Auxiliary Systems: Section 9.2.1 Plant Service Water System (PSWS) (cont)

- DCD conceptual design provides for connection between PSWS and normal plant heat sink (CIRC)
- NAPS U3 design does not include this connection
  - Operating conditions might require design changes for other systems (RCCW, TCCW)
  - LOOP would require re-alignment
  - Increased capital cost (e.g., larger normal power heat sink, added booster pumps or larger PSWS pumps) and operating cost (e.g., added booster pump maintenance and increased auxiliary power demand) without offsetting benefits
- NAPS U3 PSWS meets the DCD's performance requirements without cross-connect

Chapter 9, Auxiliary Systems: Section 9.2.1 Plant Service Water System (PSWS) (cont)

- Piping for underground portion of system will be fiberglass reinforced polyester (FRP) pipe
- Rationale
  - Meets design requirements
  - Advantages over other options (e.g., long-term resistance to internal and external corrosion and biological attack)
- ASME code case being used as guidance to develop applicable requirements

Conclusions

- R-COLA work continues
  - Addressing remaining RAIs
  - Advanced SER with no Open Items scheduled to be issued September 2, 2010
  - ACRS review of SER with no Open Items scheduled to be completed December 21, 2010

Presentation to the ACRS Full Committee

North Anna Unit 3 COL Application

Safety Evaluation Report (SER) with Open Items (OI)

October 8, 2009

Office of New Reactors

ACRS Full Committee Presentation
North Anna SER/OI

Agenda

- Opening Comments
- Overview – Applicant and Staff
- Open Items Update
- Hydrologic Engineering (SER Section 2.4)
- Geology, Seismology, & Geotechnical Engineering (SER 2.5)
- Functional Design, Qualification, and Inservice Testing Programs and Environmental Qualification of Equipment (SER 3.9.6, 3.11)
- Plant Water Systems – Circulating, Makeup, & Plant Service (SER 10.4.5, 9.2.3, & 9.2.1)
- ITAAC (SER 14.3)
- COL Application Review Schedule

ACRS Full Committee Presentation
North Anna SER/OI

Staff Overview

- SER/OI complete
  - 19 chapters + appendices
  - DNRL Memorandum to ACRS [ML092150277]
- North Anna COL Application, Revision 1 (12/08)
  - Incorporates by reference
    - Early Site Permit (ESP-003) – Site Safety Analysis Report
    - ESBWR Design Control Document, Revision 5
  - Includes responses to staff RAIs (Phase 1)

ACRS Full Committee Presentation
North Anna SER/OI

Staff Overview (continued)

- ACRS Subcommittee Meetings
  - Focus – North Anna COL application
  - Presentation topics
    - June 18 – Chapters 1, 4, 6, 7, 8, 15, 17, 18, & 19
    - July 21-22 – Chapters 5, 9, 10, 11, 12, 13, and 16
    - August 21 – Chapters 2, 3, and 14
  - No significant issues identified
  - Questions/comments
    - Addressed in meetings or written response
    - DCD resolution

ACRS Full Committee Presentation
North Anna SER/OI

Staff Overview (continued)

- Today's Presentations
  - Open Item update
  - Site characteristics
    - relationship with ESP & DCD
  - Site-specific systems
    - interface DCD
  - Subcommittee transcript comments
- Presentation format
  - Dominion – application (FSAR) content
  - Staff – safety evaluation
- Milestone – 1st COLA SER/OI presentation to ACRS

ACRS Full Committee Presentation
North Anna SER/OI

BACKUP SLIDES

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North Anna SER/OI

Staff Overview (Backup)

- Lesson Learned – ACRS subcommittee feedback (June 18th) regarding evaluation of "IBR" information
  - SER: "The staff reviewed … FSAR and checked the referenced DCD to ensure that the combination of the DCD and the information in the COL application represent the complete scope of information relating to this review topic. The review confirmed that the information contained in the application and incorporated by reference addresses the relevant information related to … ."
    Staff presentations to include examples

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Open Items

- Open Item 1-1 – ESBWR DCD not finalized (IBR)
- E-RAI Report – Open RAIs
  - Staff: All open RAIs on path toward resolution
  - RAIs in Sections 2.4, 2.5, 3.9, 3.11, & 9.2 – addressed in technical presentations
  - 5.3.2 PTLR – staff review recent submittal
  - 13.3 FEMA – multiple being addressed by offsite entities;
  - 13.6 Security – on-going review with RAIs
  - Chapter 16 – finalize plant-specific technical specifications
  - Turbine Missile Probability Analysis – review commencing – possible RAIs

Open RAIs

RAI	Status	Subject
01-4	Responded	source, byproduct and special nuclear material
02.04.02-4	Responded	Local intense precipitation flood calculation
02.04.02-5	Responded	Local intense precipitation flood calculation
02.04.02-6	Responded	Local intense precipitation flood calculation
02.04.02-7	Responded	Local intense precipitation flood calculation
02.04.12-2	Responded	groundwater modeling.
02.04.13-4	Responded	radionuclide transport analysis.
02.05.04-20	Issued/Open	backfill ITAAC
02.05.04-21	Issued/Open	properties of concrete fill.
03.02.01-7	Issued/Open	SSCs necessary for continued safe operation following OBE
03.09.06-1	Responded	design and qualification of safety-related valves and dynamic restraints
03.10-1	Issued/Open	seismic qualification of equipment.
03.11-1	Responded	environmental qualification operational program
03.11-2	Responded	implementation of environmental qualification approach in DCD
05.03.02-2	Issued/Open	Pressure-temperature limits (GEH Licensing Topical Report NEDC-33441P)
05.03.02-3	Issued/Open	Pressure-temperature limits
05.03.02-4	Issued/Open	Pressure-temperature limits
05.03.02-5	Issued/Open	Pressure-temperature limits.
05.03.02-6	Issued/Open	Pressure-temperature limits.
05.03.02-7	Issued/Open	Pressure-temperature limits
05.03.02-8	Issued/Open	Pressure-temperature limits
09.02.01-13	Responded	fiberglass reinforced polyester pipe in Plant Service Water System
11.04-3	Issued/Open	storage of low-level radioactive wastes.
12.03-12.04-8	Responded	minimization of radioactive contamination and waste (10 CFR 20.1406(a))
13.03-5	Issued/Open	ITAAC for Emergency Operations Facility.
13.03-6	Responded	ITAAC for emergency exercises.
13.03 FEMA	Issued/Open	offsite emergency planning issues
13.06 Security	Issued/Open	RAIs on several topics
16-1	Responded	plant-specific technical specifications.

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Section 2.4

Hydrologic Engineering

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North Anna SER/OI

Section 2.4 - General Remarks

- Section 2.4 includes many different water topics, from floods and tsunamis to ice and groundwater. Emphasis is on physical rather than biological issues.
- The ESP review was comprehensive.
- Six variances exist in section 2.4, but were of minor technical consequence.

ACRS Full Committee Presentation
North Anna SER/OI

Section 2.4 - General Remarks (Continued)

- ESP Permit Condition 3.E(2) – Single unit only. The permit condition for second unit cooling no longer applies.
- ESP Permit Condition 3.E(3) Features to Preclude Accidental Releases of Radionuclides into Potential Liquid Pathways
  - Below-grade tanks are in steel-lined compartments large enough to contain entire contents
  - Above-grade condensate storage tank is in a basin large enough to contain entire contents
  - *Staff concluded that these design features satisfy the permit condition*

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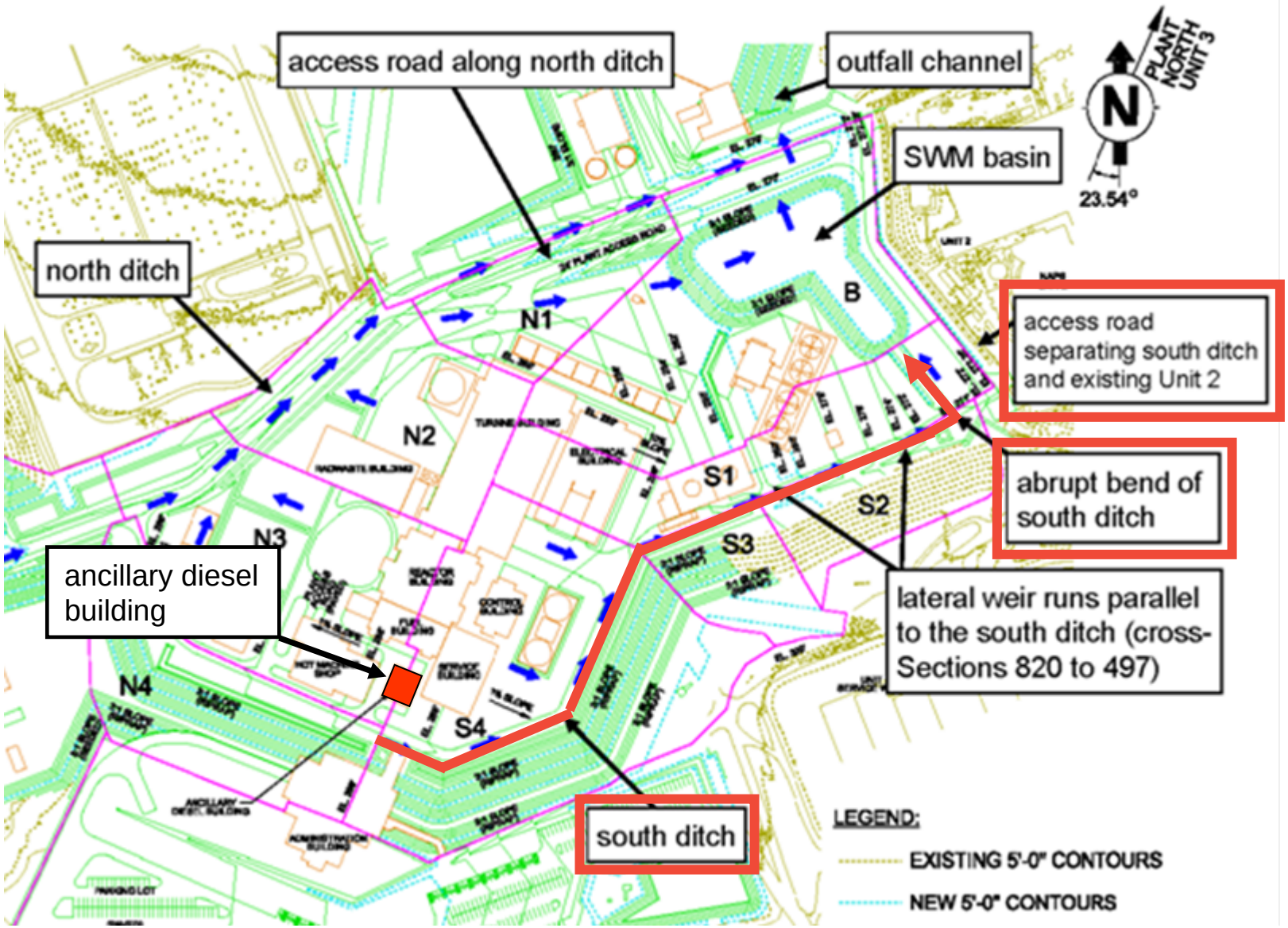
North Anna SER/OI

Section 2.4.2 – Floods

There are two different flooding issues:

- Watershed-Scale Flooding
  - The design plant grade elevation is above probable maximum flood (including dam breaks and wave action)
  - No Open Items
- Locally Intense Precipitation Flooding
  - Two Open Items

Section 2.4.2 – Floods (continued)



Based on FSAR Figure 2.4-201

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Section 2.4.2 – Floods (continued)

- Locally Intense Precipitation Flooding
  - Water Levels near Unit 3
    - Modeling analysis demonstrates that water elevations are not high enough near Unit 3 nuclear island to be problematic.

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North Anna SER/OI

Section 2.4.2 – Floods (continued)

- Locally Intense Precipitation Flooding (continued)
  - Open Item 2.4.2-2
    - Provide updated HEC-RAS input files for NRC review. Addresses DCD Rev. 5, which adds Ancillary Diesel Building.
    - Provide additional details on the South Ditch to ensure system will function as described.
    - Three RAIs under review (2.4.2-4, 2.4.2-5, 2.4.2-6)

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Section 2.4.2 – Floods (continued)

- Locally Intense Precipitation Flooding (continued)
  - Open Item 2.4.2-3
    - Uncertainty that flood will overtop access road/safety dike that protects existing units (10 CFR 52.79(a)(31))
    - One RAI under review (2.4.2-7)

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North Anna SER/OI

Section 2.4.12 – Groundwater

- Open Item 2.4.12-2
  - Concern: Groundwater level must be more than 2 ft below plant grade of 290 ft
  - Staff Issues:
    - Model sensitivity studies of effect of drain cell properties on groundwater elevations
    - Effectiveness of surface water drainage features as groundwater drains
  - One RAI under review (2.4.12-2)

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North Anna SER/OI

Section 2.4.13 Accidental Releases of Radioactive Liquid Effluents

- Open Item 2.4.13-4
  - Concern: Staff wants to confirm that the Applicant's groundwater transport analysis is a bounding analysis.
  - Staff issues:
    - Certain literature K_d values used in transport analysis were greater than minimum measured onsite K_d
    - Hydraulic conductivity used in transport analysis was less than the maximum measured onsite
  - One RAI (2.4.13-4)

QUESTIONS ?

Section 2.4
Hydrologic Engineering

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Section 2.5

Geology, Seismology, and Geotechnical Engineering

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- 2.5.1 Basic Geologic & Seismic Information
- 2.5.3 Surface Faulting
- 2.5.5 Stability of Slopes
  - Provided additional site geologic and seismic information
  - Performed new slope stability analyses using updated site data
  - No outstanding issues

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North Anna SER/OI

- 2.5.2 Vibratory Ground Motion
  - Addressed COL items and ESP permit conditions
  - Revised Ground Motion Response Spectra (GMRS) based on updated site subsurface profile
  - No outstanding issues

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North Anna SER/OI

- 2.5.4 Stability of Subsurface Materials and Foundations
  - Addressed the COL items by providing additional boring data, site soil profiles, subsurface material properties, and stability analyses
  - Resolved RAIs/OIs related to:
    - static and dynamic properties of backfill soil
    - foundation stability analysis
    - coefficient of friction at foundation interface
    - dynamic settlement calculation of soil slope

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North Anna SER/OI

- 2.5.4 Stability of Subsurface Materials and Foundations – Remaining Open Items
  - Clearly describe acceptance criteria in backfill soil ITAAC and define field density test frequency (02.05.04-20)
  - Clearly define concrete fill properties in COL FSAR (02.05.04-21)

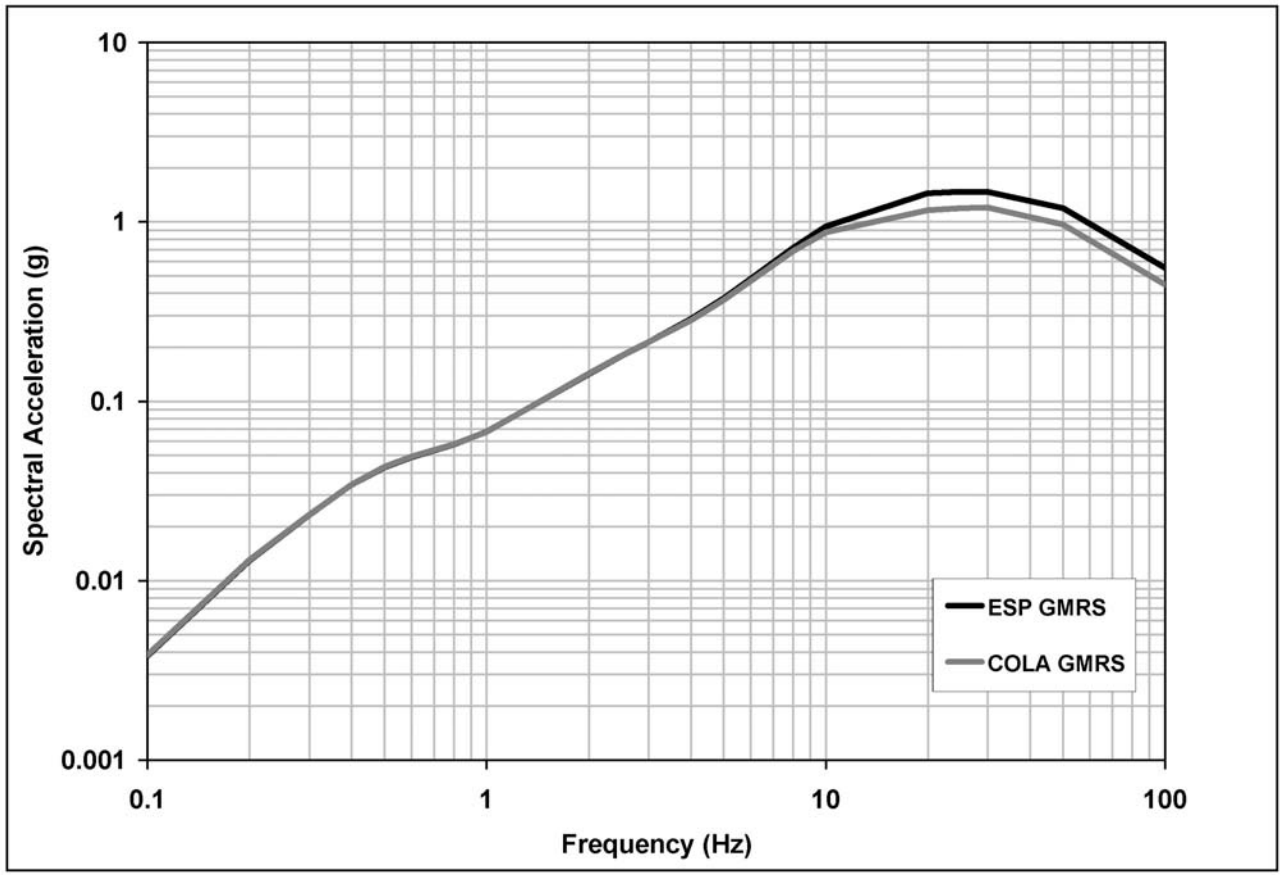
QUESTIONS ?

Section 2.5
Geology, Seismology, and Geotechnical Engineering

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Back up slides

ACRS Full Committee Presentation
North Anna SER/OI



Comparison of Horizontal Ground Motion Response Spectra (GMRS)

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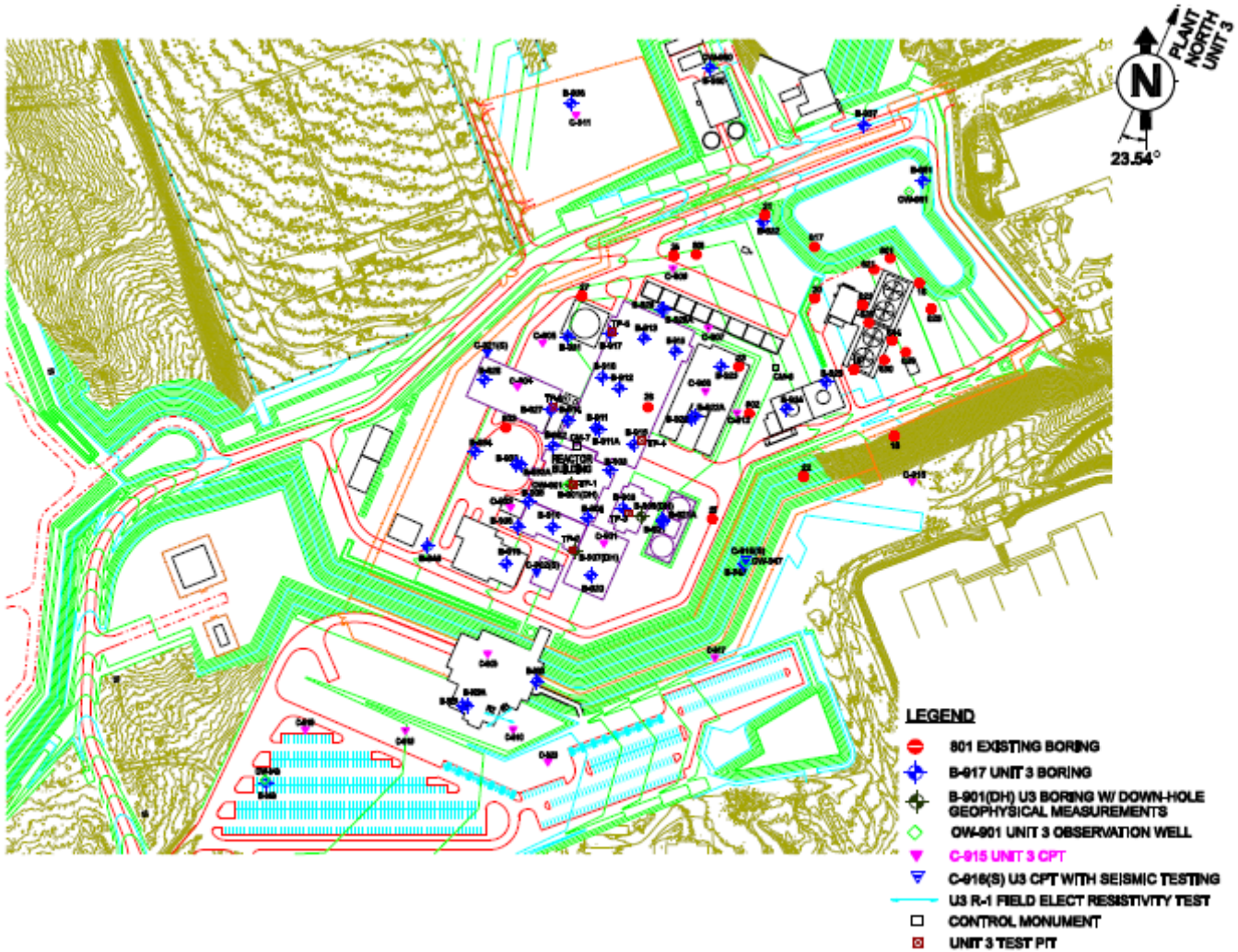

Unit 3 Boring Locations – Power Block

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

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North Anna SER/OI

Site Investigations	ESP	COL
Borings	7	55
Corn Penetration Test	8	23
Test Pits	0	6
Observation Wells	9	7
Velocity Test	5	6

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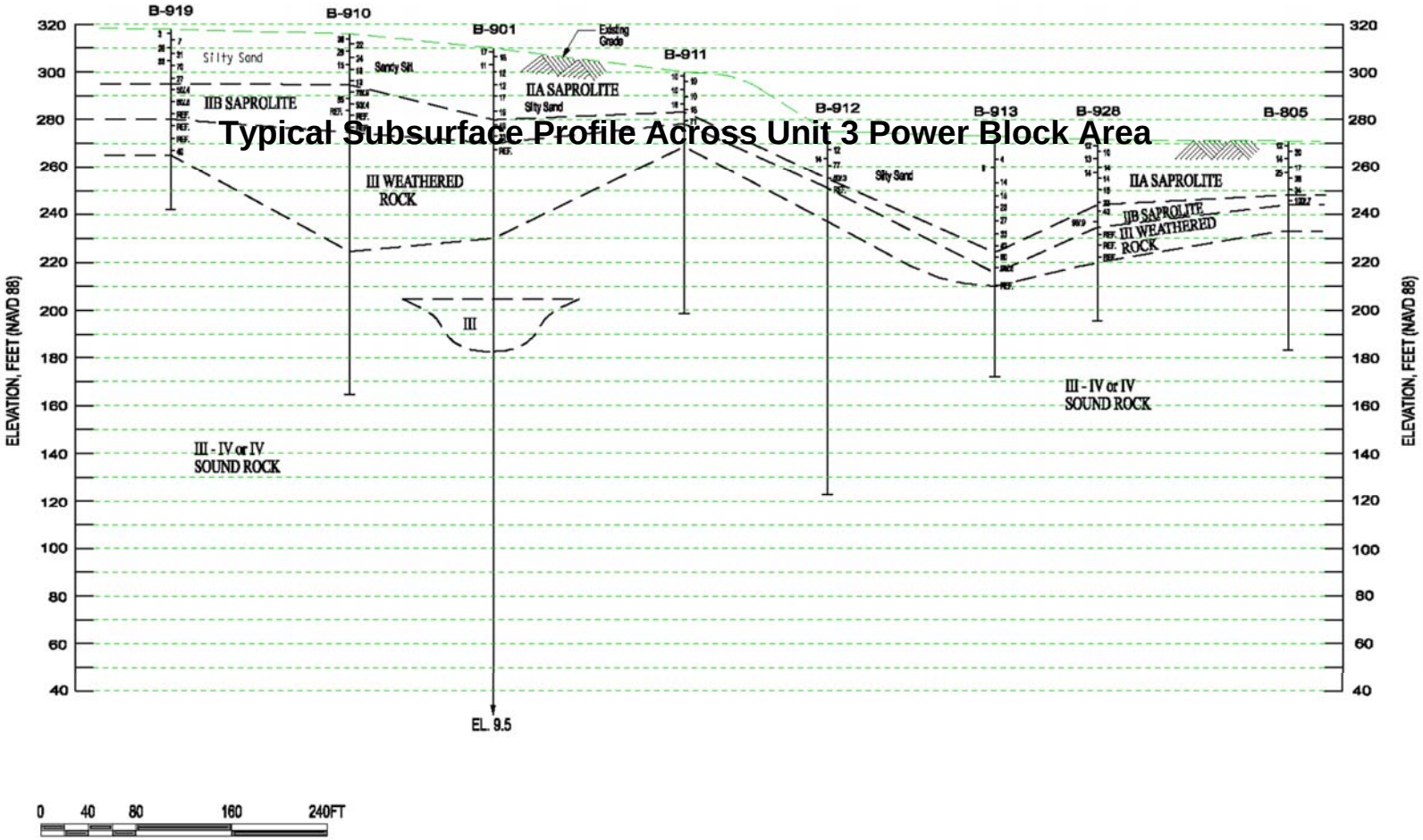
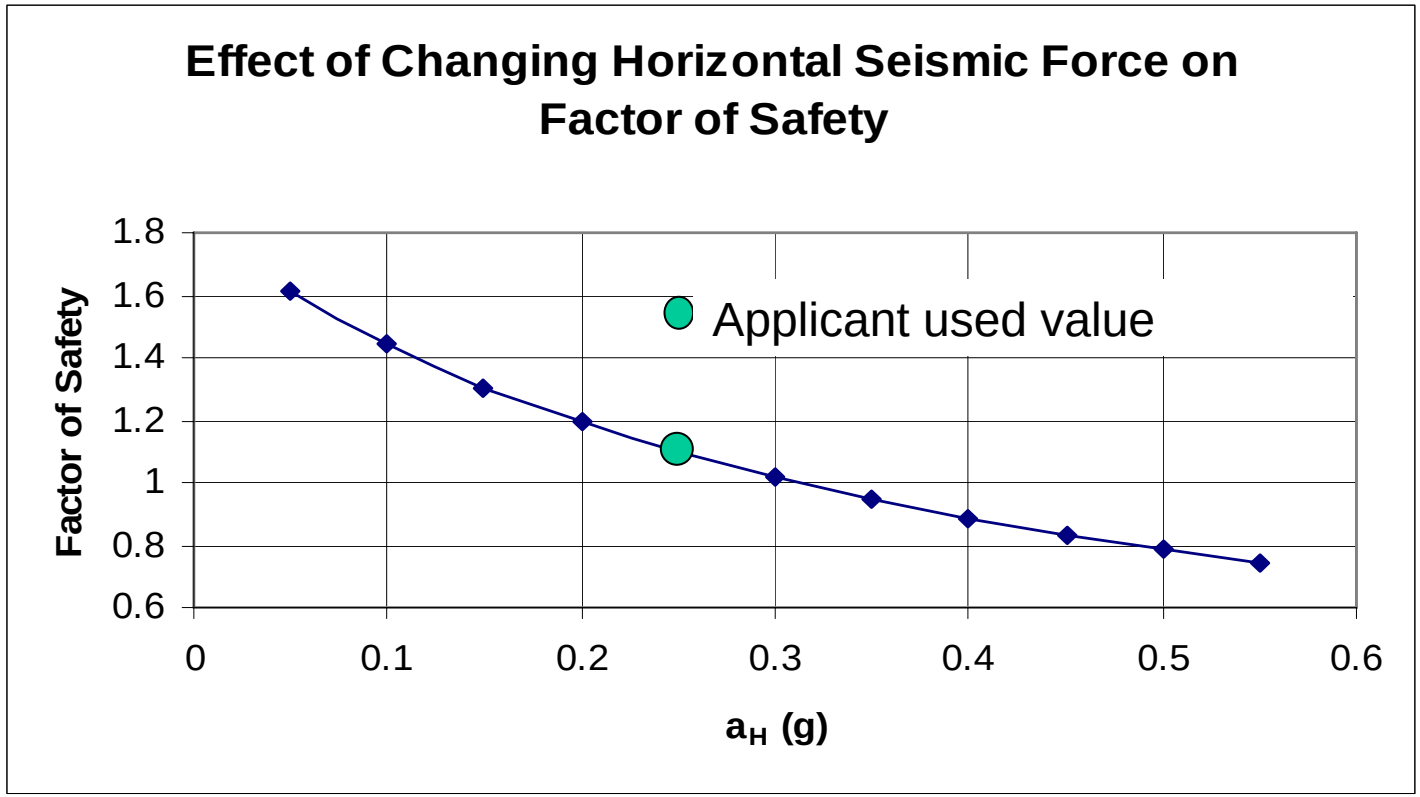


Typical Subsurface Profile Across Unit 3 Power Block Area

ACRS Full Committee Presentation
North Anna SER/OI

Staff Stability of Slopes Confirmatory Analysis Results



Note: The return period for 0.3g at 2.5 Hz is about 70,000 years, which corresponds to an annual frequency of exceedance of 1.4 x 10-5 per year.

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North Anna SER/OI

Section 3.9.6

Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints

Section 3.11

Environmental Qualification of Mechanical and Electrical Equipment

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Section 3.9.6

- COL application relies on ESBWR DCD and FSAR to fully describe functional design, qualification, and IST programs for pumps, valves, and dynamic restraints
- In response to RAIs, Dominion and GEH revised FSAR and DCD
- NRC performed audit of GEH design and procurement specifications in July 2009 to evaluate implementation of DCD provisions

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Section 3.9.6

- DCD specifies ASME Standard QME-1-2007 for functional design and qualification of new valves with key QME-1 aspects required for previously qualified valves
- DCD describes valve IST program based on 2001 Edition/2003 Addenda of ASME OM Code
- FSAR supplements DCD for IST provisions and periodic verification of design-basis capability of power-operated valves
- FSAR describes snubber IST program consistent with ASME OM Code, Section ISTD

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Section 3.9.6

- Open Item 3.9.6-01 (Implementation of DCD qualification and IST provisions)
  - NRC audit results summarized in memorandum dated September 1, 2009
  - GEH provided response to audit follow-up items in letter dated September 21, 2009
  - GEH response resolves most audit follow-up items with minor issues remaining to close open item

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Section 3.11

- FSAR incorporates by reference DCD description of EQ program for mechanical and electrical equipment
- DCD description of EQ process acceptable based on NUREG-1503 (ABWR SER)
- ITAAC will confirm electrical and mechanical EQ
- NRC performed audit of GEH design and procurement specifications in July 2009 to evaluate implementation of DCD EQ provisions

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Section 3.11

- Open Item 3.11-01 (Implement DCD EQ provisions)
  - NRC audit summarized in September 1, 2009, memorandum
  - GEH response provided in September 21, 2009, letter that resolves most audit follow-up items
- Open Item 3.11-02 (Safety-related mechanical equipment EQ)
  - NRC staff review of GEH audit response to be used to close open item

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QUESTIONS ?

Section 3.9.6
Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints

Section 3.11
Environmental Qualification of Mechanical and Electrical Equipment

Section 9.2.1

Plant Service Water System

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Section 9.2.1
Plant Service Water System (PSWS)

- Open Items – Subcommittee interest (July meeting)
- Open Items Resolved (Confirmatory Items)
  - OI #8 - Revised Interface Requirement (ITAAC)
    - PSWS ability to remove necessary heat load
  - OI #9 - Above ground PSWS piping materials
  - OI #10 - PSWS treatment under Maintenance Rule
  - OI #11 - Clearly identify CDI and COL items in FSAR
  - OI #12 - Design capability of the Auxiliary Heat Sink
    - Initial plant test program
    - Water hammer design features

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Section 9.2.1
Plant Service Water System (PSWS)

- Open Item #13 – Buried Fiberglass Piping (FRP)
- Proposed Resolution
  - NRC-approved ASME Code, Section III Code Case N-155-2, provides rules for fabrication and installation of FRP
  - Procedures in Code Case provide for FRP to be properly designed and installed and thus provide for structural integrity and safety commensurate with RTNSS status of the system
  - Applicant agreed to use Code Case to design/install FRP
  - Path to resolve OI #13:
    - Staff will audit applicant's use of Code Case
    - Applicant will document use of Code Case in FSAR

QUESTIONS ?

Section 9.2.1
Plant Service Water System

Section 14.3

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

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Section 14.3 – ITAAC

North Anna ITAAC

- ESBWR Certified Design ITAAC
  - ESBWR DCD Tier 1
  - North Anna incorporates by reference
- Emergency Planning
  (SER 13.3 – 2 open RAIs)
- Physical Security
  (SER 13.6 – on-going review)
- Site-specific Structures, Systems, Components

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Section 14.3 – ITAAC

Site-specific Structures, Systems, Components

- Selection Criteria and Methodology
  - North Anna – same as DCD
  - Applied to systems not evaluated in DCD
- Backfill under Category I Structures
  (SER 2.5.4 – open RAI)
- Plant Service Water System
  (SER 9.2.1 – confirmatory RAI)
- Offsite Power
  (SER 8.2 – confirmatory RAI)
- Other Systems - No ITAAC

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Section 14.3 – ITAAC

Post COL Activities

- ITAAC – license condition to be satisfied before fuel load
- ITAAC closure process – Regulatory Guide 1.215

QUESTIONS ?

Section 14.3

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

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COL Application Review Schedule

Safety Review

- Phase 1 - Requests for Additional Information – Complete
- Phase 2 - SER with Open Items – Complete
- Phase 3 - ACRS Review of SER with Open Items – 11/04/09
- Phase 4 - Advanced SER with no Open Items – 09/10
- Phase 5 - ACRS Review of SER with no Open Items – 12/10
- Phase 6 – Issue Final SER – 02/11

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QUESTIONS / COMMENTS

Susquehanna Steam Electric Station

License Renewal ACRS Meeting

October 8, 2009

Introductions

- Rick Pagodin, General Manager – Engineering
- John Krais, Manager – Special Projects
- Dale Roth, Supervisor Programs and Testing
- Dave Flyte, License Renewal Lead Engineer
- Subject Matter Experts and LR Project Team

Agenda

- Site Description
- Operating Experience Review
- Commitment Process
- Subcommittee Follow-Up Items
  - Underground Medium Voltage Cables
  - Station Blackout Recovery Scoping
  - Condition of Containment
  - Main Steam Line Flow Restrictors
- Summary

Site Description

- Location: Northeast Pennsylvania
- Plant Owners
  - PPL Susquehanna, LLC (90%)
  - Allegheny Electric Cooperative (10%)
- Licensee/Operator: PPL Susquehanna, LLC

Site Description

- Two Units - BWR/4
  - 3952 MWt
  - 1300 MWe
- General Electric (NSSS), Bechtel (AE)
- Ultimate Heat Sink – Spray Pond
- Turbine Cycle Cooling Provided By Natural Draft Towers
- Make-up Water – Susquehanna River

Site Description

Construction Permit	November 1973
Operating License (Unit 1/Unit 2)	July 1982/March 1984
Stretch Power Uprate ~4.5% (Unit 1/Unit 2)	February 1995/April 1994
MUR Power Uprate ~1.5%	July 2001
LRA Submitted	September 2006
Extended Power Uprate ~14%	January 2008
License Expires (Unit 1/Unit 2)	July 2022/March 2024

Operating Experience Review

- Operating Experience Review Per the Guidance in NEI 95-10
- EPRI Tools Used as Starting Point to Identify Aging Effects
- Five Years of SSES and Industry Operating Experience Reviewed to Ensure Identification of Aging Effects

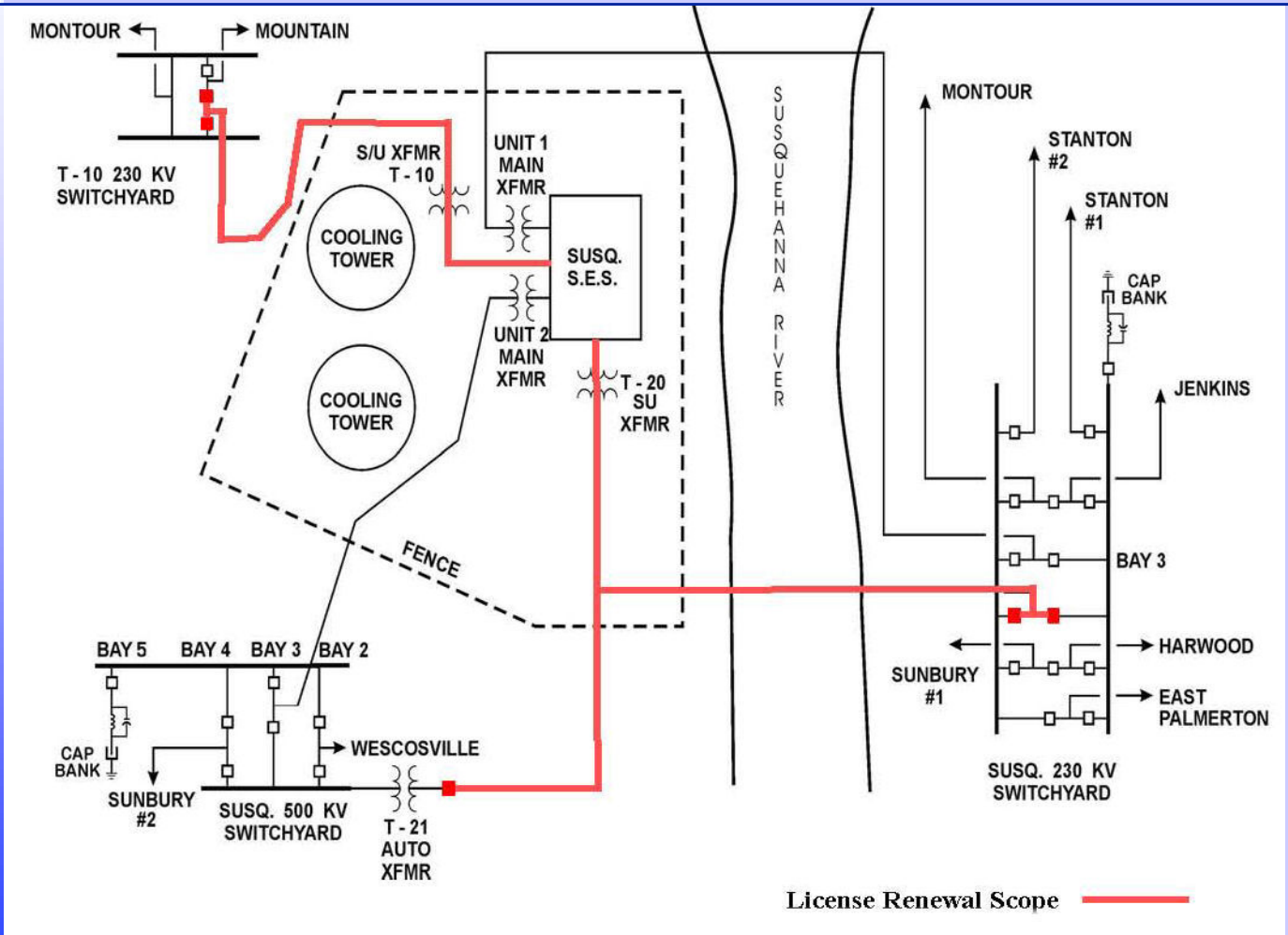
Commitment Process

- 60 Regulatory Commitments for License Renewal
- Entered into the SSES Commitment Tracking Process
- Commitments Assigned to Station Personnel for Implementation

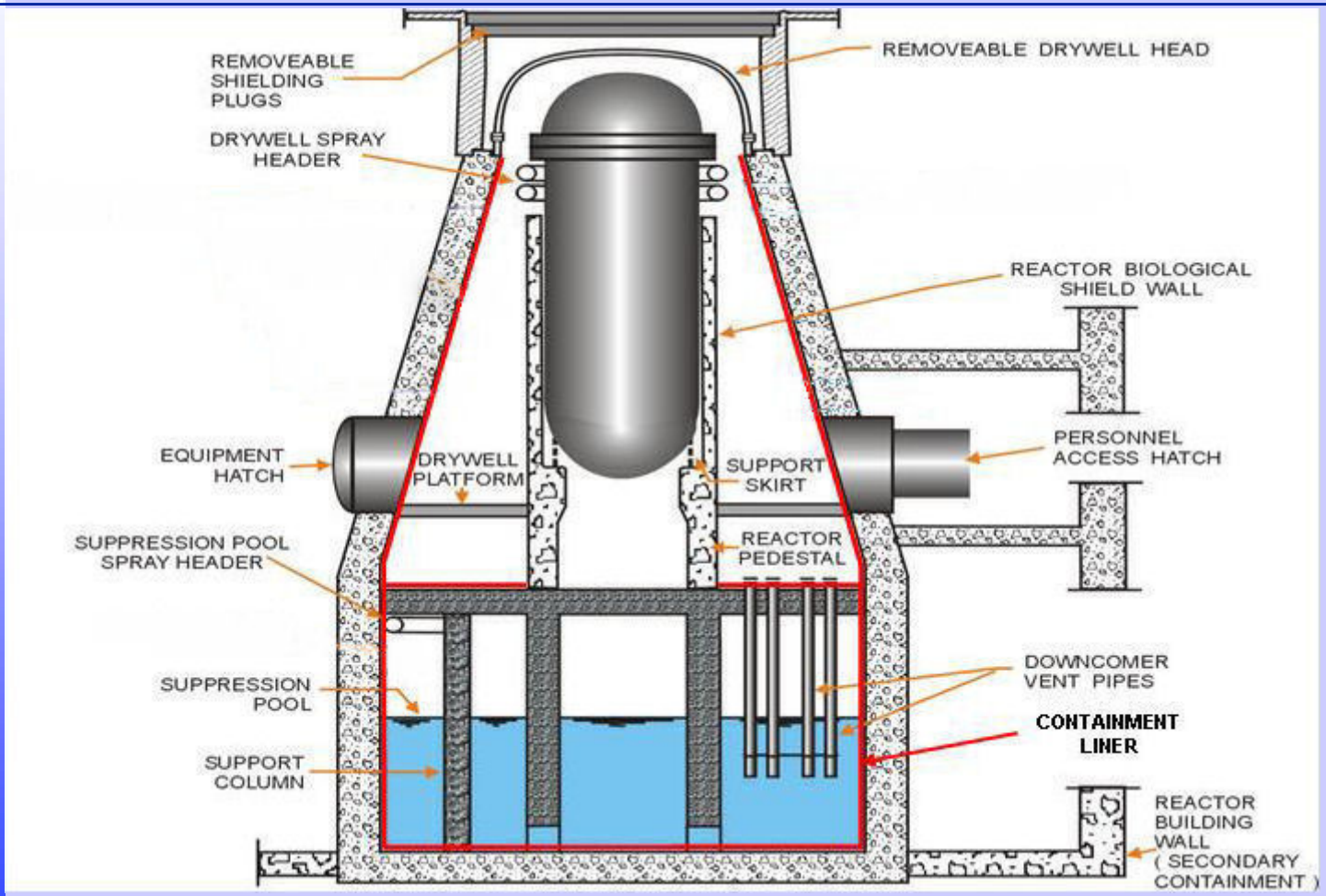
Subcommittee Follow-Up Items Underground Medium Voltage Cables

- 5 kV Safety-Related Cables – No Submerged Cables Observed
- 15 kV Non Safety-Related Cables – Submerged Cables Observed in 2 Manholes
  - License Renewal Commitment Consistent with GALL
    - Test Underground MV Cables
    - Inspect/Pump Manholes to Keep MV Cables from Exposure to Standing Water
  - Testing/Pumping Preventive Maintenance Activities In Place Today

Subcommittee Follow-Up Items
Station Blackout Recovery Scoping



Subcommittee Follow-Up Items
Condition of Containment



MARK 2 CONTAINMENT

Subcommittee Follow-Up Items Condition of Containment

- Nominal 6 Foot Reinforced Concrete – Pressure Retaining Structure
- 1/4 Inch Steel Liner – Leaktight Membrane
- Visual Inspections Under ISI Program per ASME Section XI
- Pressure Tested in Accordance with 10 CFR 50 Appendix J

Subcommittee Follow-Up Items Condition of Containment

- Inspections Have Not Identified Any Significant Degradation Over the Past 25 Years
- Pressure Testing has Not Identified Any Indication of Degradation
- Programs are Effectively Managing Aging of Containment – No Degradation of Design Function

Subcommittee Follow-Up Items Main Steam Line Flow Restrictors

- Potential Loss of Material Due to Corrosion and Erosion Evaluated
- Loss of Material Due to Corrosion Managed by BWR Water Chemistry Program
  - Dry Steam Environment Not Conducive to Corrosion
  - One-Time Inspection of Bounding Components to Verify Effectiveness of BWR Water Chemistry Program
- Erosion Evaluated as TLAA
  - Dispositioned Per 10 CFR 54.21(c)(1)(ii) – Projected for 60 Years
  - Dry Steam Environment and Resistant Material

Summary

- LRA Conforms to Regulatory Requirements and Follows Industry Guidance
- Thorough Review of the Application and Supporting Documentation
- No Open Items
- SSES Will Manage Aging in the Period of Extended Operation

Advisory Committee on Reactor Safeguards

Susquehanna Steam Electric Station Units No. 1 and 2 Safety Evaluation Report

October 8, 2009

Evelyn Gettys, Project Manager
Office of Nuclear Reactor Regulation

Introduction

- Overview of SSES license renewal review
- Section 2: Scoping and Screening Review
- License Renewal Regional Inspection
- Section 3: Aging Management Program and Review Results
- Section 4: Time-Limited Aging Analyses (TLAAs)
- Conclusion

Overview

- LRA Submitted by letter dated Sept 13, 2006
- GE Boiling Water Reactor (BWR), Mark II containment
- Jan. 30, 2008 - Extended Power Uprate (EPU) was granted to SSES to operate at
  3952 MWth, 1300 MWe for Units 1 & 2
- Operating license for Unit 1- NPR-14 expires July 17, 2022
- Operating license for Unit 2- NPR-22 expires March 23, 2024
- Located approximately 7 miles NE of Berwick, PA

Overview

Recap of the April 1, 2009 ACRS subcommittee meeting

- SER issued March 2009
- 278 RAI's
- 59 Commitments
- No Open or Confirmatory Items

Overview

Subsequent to subcommittee meeting

- 8 RAIs issued:
  - 5 Boral
  - 3 Metal Fatigue
- 1 Commitment

Section 2 – Conclusion for Scoping and Screening

- The applicant's scoping and screening methodology is consistent with the requirements of 10 CFR 54.4 and 10 CFR 54.21(a)(1)
- The applicant adequately identified those SSCs within the scope of license renewal in accordance with 10 CFR 54.4(a) and those SCs subject to an AMR in accordance with 10 CFR 54.21(a)(1)

Regional Inspection Conclusions

- Scoping of non-safety SSCs and aging management programs are acceptable
- Inspection results support a conclusion of reasonable assurance that aging effects will be managed and intended functions will be maintained

Section 3.0 Aging Management Review Results

- Section 3.0 – Aging Management Programs
- Section 3.1 – Reactor Vessel & Internals
- Section 3.2 – Engineered Safety Features
- Section 3.3 – Auxiliary Systems
- Section 3.4 – Steam and Power Conversion System
- Section 3.5 – Containments, Structures and Component Supports
- Section 3.6 – Electrical and Instrumentation and Controls System

Section 3.3.2.2.6 Reduction of Neutron-Absorbing Capacity and Loss of Material due to General Corrosion

- Commitment to continue spent fuel pool Boral coupon testing into the period of extended operation
- Water Chemistry Program & Boral Coupon Testing Program

Section 3 Conclusion

- Based on its review of the LRA and additional information submitted as the result of RAIs, the staff concluded that the effects of aging will be managed so that intended function(s) will be maintained during the period of extended operations, per 10 CFR 54.21(a)(3).

Section 4: Time-Limited Aging Analyses

- 4.1 Identification of Time Limited Aging Analyses (TLAAs)
- 4.2 Reactor Vessel Neutron Embrittlement
- 4.3 Metal Fatigue
- 4.4 Environmental Qualification of Electrical Equipment
- 4.5 Concrete Containment Tendon Prestress (N/A)
- 4.6 Containment Liner Plate and Penetration Fatigue
- 4.7 Other Plant-Specific TLAAs

- **Section 4.3.1 – Reactor Pressure Vessel Fatigue Analyses**
  - Requested transient event design specifications and recording of transient data.
  - Applicant responded and indicated that all monitored transient events are bounded by the design specifications and that cycle tracking activities covered the entire operating history of the plant.

Section 4.3.1 – Reactor Pressure Vessel Fatigue Analyses

- Requested justification for the large reductions in projected 60-year CUF values seen for some locations, with relatively minor reduction in cycle projections, even though some transients have projected cycles higher than design cycles
- Applicant provided justification: 40-year design CUF values have incorporated extra conservatism, including contribution from faulted condition loads, directly using results of bounding location, and conservative design cycles.

Section 4.3.3 Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping

- Requested pre –1994 data to supplement the data previously provided and also justification that the DO data is applicable to all NUREG/CR-6260 components
- Applicant provided the DO data, described the water sampling schedule schemes, the water flowing routes, and demonstrated applicability of the sampled data to all of the NUREG/CR-6260 locations considered

Section 4 Conclusion

Based on its review of the LRA and additional information submitted as the result of RAIs, the staff concluded that the applicant provided an adequate list of TLAAs, per 10 CFR 54.3, and that the:

- TLAAs will remain valid for the period of extended operation, per 10 CFR 54.21(c)(1)(i)
- TLAAs have been projected to the end of the period of extended operation, per 10 CFR 54.21(c)(1)(ii); or
- Aging effects will be managed for the period of extended operation, per 10 CFR 54.21(c)(1)(iii)

Conclusion

- The staff has concluded there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB, and that any changes made to the SSES CLB in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations

STEAM GENERATOR ACTION PLAN
OPENING REMARKS

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

OCTOBER 8, 2009

TIMOTHY J. MCGINTY
Director, Division of Policy and Rulemaking
NRC Office of Nuclear Reactor Regulation

STEAM GENERATOR ACTION PLAN BACKGROUND AND OVERVIEW

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

OCTOBER 8, 2009

DAVID BEAULIEU
Project Manager, Division of Policy and Rulemaking
NRC Office of Nuclear Reactor Regulation

Steam Generator Action Plan History

- 1985-1990 NUREG 1150 severe accident risk studies evaluated the issue of “consequential” steam generator (SG) tube rupture.
- For severe accident induced consequential SG tube ruptures, concern was that the high temperature gases created during core damage sequences could cause SG tubes to be the first component of the reactor coolant pressure boundary to fail, resulting in a potential containment bypass and the release of large amounts of radioactive material outside containment.
- NUREG-1150 quantified frequency in the low 10−6/reactor-year range on the basis of expert elicitation.

SG Action Plan History (cont'd) Differing Professional Opinion

- In the early 1990s, the industry made several requests for relaxation of regulatory requirements for SG tube integrity.
- A Differing Professional Opinion (DPO) was filed involving concerns associated with this relaxation given that while eddy-current methods could accurately measure uniform tube wall thinning due to general corrosion, they were less reliable in detecting and sizing cracks that occur in Alloy 600 tubes due to stress corrosion cracking.
- Staff review of those relaxation requests identified that granting them might substantially increase the conditional probability of containment bypass during core damage accidents.

SG Action Plan History (cont'd)

- In the early 1990s, the NRC staff began a study of the effects of severe accident conditions on SG tube integrity as background information for a proposed new rulemaking on SG tube integrity.
- The results from this study, published as NUREG-1570, indicated that the risk is controlled by the current tube integrity requirements to a value that is low enough that no new rulemaking was needed.
- The DPO remained open.

SG Action Plan History (cont'd) DPO Was Referred to ACRS for Resolution

- In 2000, the DPO was referred to the ACRS for resolution.
- After extensive public meetings and review of the issues raised in the DPO, the ACRS published NUREG-1740 to present its conclusions and recommendations.
- In particular, the ACRS concluded that the methodology being used to quantify the risk of containment bypass due to high-temperature challenges to SG tubes was "not technically defensible."

SG Action Plan (Section 3) Created to Address ACRS Recommendations

- Technical staff in NRR and RES jointly reviewed the full text of NUREG-1740 to extract the list of issues that required additional work.
- Those tasks were incorporated into a new section (Section 3) of the SG Action Plan .

SG Action Plan Tasks

- Most, but not all, SG Action Plan tasks involve severe accident induced SG tube ruptures.
- SG Action Plan also includes tasks work that involved design basis events, which addressed the potential for damage progression of multiple SG tubes due to SG depressurization. (e.g., during a main steam line break (MSLB) or other type of secondary side design basis accident).

Design Basis Event Tasks Closed – ACRS Review Complete

- The staff's work to address SG action plan items involving design basis events is complete, and;
- ACRS has previously reviewed and endorsed the closure of these items.
- Basis - Dynamic loads from such design basis events are low and do not affect the structural integrity of tubes or lead to additional leakage or ruptures beyond what would be determined using differential pressure loads alone.

SG Action Plan Status

- The staff has completed its work to close all SGAP items.
- Closeout documentation has been provided to ACRS.
- The purpose of this meeting is for ACRS to review all SGAP items that ACRS has not previous reviewed and closed.

Desired Outcome of ACRS Review

- The staff requests an ACRS letter that finds acceptable the staff's closeout of each SGAP item that ACRS has not previously reviewed and closed which are:
  - SG Action Plan Items 3.1.k, 3.4, 3.5, 3.10, 3.11, and 3.12

Agenda

- Essentially all of today's items are directly related to the work to define the risk associated with severe accident induced SG tube ruptures leading to containment bypass.
- This work involved the following technical areas of research:
  - thermal-hydraulics,
  - steam generator tube material failures;
  - reactor coolant system material failures;
  - component behavior studies, and
  - probabilistic risk assessment

SG Action Plan Closeout
Future Activities Outside of Action Plan Process

- SG Action Plan work is complete and, following ACRS review, the staff would like to close the SG Action Plan.
- NRR User Need to RES is in concurrence -- Requests specific research products to facilitate the development and review of future risk assessments involving consequential SG tube rupture events. These products will build upon analysis methods, tools, and expertise developed as part of the SG Action Plan.
- The RES work to address the NRR User Need no longer requires the level of coordination and agency focus required to implement the action plan process.
- Future work activities associated with this topic will be coordinated using other agency tools such as the User Need and the Planning, Budgeting, and Performance Management processes.

Steam Generator Action Plan Item 3.11

- SGAP Item 3.11 states, "In order to resolve GSI 163, it is necessary to complete the work associated with tasks 3.1, 3.7, 3.8, and 3.9. Upon completion of this task, develop detailed milestones associated with preparing a GSI resolution document and obtaining the necessary approvals for closing the GSI, including ACRS acceptance of the resolution."

- ACRS letter dated May 20, 2009, endorsed the closure of GSI-163.

- The staff requests that ACRS document that ACRS considers the staff's closeout of SGAP Item 3.11 acceptable.

ACRS
Steam Generator Action Plan (SGAP)

October 8, 2009
Rockville, MD

SGAP Items 3.4.a-g
Thermal-Hydraulics
System Code Predictions and CFD

Christopher Boyd RES/DSA

Background

- SGAP section 3.4.a-g
  - thermal-hydraulic behavior of the system during severe accidents

- NUREG/CR-6995 (draft) summarizes the overall project and focuses on the system level predictions. This work improves our understanding of the T/H behavior of the plant during these types of scenarios and addresses criticisms of past analyses.
  - Covers 3.4.a, b, d, f

- Supporting CFD analyses improves our technical basis and helps to extend experimental results to full-scale conditions
  - Covers 3.4.c, e, g
    - NUREG-1781 and NUREG-1788
    - NUREG-1922 (draft)

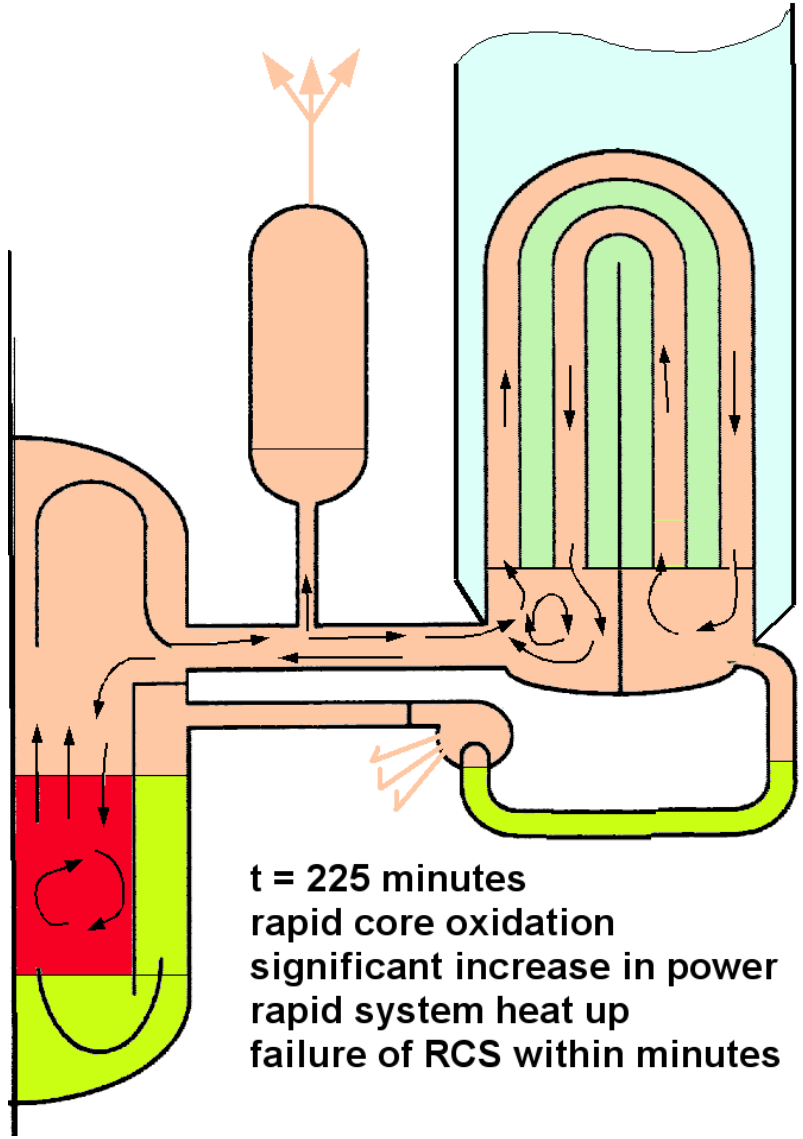
A Fast Scenario

RCS failure within 4 hours

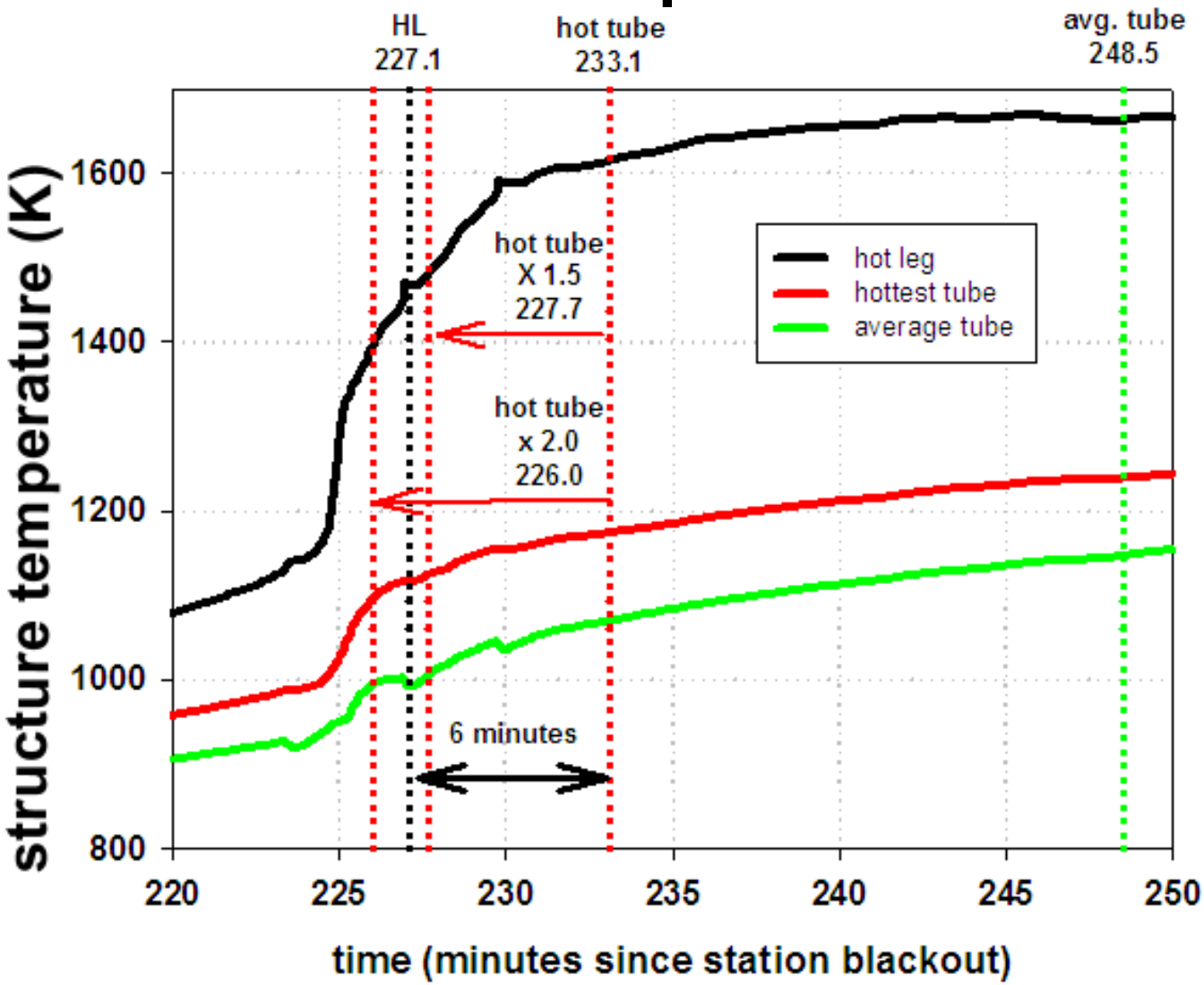
- loss of offsite power, failure of diesel generators, and failure of all auxiliary feedwater
- Reactor coolant pump seal LOCA and secondary side boil off
- secondary system dry out, safety relief valves cycling, primary inventory lost through valves and pump seal LOCA
- loop circulation stops
- Inventory falls below hot legs, natural circulation of superheated steam begins, system heats up

- Core uncovers, core oxidizes and releases significant energy, system heat up accelerates and induced failure is predicted for RCS components.
- High-Dry-Low conditions needed to challenge the SG tubes.

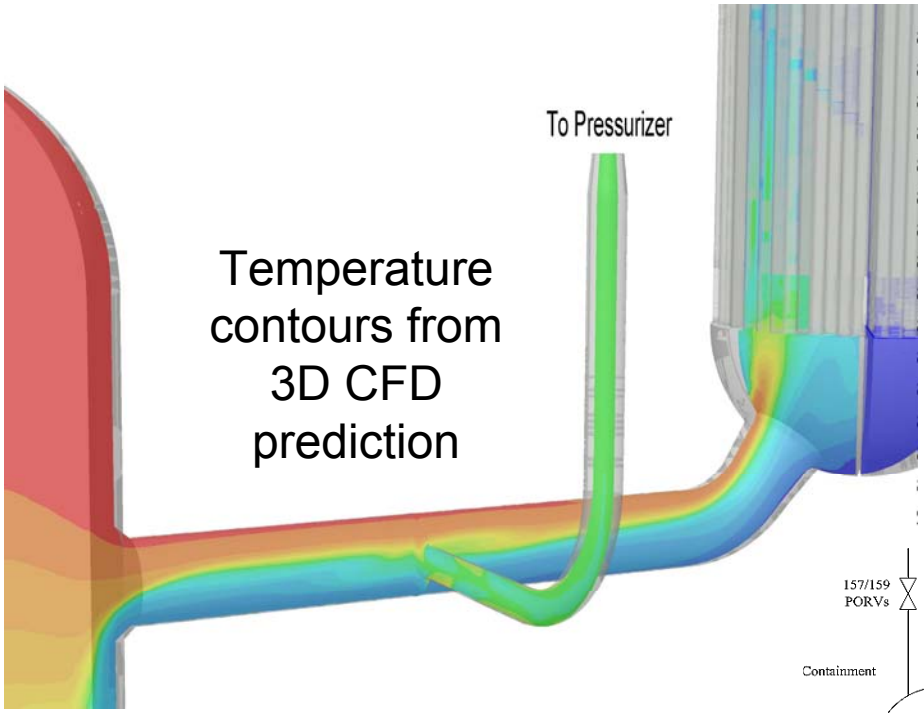
- Clearing the loop seals creates direct circulation flow path and presents a severe challenge to the SG tubes.



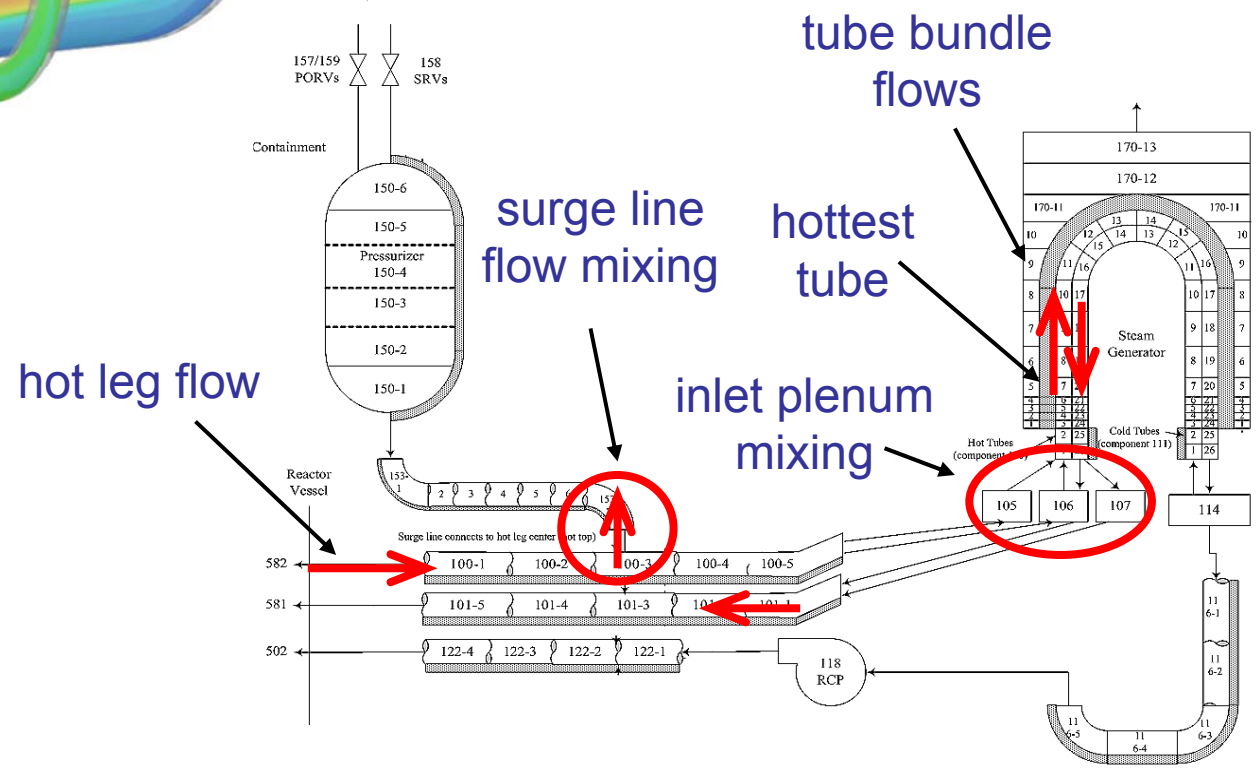
RCS Temperatures



System Model is adjusted to be consistent with 3D Computational Fluid Dynamics and Experiments



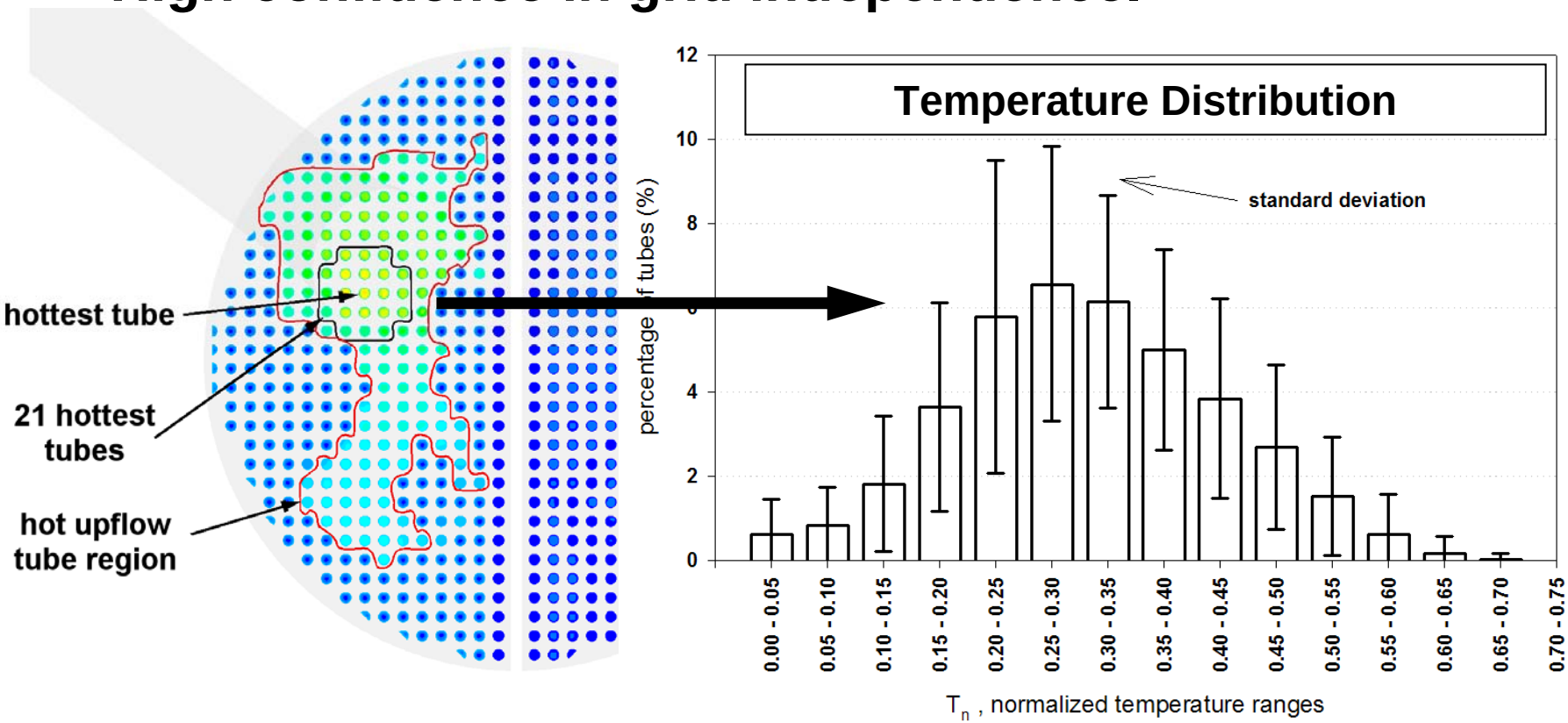
CFD modeling used to extend experimental results to full-scale SGs



CFD Predicts Peak Temperatures

- **Consistent results over 3 different models and with multiple turbulence options.**

- **High confidence in grid independence.**



Summary

- SGAP items 3.4.a, b, d, f are addressed by NUREG/CR-6995.
- SGAP items 3.4.c, e, and g are addressed by NUREG-1781 and NUREG-1788.
  - Specific ACRS concerns are further supported by the updated modeling in NUREG-1922
- Future work will be completed in support of the agencies ongoing research into steam generator tube integrity issues as needed.

STEAM GENERATOR ACTION PLAN
ITEM 3.4h POTENTIAL RCS FAILURE LOCATIONS

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

OCTOBER 8, 2009

T. R. LUPOLD
Branch Chief, Corrosion & Metallurgy Branch
NRC Office of Nuclear Regulatory Research
Timothy.Lupold@nrc.gov

3.4h: Systematic Examination of the Alternate Vulnerable Locations in the RCS.

h.1 Evaluate Creep failure of primary system passive components.

* i.e pressurizer surge line, hot leg
* Consider material properties, residual & applied stresses pressure stress, and presence of flaws

h.2 Evaluate active component failure.

* i.e. PORVs, safety valves, and bolted seals
* Based on operability and “weakest link” considerations

h.3 Determine Feasibility of extending the Rhodes model (RCP seals leakage/failure) to severe accident conditions.

h.4 Conduct large scale tests if needed.

Three phase study:

- Phase I:
  - Reviewed methods & models (failure modes and times-to-failure)
  - Identified additional information needed,
  - Scoped RCS components that might be “weak links”
- Phase II:
  - Developed 3-D computer models of select RCS components (Westinghouse 4-Loop)
  - Utilized detailed plant drawings
  - Included analyses of component operating history
- Phase III:
  - Calculated RCS component failure sequence
  - Utilized improved codes and modeling
    - (RELAP5) Reactor Leak and Power Safety Excursion code
    - (CFD) Computational Fluid Dynamics
  - Expanded high-temperature materials database

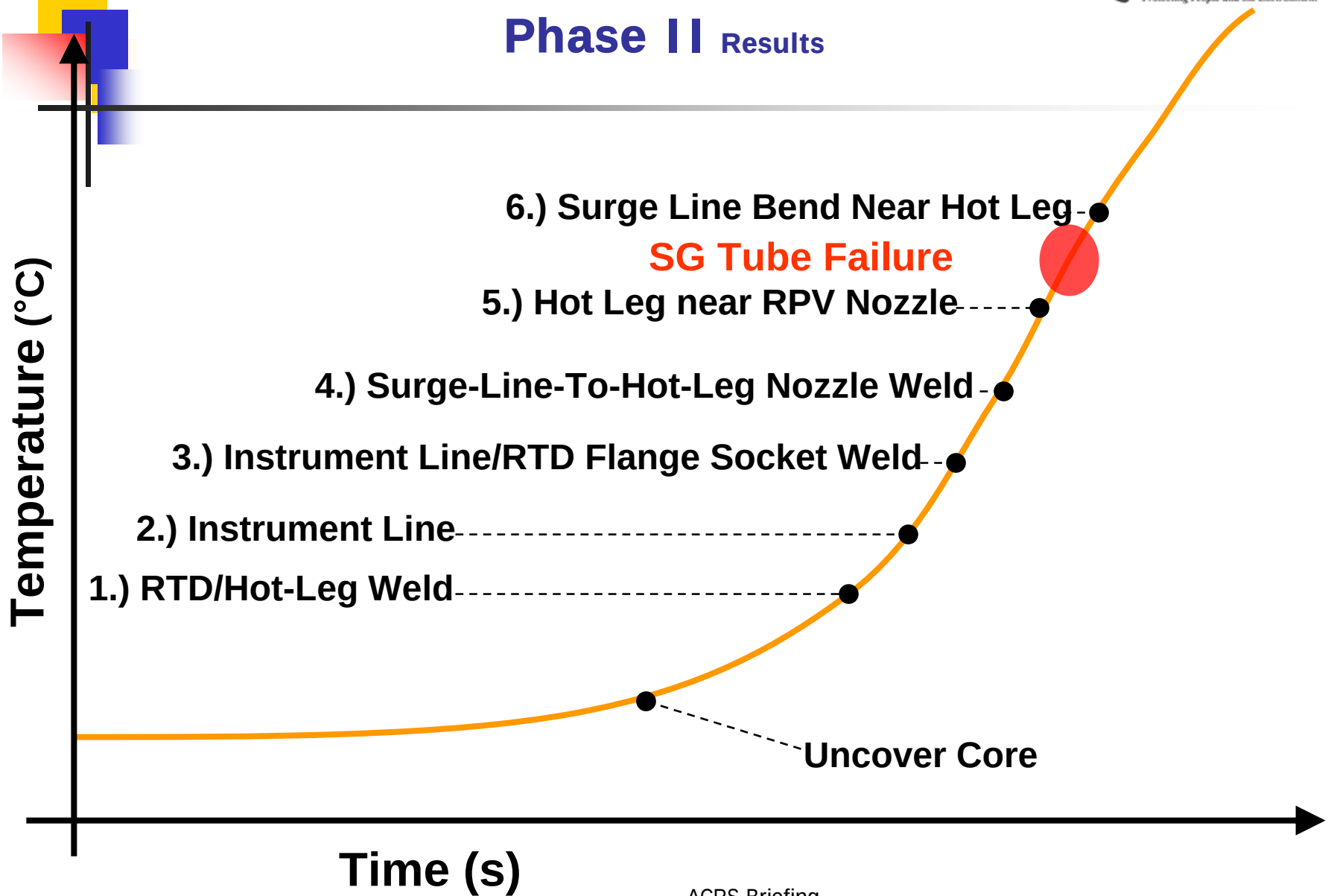
Phase I

- November 2001 Workshop held to discuss expected behavior of non-SGT RCS components and bolted connections during severe accidents in PWRs
  - Workshop concluded it would be possible to analytically predict behavior during severe accidents of certain components
- Following Workshop, non-SGT RCS components and bolted connections were modeled to predict failure times
  - In view of closeness of predicted rupture times for SGTs, surge lines and hot-leg piping reported in NUREG-1570, NRC initiated effort to develop improved models for predicting failure of non-SGT RCS components
- Selected Components for Phase II analysis

Phase II

- Analysis based on Zion Nuclear Station
  - Hot leg and nozzles of Loop 4, which include pressurizer and surge line, were analyzed for reference station blackout (SBO) severe accident transient that involved "high-dry" sequence
  - Results from RELAP5 thermal hydraulic analysis of surface heat flux were used as input for thermal-conduction and stress-strain analyses
  - Failure times due to tensile and creep rupture were calculated with data from literature when available,
    - developed or extrapolated data were only available at lower-than-severe accident temperatures
  - Sensitivity analyses were conducted to determine variability of predicted failure times
    - due to variations of surface heat flux, thermal conductivity, creep rate, and yield strength
  - Also analyzed was stress-strain response due to repeat plug-to-seat impact of typical PORV

Phase II Results



Phase III

- Improvements were made to the thermal hydraulic modeling
  - Refinements were made to surge-line-to-hot-leg connection in RELAP5 model
  - Thermal hydraulic data calculated using RELAP5 were improved to account for entrance effects and flow reversals during PORV cycling
  - High-temperature materials database was expanded by conducting high-temperature tensile and creep tests on stainless steel and carbon steel weldments.
- Enhancements changed calculated failure sequence
  - Resulted in hot leg failing first
  - Suggested that RCP seals could possibly fail prior to SGTs
- Expert Workshop was held to evaluate new findings
  - It was agreed that seal failure could occur sooner than previously estimated and could possibly avert or mitigate containment bypass
    - Based on judgment and experience with RCP seals

Conclusions

- Research accomplished overall goal of developing improved models for evaluating non-SGT PWR RCS components under severe accident conditions
- Determined that differences in times-to-failure between non-SGT (except RCP seals) RCS components were relatively close to each other for transients evaluated
  - Failure times were only nominally affected by variations in heat transfer entrance effects and other sensitivity variations
  - Analysis results for some components (e.g., RTD welds, primary manway, instrument line welds) are sufficiently generic to be applicable to other Westinghouse plants
  - Analyses related to hot leg and surge line may also be applicable to other plants provided relevant dimensions and constraints are comparable
- Determined that RCP seals could fail prior to SGTs, which could avert or mitigate containment bypass
  - Quantifying RCP seals time-to-failure with adequate confidence would require additional experimental and analytical studies
- NRR and RES looking at follow-on research

Steam Generator Action Plan Item 3.10

- 3.10 - To address concerns in the ACRS report regarding our current level of understanding of stress corrosion cracking, the limitations of current laboratory data, the difficulties with using the current laboratory data for predicting field experience (crack initiation, crack growth rates), and the notion that crack growth should not be linear with time while voltage growth is, the following tasks will be performed:

Conduct tests to evaluate crack initiation, evolution, and growth. Tests to be conducted under prototypic field conditions with respect to stresses, temperatures and environments. Some tests will be conducted using tubular specimens.

Using the extensive experience on stress corrosion cracking in operating SGs, and results from laboratory testing under prototypic conditions, develop models for predicting the cracking behavior of SG tubing in the operating environment.

Based on the knowledge accumulated on stress corrosion cracking behavior and the properties of eddy current testing, attempt to explain the observed relationship between changes in eddy current signal voltage response and crack growth.

Staff Closure of SGAP 3.10

- Not based on a specific ACRS recommended action in NUREG-1740
- Operating experience indicates that plant practices for predicting flaw initiation and growth have been effective in ensuring tube integrity consistent with our assumptions in risk analyses
- NRC staff monitors plant operating experience through the inspection process and our review of licensee reports on the results of their steam generator tube inspections. If analysis of future operating experience or research indicates the need to revisit this area, it will be considered and prioritized consistent with our budget process.

SGAP TASK 3.1K
SGAP TASKS 3.4J AND 3.4K
SGAP TASK 3.12

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
OCTOBER 8, 2009

ROBERT PALLA, NRR/DRA

Task 3.1k – Based on Tasks 3.1a – 3.1j, evaluate the probability of multiple tube failures in risk assessments for SG tube ARC

- Objective of Task 3.1k – develop probability distribution for total SG leakage under ΔP loads alone
- The need for this calculation was diminished for several reasons
  - Postulated phenomena associated with depressurization (addressed in Tasks 3.1a – 3.1j) did not prove to be realistic
  - Performance-based TS provides reasonable assurance that DBA leakage will be small and well within that assumed in risk studies
  - Replacement SGs result in fewer flawed tubes left in service and fewer proposals to increase allowable leakage
- The calculations planned under Task 3.1k are not needed to support closeout of GSI-163
- This task can be closed

Task 3.4j – Develop probability distribution for rate of tube leakage for ARC applied to flaws in restricted places

- Task 3.4i provided predicted flaw areas and leak rates from cracks under the TSP during MSLB and severe accidents
- As part of Task 3.5 SNL/SAIC developed a methodology to compute the probability of tube failure during an accident which uses the models developed in Task 3.4i
- The SNL/SAIC methodology can be used to assess the impact of alternate assumptions or models for flaws in restricted places
- This effort has achieved the intent of SGAP Task 3.4j
- This task can be closed

Task 3.4k – Integrate information provided by Tasks 3.4a – 3.4j & 3.5 to address ACRS criticisms on risk assessments for ARC

- Specific concern: ARC that credit "indications restricted against burst"
- Concern was specific to South Texas Project unit with stainless steel drilled-hole support plates (the only SGs of this type in US)
- To limit displacement, tubes were expanded at various elevations
- Staff estimated conservative leak rate of 5 gpm per burst tube within TSP region
- Result could be included in a risk calculation but was not pursued because South Texas Project SGs have been replaced

Task 3.4k (Continued)

- Broader concern: Other SG tube integrity and licensing issues related to flaws in the free-span of SG tubes, and the ability to perform severe accident calculations in a technically defensible manner
- SGAP Task 3.5, “Develop improved methods for assessing the risk associated with SG tubes under accident conditions” was specifically intended to address this concern
- Although additional research related to C-SGTR is planned, the work completed has achieved the intent of SGAP Task 3.4k
- This task can be closed

Task 3.12 – Review Insights from Task 3.5 and Assess Need for Completing Additional Regulatory Guidance

- Staff has assessed the need for guidance on SGTR, and concludes additional guidance and tools are still needed
- Decision rationale
  - Task 3.5 and other studies have not generically dispositioned the issue. Plant-specific PRAs should continue to address C-SGTR
  - Limitations of current work restrict its usefulness in supporting future risk assessments (flaw distributions, TH, documentation & tools)
  - Alternative methods have been developed by industry and are being used by licensees but have not been reviewed by NRC
  - Effectiveness of the peer review process in assuring technical adequacy of this PRA element is not clear
- Development of this guidance will be part of an RES User Need now in concurrence

User Need on C-SGTR

- Additional TH analysis
  - CFD and system code TH analyses for CE plants
  - Impact of incore instrument tube failure on C-SGTR
- Updated flaw distributions and RCS structural analyses
  - Distributions for remaining alloy 600 SGs and replacement 690 SGs
  - Finite element analyses of RCS components
- Guidance and tools for future risk assessments
  - Simplified tools and supporting documentation
  - Reassessment of $P_{C\text{-}SGTR}$ based on updated TH and flaw distributions
  - RG on RI-decisionmaking related to C-SGTR
- A document compiling/summarizing key research

A RISK ASSESSMENT OF CONSEQUENTIAL STEAM GENERATOR TUBE RUPTURES

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

OCTOBER 8, 2009

Dr. Selim Sancaktar
Probabilistic Risk Assessment Branch
NRC Office of Nuclear Regulatory Research

Task 3.5 – Main Objectives

- Develop improved methods for assessing the risk of consequential SG tube rupture (C-SGTR) (3.5.d)
  - Extend method to CE plants (3.5.e)
  - Extend method to external events and low power/ shutdown (3.5.f)
  - Extend method to include consideration of Main Steam Line Breaks (MSLBs) (3.5.g)

- Demonstrate the method with illustrative calculations of the frequency of containment bypass at example plants

General Approach

- Method should be straightforward and scrutable
  - Previous approaches to model C-SGTR have relied on intrusive changes to the PRA
- Reflect current understanding of factors leading to C-SGTR
  - The conditions and sequences that can lead to C-SGTR are well studied and understood
  - Consider plant specific features (SG tube flaw distributions, mitigative credit)
- Utilize Agency Standardized Plant Analysis risk (SPAR) Models
  - Represents as-built, as-operated plant and verified through benchmarking and review activities
- Information and data available at the time of performance of tasks is used – considered prudently conservative

Analysis Method

- A straightforward systematic method based on the sequence results from a PRA is used:
  1) Identify sequences that challenge SG tubes
  2) Apply mitigation credit (i.e. probability that the operators mitigate the conditions leading to C-SGTR)
  3) Apply conditional C-SGTR probability (i.e. probability that SG tubes do not fail given the conditions)
  4) Calculate containment bypass frequency.
- Scope of the method does not include LERF analysis; the measure of C-SGTR importance calculated is containment bypass frequency.
- Intended to provide a method for NRR to assess C-SGTR risk: can be incorporated into a formal guidance document

Analysis Method – Identify Sequences

- Identify sequences that can lead to a C-SGTR by performing the following steps:
  1) Sort core damage sequences by their contribution to CDF
     - Identify risk-significant sequences that lead to High-Dry-Low conditions
     - Typical scenarios involve failure of AFW system and RCS feed and bleed
  2) Identify sequences that represent a pressure challenge to SG tubes (including otherwise successful sequences)
     - ATWS
     - MSLB
     - LOCA

     May require some simple modeling changes to PRA model
- Method should consider external events and shutdown initiators as appropriate

Analysis Method – Apply Mitigation Credit

- Given that a core damage event has occurred, what is the probability that recovery actions to mitigate the plant conditions leading to a C-SGTR will fail?
  - Actions generally involve restoration of secondary cooling or RCS depressurization
  - Any standard HRA method may be used – screening values are used in the report
- Recovery may not be feasible in certain situations (e.g., fast developing scenarios)
- Highly plant specific evaluation
- Certain strategies may rely on security related actions
  - Credit should conform to Regulatory Issue Summary 2008-15, "NRC Staff Position on Crediting Mitigative Strategies Implemented in Response to Security Orders in Risk-Informed Licensing Actions"

Analysis Method – Apply SG Tube Failure Probability

- A C-SGTR Tube Failure Probability Calculator Integrates thermal-hydraulic and materials information
  - Flaw distributions
  - Pressure and temperature history
  - Material properties
  - Failure times for alternate RCS locations
- C-SGTR probabilities generally fall into four bins:
  - High probability (0.4)
  - Low probability (0.02)
  - Negligible probability (0.001)
  - Unanalyzed (0.5)

Illustrative Examples

- The quantitative risk assessment method and its application to multiple plants are in a technical PRA report prepared by RES.
- Method applied to
  - 4-loop Westinghouse (Plant 1)
  - 2-loop Westinghouse (Plant 2)
  - 2-loop Combustion Engineering (Plant 3)
- Includes risk from internal and other hazard categories (“external events”)
- Includes C-SGTR caused by initiating events, as well as RCS conditions after core damage sequences
- Examples of three dominant C-SGTR sequences are given in the backup slides.

Plant-1 4-Loop Westinghouse single unit on a site - Results

Event Category	CDF	Bypass frequency due to C-SGTR	Bypass Freq. as % of Total Plant CDF
Internal Events at Power	2.0E-05	9.0E-07	1.55%
Internal Flooding Events	3.9E-06	1.9E-07	0.33%
Internal Fire Events	3.4E-05	2.0E-07	0.35%
Large Steam Line Break Events	2.5E-08	1.2E-08	0.02%
Seismic Events Bin 1 (pga 0.05 to 0.3g)	2.7E-08	1.3E-09	0.00%
Seismic Events Bin 2 (pga 0.3 to 0.5g)	7.9E-08	1.3E-08	0.02%
Seismic Events Bin 3 (pga > 0.5g)	6.9E-07	3.0E-07	0.52%
Totals	**5.8E-05**	**1.6E-06**	**2.8%**

Conclusions

- Key insight - the fraction of CDF from potential C-SGTR sequences is lower or at the same order of containment bypass fraction that may exist for the currently postulated containment bypass fraction of internal events (e.g. 0.1 or less)
  - C-SGTR is not a negligible part, nor a main contributor of total plant risk. It should be considered and monitored in plant risk assessments in a manner commensurate with its expected importance for each plant.

- An otherwise “successful” accident sequence turning into a core damage sequence due to C-SGTR is not a major contributor to plant risk. However, an accident sequence initially progressing as core damage turning into a containment bypass sequence due to C-SGTR may not be negligible.

Recommendations

- Plant PRAs should address C-SGTR in their evaluation of plant LERF or level 2 analyses on a plant-specific basis, referring to the existing PRA standard. However, an in-depth and intrusive modeling of C-SGTR within the current level 1 models may not be necessary.

- Results sensitive to degree of credit allowed for mitigative actions

- Care must be taken not to take excessive credit for SAMG actions, other recovery actions, and non-safety-related equipment to avoid C-SGTR, especially for external event scenarios.

Programmatic recommendation:

- Task 3.5 of the SGAP has been completed and should be closed.

BACKUP SLIDES

An Internal Event C-SGTR Sequence for Westinghouse 4-Loop Plant

- A loss of offsite power initiating event occurs and results in station blackout; Initially AFW system is operable via the turbine-driven AFW pump; rapid secondary depressurization is successful; 21 gpm per pump RCP seal LOCA occurs; recovery off offsite and onsite power in 8 hours fail; loss of AFW pump due to battery depletion is postulated (operator action for pump control is not credited in SPAR models to avoid core damage); core damage is postulated.

- Sequence proceeds as a very Small LOCA without AFW and without feed and bleed. Sequence CDF is 3.35E-06/year; due to availability of long time window, mitigation credit for secondary cooling to avoid C-SGTR is credited as 0.1. The sequence is assigned a high conditional C-SGTR probability of 0.4.

- The estimated C-SGTR frequency is

  F = 3.35E-06 * 0.1 * 0.4 = 1.3E-07/year.

High-pga Seismic Event for W 2-Loop Plant

- Seismic events with high pga (> 0.5 g) causing failure of structures, where core damage is postulated.
  - The frequency of this seismic event for plant-2 is three times the frequency of the corresponding one for plant-1. There are large uncertainties in which equipment is failed and which equipment may be still available. Conservative assumptions may be present in the classification of this “sequence”.
- The initial sequence CDF is 3.8E-06. The large structural failures considered are reactor vessel, steam generators, RCS piping, containment building, turbine building, and auxiliary building.
- 35.4% of the sequence already has either containment bypass or vessel failure. In order to estimate the additional containment bypass risk due to C-SGTR, the original sequence CDF is multiplied by 0.646 (1.0 - 0.354).

High-pga Seismic Event (continued)

- This is a high-pga seismic event beyond design basis:
  - no mitigative strategy is credited (no recovery). No credit is given for B5b equipment.
  - the highest conditional C-SGTR probability of unanalyzed case = 0.5 is used.
    - (For those sequences that are not studied in the Sandia report, a screening value of 0.5, which is very similar to this highest reported conditional C-SGTR frequency, is selected. Use of such a screening value in PRA acknowledges that information about this parameter is not known: however it is neither deemed to occur with certainty, nor it is deemed to be insignificant.)

- The estimated C-SGTR frequency is:

$$F = 3.8E\text{-}06 * (1 - 0.354) * 0.5 = 1.2E\text{-}06/yr.$$

A Fire Event Sequence for W 2-Loop Plant

- Internal fire event that result in evacuation of the main control room and causes LOOP.
- Control of the plant at the dedicated shutdown panel and AFW fails. Both the fire scenario frequency and the equipment that can be credited to deal with the sequence have large uncertainties and possibly conservative assumptions.
- Initial sequence CDF is 3.23E-06.
- Since AFW fails and no safety injection is credited, core damage occurs relatively rapidly. Due to the short available time for the operator actions, use of B5b equipment or AFW recovery are not credited.
- High conditional C-SGTR probability of 0.4 is used. This probability applies to transient or an RCP seal leak of up to 150 gpm per pump (RCS pressurized; no relief or small leak).

- The estimated C-SGTR frequency is:

  F = 3.23E-06 * 0.4 = 1.3E-06/yr.

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Oyster Creek
3D Drywell Analysis

ACRS Full Committee
October 08, 2009

Agenda/Presenters

- ✓ Summary of 3D Drywell Analysis
- ✓ Resolution of Questions from September 23, 2009 Subcommittee Meeting
- ✓ Presenters:
  - Mike Gallagher
  - John O'Rourke
  - Stan Tang
  - Dr. Clarence Miller

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Background

- ✓ Exelon made a commitment to perform 3D Drywell Analysis at the February 01, 2007 ACRS Meeting
- ✓ The ACRS described Exelon's commitment in its February 08, 2007 letter to Chairman Klein as follows:
  - "The applicant has committed to perform a 3D finite-element analysis of the OCGS drywell to determine the margin of the shell in the as-found condition using modern methods. This analysis will provide a more accurate quantification of the margin above the Code required minimum for buckling. …. The analysis should include sensitivity studies to determine the degree to which uncertainties in the size of thinned areas affect the Code margins"
- ✓ Exelon formalized this commitment in a letter dated February 15, 2007
- ✓ The analyses were submitted to the NRC on January 22, 2009.

Overall Results (Sandbed Region)

Load Combination Case	Required Safety Factor	Base Case Safety Factor (Limiting)	Case 1 Safety Factor (Limiting)	Case 2 Safety Factor (Limiting)
		Based on 2006 data confirmed by 2008 data	100 mil local reduction in Bay 1	50 mil reduction of Bay 19
Refueling	2.0	3.54 Bay 3	3.21 Bay 3	3.46 Bay 3
Post Accident Flooding	1.67 (Service Level C)	2.02 Bay 19	2.01 Bay 19	1.98 Bay 19

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Drywell Description

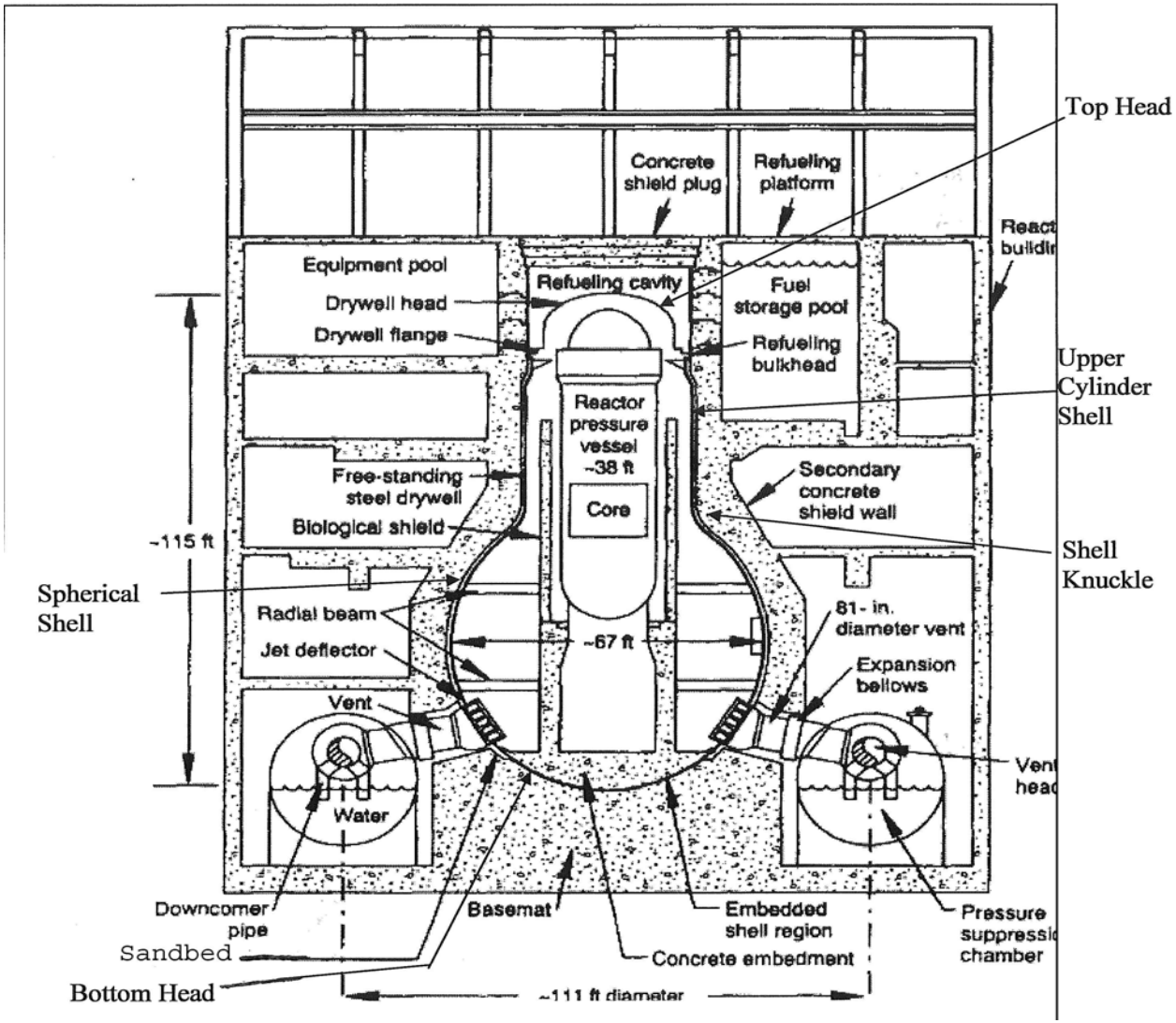

Figure 1-1: Oyster Creek Containment Schematic

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Analysis Summary

- ✓ **Detailed finite element model developed using design drawings**
- ✓ **Shell thicknesses were developed based on UT thickness data**
- ✓ **Current licensing basis inputs such as code of record, loads and load combinations were used**
  - **The Oyster Creek drywell vessel was designed, fabricated and erected in accordance with the 1962 Edition of ASME Code, Section VIII and Code Cases 1270N-5, 1271N-5 and 1272N-5**
  - **For the size of the region of increased membrane stress, guidance sought from the 1989 Edition of ASME Code (including Winter 1991 Addenda), Section III, Subsection NE, Class MC Components**
  - **For the post-accident stress limits, Standard Review Plan Section 3.8.2 was used as guidance**
- ✓ **Buckling analysis performed**
  - **Allowable Stress Values per Section NE-3222 of the 1989 Edition (with Winter 1991 Addenda) of the ASME Code, Section III, Division 1, Subsection NE, Class MC Components**
  - **Modified Capacity Reduction Factor utilized in accordance with Code Case N-284-1 whose primary author was Dr. Clarence Miller**
  - **Code minimum safety factors satisfied**
- ✓ **Stress evaluation performed**
  - **Stresses are within Code allowables**

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Drywell Thickness Description

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Sandbed Measurement Locations

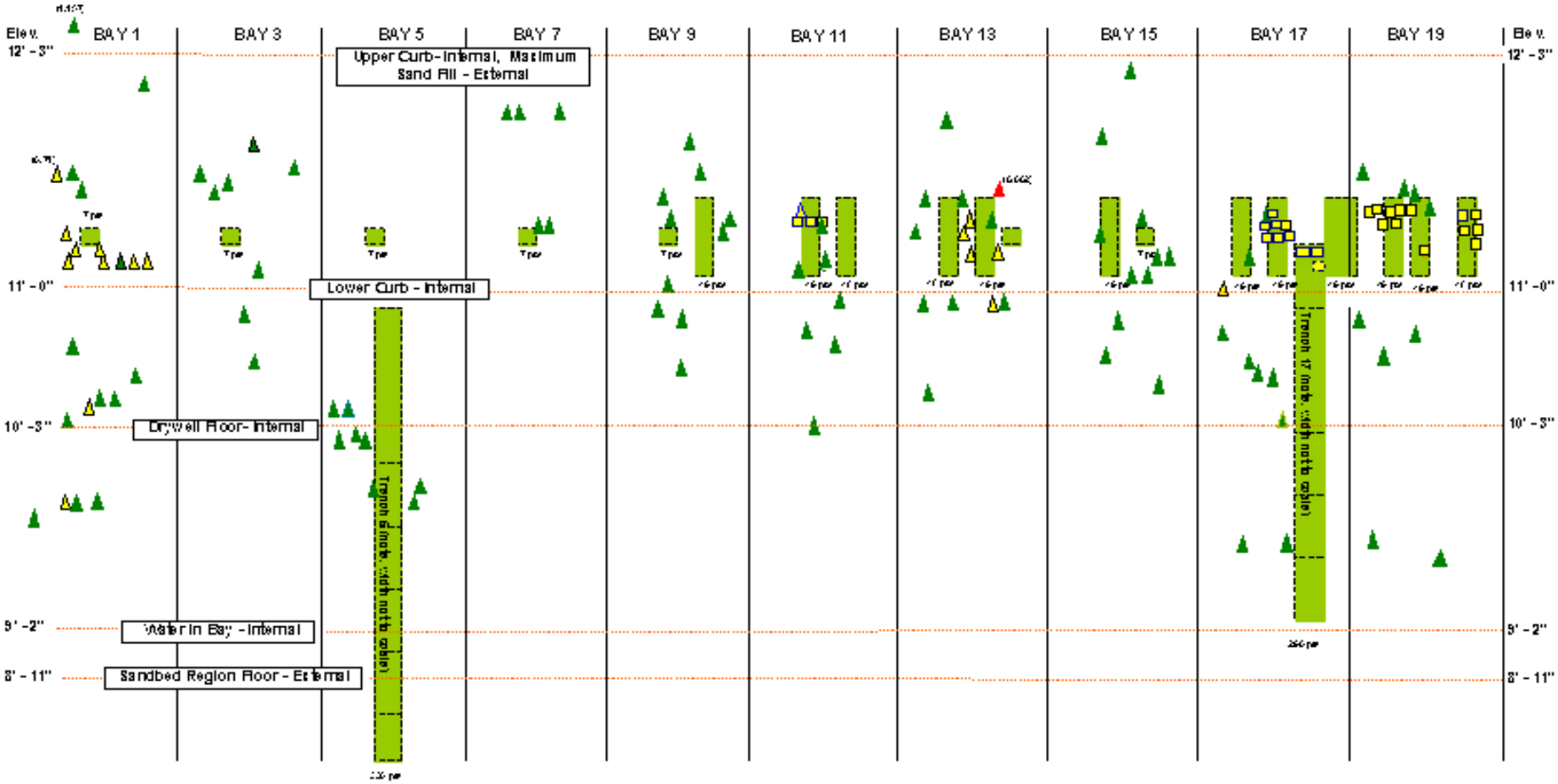
2006 Measurement Locations in the Sandbed Region

Color Code for thickness:

Green =UT Measurements > 736 Mils
Yellow = UT Measurements Between 636 and 736 Mils
Red = UT Measurements Between 536 and 636 Mils

Location / Type of UT Measurement

△ External Point UT Measurements
⛶ Internal Grid UT Measurements
□ Internal Point UT Measurements



Subcommittee Question #1

- ✓ Why is it appropriate to use the UT Internal Grid averages as the modeled thickness for large areas of the sand bed bay?
- ✓ Response:
  - The internal UT grids were located based on a series of data taken to locate the thinnest general areas for ongoing monitoring. Therefore, these grids are representative of the general area average thickness, biased on the thin side.
  - The half buckling wavelength is large, on the order of 5 to 8 feet, which indicates a bay-wide effect. Therefore, the use of average thickness over a bay is acceptable.

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Sandbed Internal UT Measurements

- ✓ From 1983 to 1986, measurements were taken around the inside of the drywell at elev. 11'3"
  - Over 500 initial data points were measured
  - When thin locations were identified, UT measurements were taken horizontally and vertically to locate the thinnest locations
  - UT grid measurements were taken at the thinnest locations
  - 19 locations were selected for ongoing corrosion monitoring with at least one grid is located in each of the 10 bays
- ✓ UT measurements are obtained at these locations every other refueling outage as part of the ongoing aging management program

The UT Internal Grid locations are representative of the general area average thickness, biased on the thin side. Therefore, the UT Internal Grids measurements are appropriate to use as inputs to conservatively model general bay thicknesses.

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Sandbed Trench UT Measurements - Background

- ✓ Trenches in Bays 5 and 17 were excavated in 1986 to characterize the extent of corrosion in sand bed at elevations below the drywell interior floor
  - Bays 5 and 17 were selected because UT measurements indicated these bays had the least and the most corrosion, respectively
  - The trenches extend to about the elevation of the bottom of the sand bed
  - UT measurements taken in the trenches confirmed that the corrosion below elev. 11’ 3” was bounded by the monitoring at elev. 11’ 3”.

This supports our conclusion that the UT Internal Grid measurements are appropriate to use as inputs to conservatively model general bay thickness.

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Use of Average Thicknesses

- ✓ Local Buckling Stresses depend on the applied stress over the half buckle wavelength L_c
  - $L_c = 3.72(Rt)^{0.5}$
  - For the Oyster Creek Drywell Shell, the half buckle wavelength varies between 62 inches and 89 inches
  - For shells with variable thickness, the thickness can be averaged over a distance of, at least, the half buckle wavelength L_c
- ✓ Tests (Miller, 1982) have been conducted on axially compressed shells with unreinforced openings which show little reduction in the buckling strength when the diameter of the hole, d, is $< 0.8(Rt)^{0.5}$. For the Oyster Creek drywell shell d = 13 in. to 18 in.
  - A hole less than 13 inches will have little effect on the buckling strength, therefore, thinned areas less than 13 in. will have little effect on the buckling strength

Conclusion: Use of average thicknesses for a bay as input to the 3D Analysis is acceptable.

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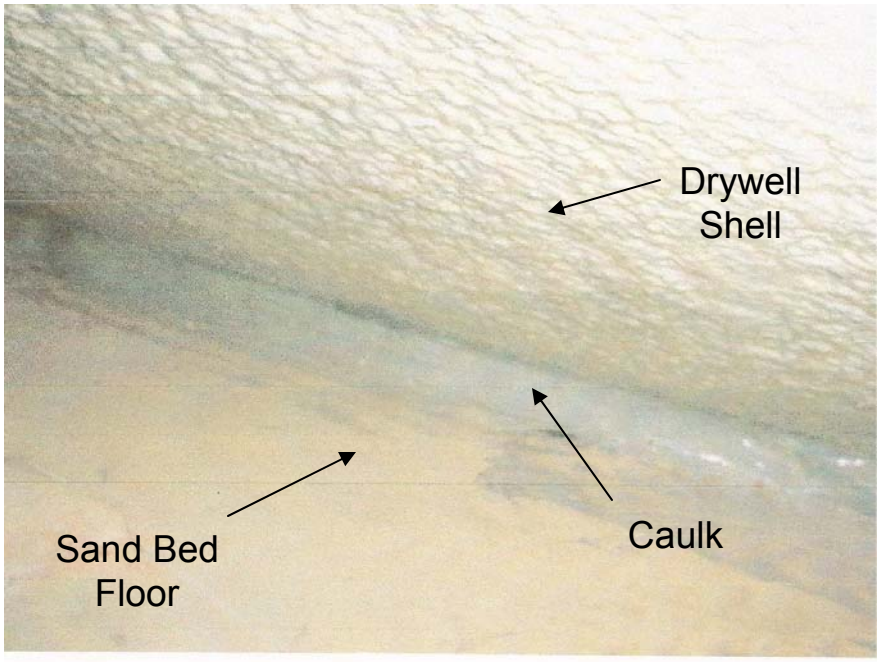
Bay 19 Thickness

- ✓ The bay exhibits historical corrosion above and below Elevation 11'-0"
- ✓ 3 internal grids (All nominally 49 points) are used to measure shell thicknesses in Bay 19
- ✓ Average of three internal grids used as general area thickness for entire bay (826 mils)



Bay 19
(2006)

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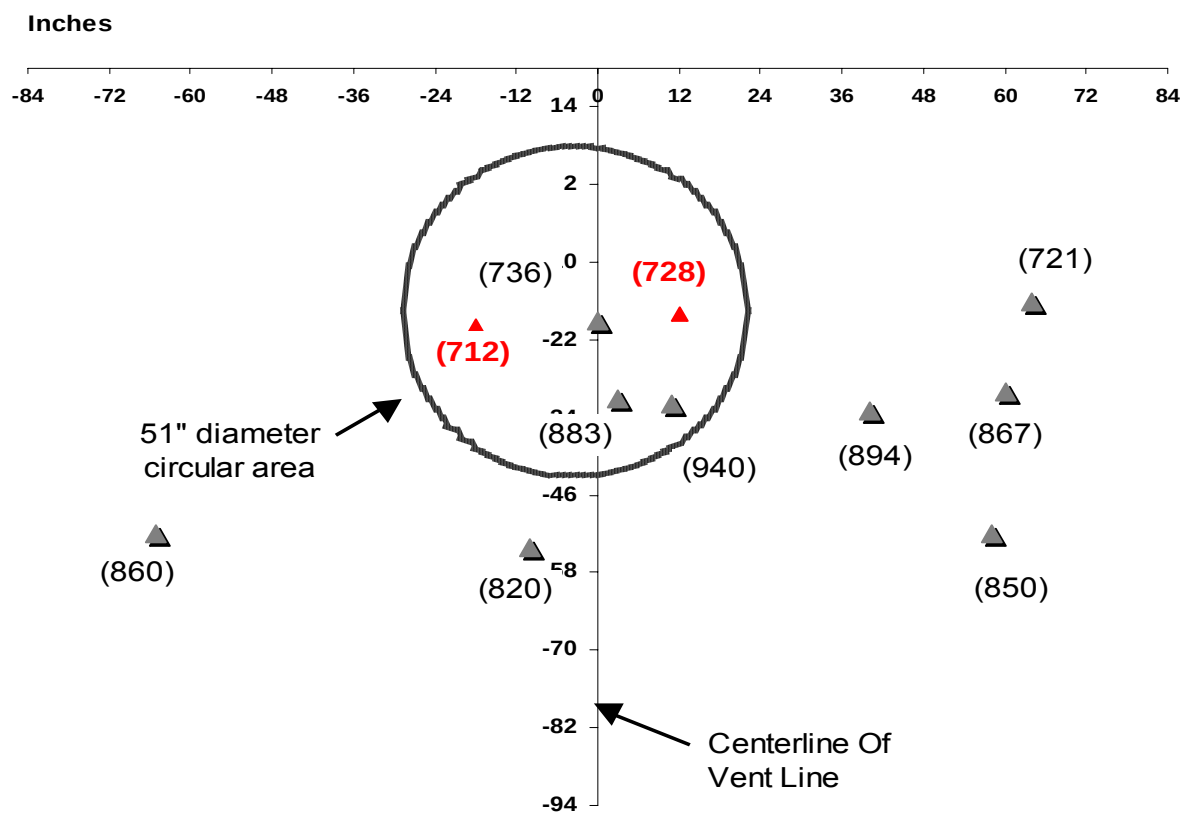
Bay 19 Locally Thinned Area

Bay 19 - Location of 2006 External Measurements
Near Modeled Locally Thin Area (Thickness in mils)



Average of Red locations (720) was input to model
Gray locations were not input to model

Reference
C-1302-187-5320-024

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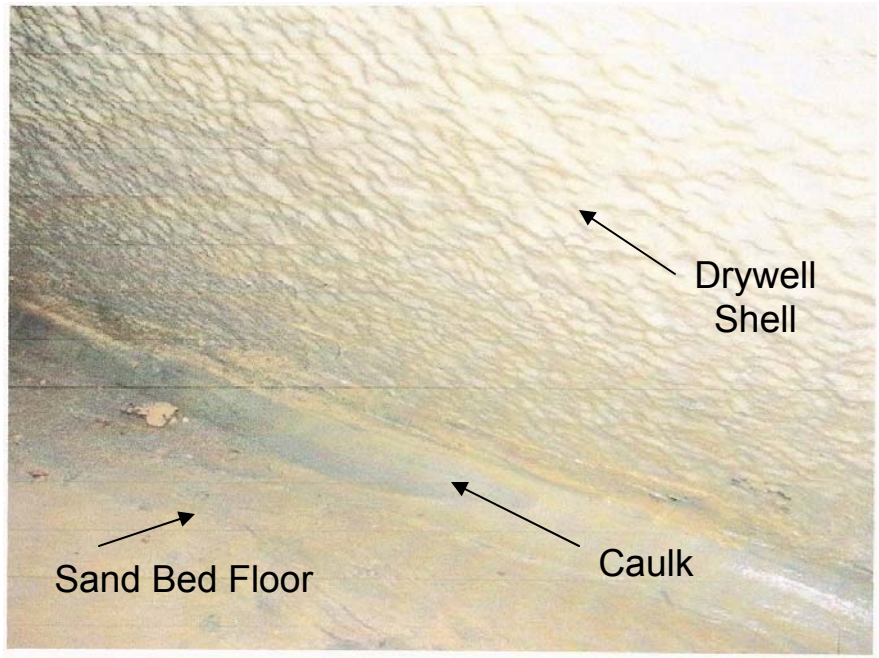
Bay 1 Thickness

- ✓ Visual observation of the external shell surface in Bay 1 indicates the presence of historical corrosion
- ✓ 1 internal grid (7 point) is used to measure shell thickness in Bay 1
- ✓ Internal near nominal grid shell readings for Bay 1 indicate that the grid location is above the former sand/water/air interface boundary
- ✓ Adjacent bay (Bay 19) is one of the most corroded bays
- ✓ Bay 1 thickness was conservatively estimated to be the same as Bay 19 thickness



Bay 1
(2006)

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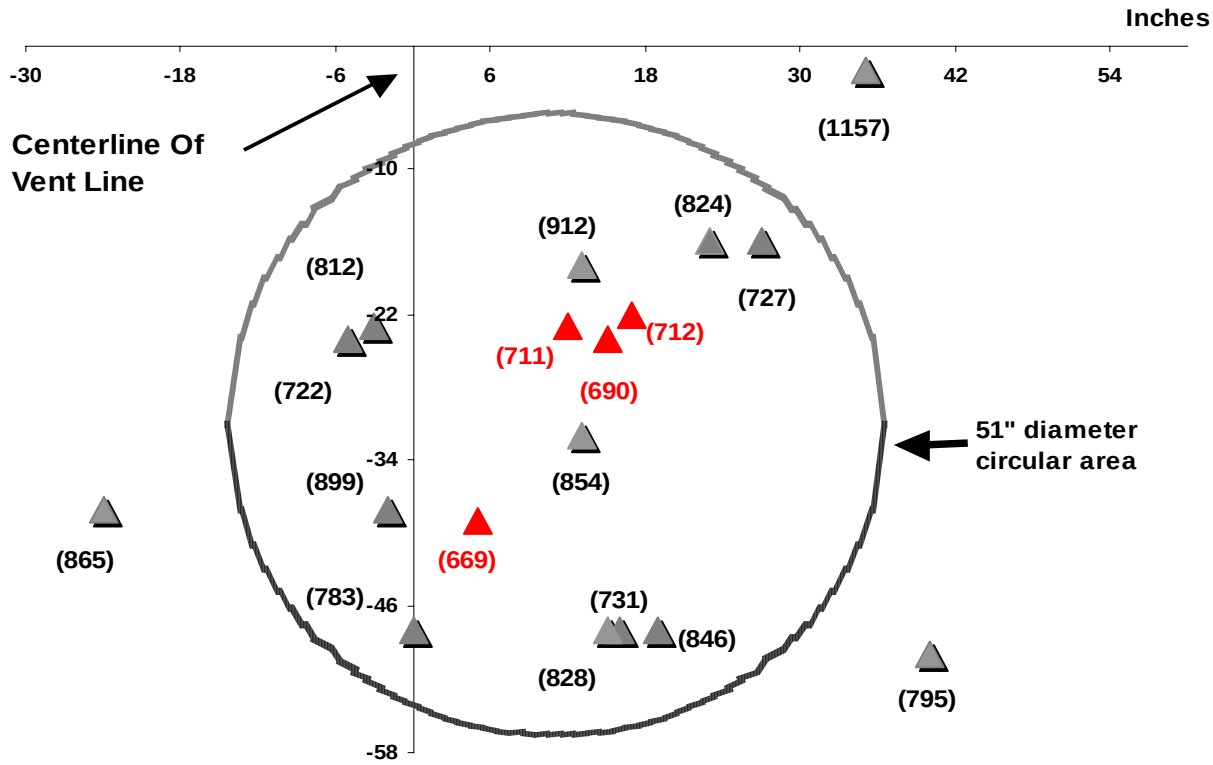
Bay 1 Locally Thinned Area

Bay 1 - Location of 2006 External Measurements
Near Modeled Locally Thin Area (Thickness in mils)



Average of Red locations (696) was input to model
Gray locations were not input to model

Reference
C-1302-187-5320-024

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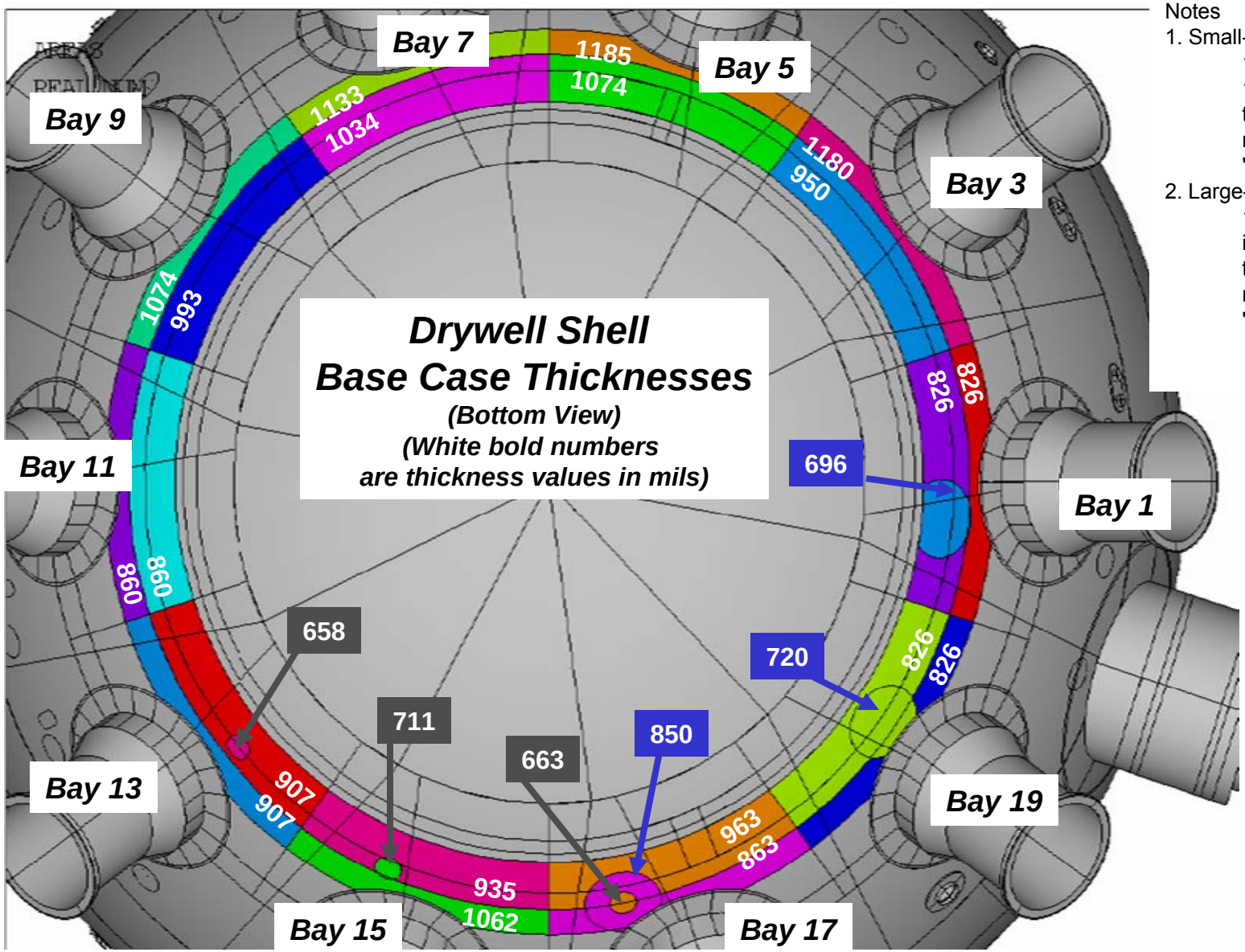

Notes

1. Small-diameter circles in Bays 13, 15, and 17 represent 18-inch diameter locally-thinned areas that were modeled as part of the "base case."
2. Large-diameter circles in Bays 1, 17 and 19 represent 51-inch diameter locally-thinned areas that were modeled as part of the "base case."

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Subcommittee Question #2

- ✓ Why are the External UT points considered biased thin?
- ✓ Response:
  - The External UT Points were identified visually as thin areas for further evaluation. The areas were also ground to permit UT measurement. Therefore, they are biased thin.

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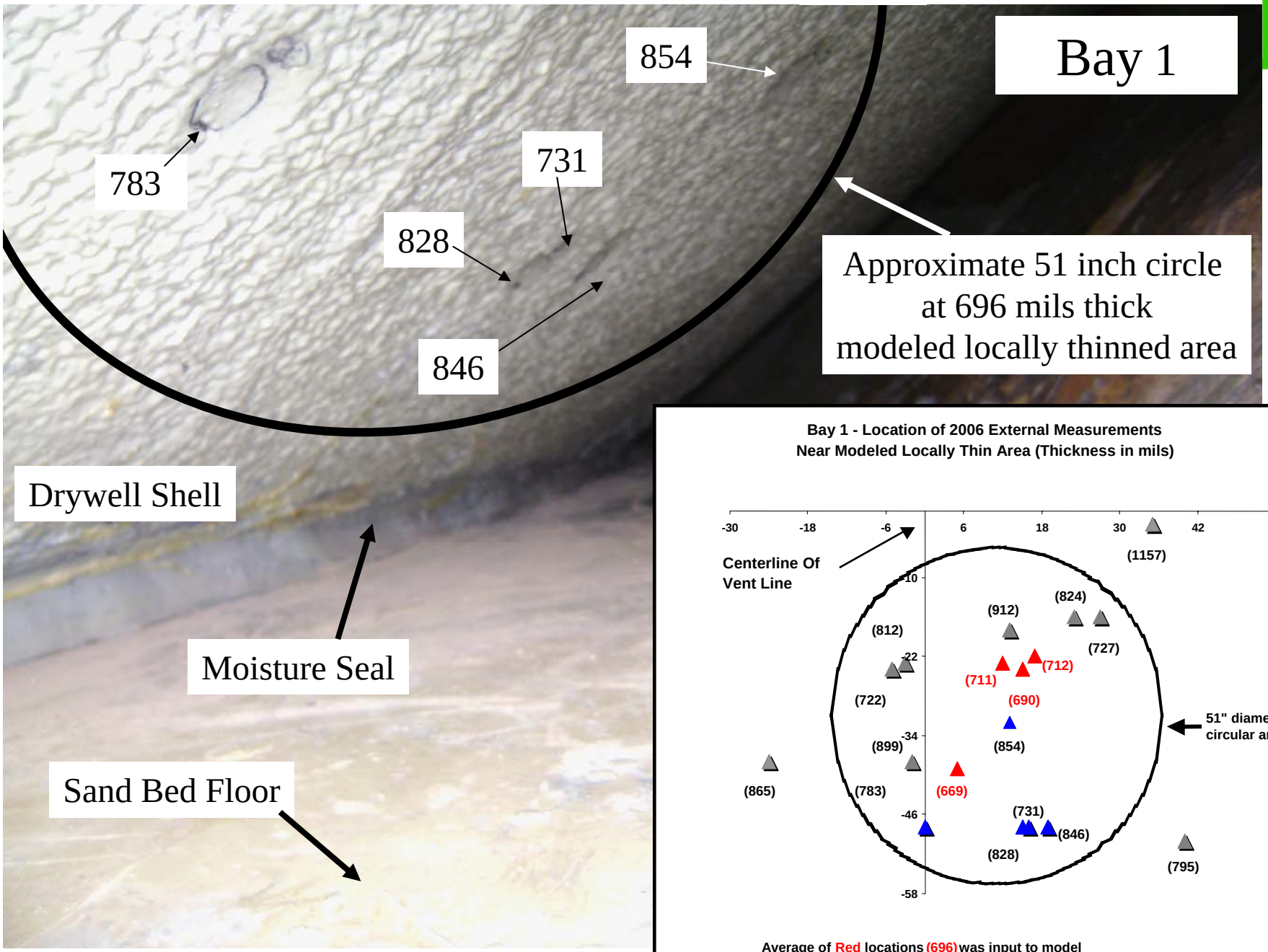
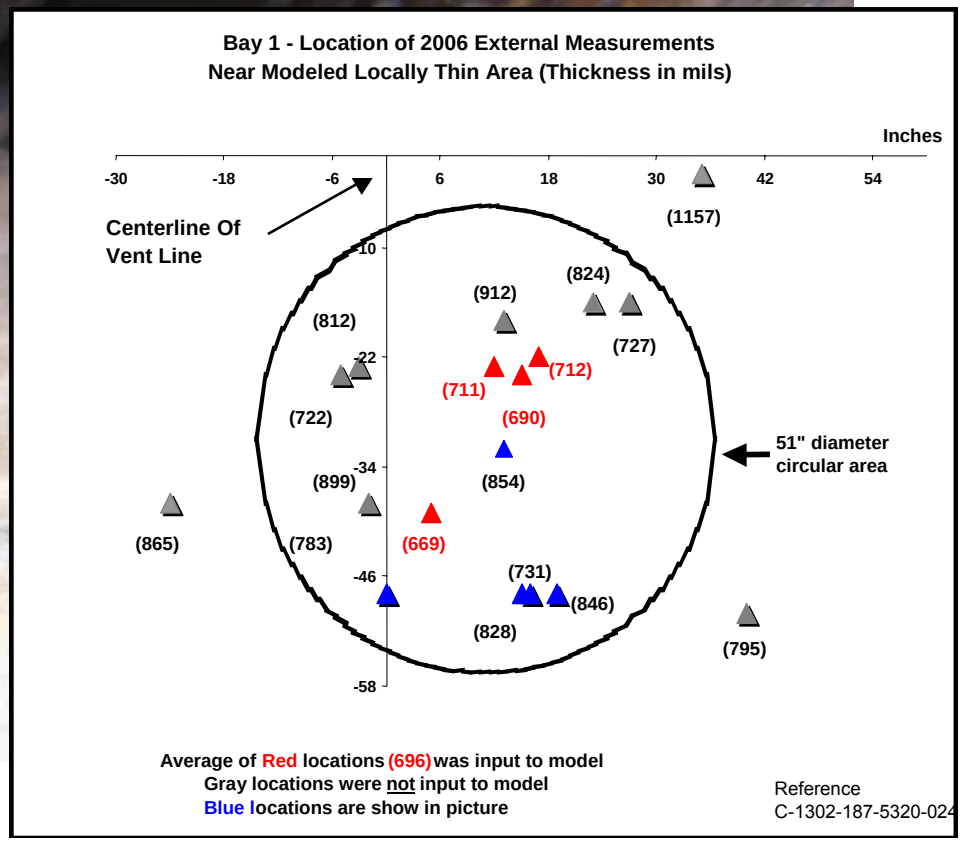
Development of External UT Locations

- ✓ After the sand and corrosion byproducts were removed from the sandbed bays, the thinnest locations in each bay were identified.
- ✓ Each bay was visually inspected and the thinnest local spots were investigated both visually and using UT measurements.
- ✓ Approximately 100 locations were identified and mapped.
- ✓ The external surface of the shell was rough, requiring most of the locations to be ground down to create small flat surfaces so the UT probe could be placed perpendicular to the shell wall.
  - This resulted in small 1 to 2 inch diameter dimples in the surface with relatively thinner local areas (biased thin) that were then measured by UT.
  - Micrometer characterization showed that the grinding process may have removed up to 100 mils.
  - Between the individual External UT points, the shell is thicker.

The External UT points are biased thin. They were used to develop the locally thinned modeled areas. It is not appropriate to use these individual measurements to determine general area thicknesses.

Bay 1


854
783
731
828
846
Approximate 51 inch circle at 696 mils thick modeled locally thinned area
Drywell Shell
Moisture Seal
Sand Bed Floor


Bay 1 - Location of 2006 External Measurements
Near Modeled Locally Thin Area (Thickness in mils)
Inches
-30
-18
-6
6
18
30
42
54
-10
-22
-34
-46
-58
Centerline Of Vent Line
51" diameter circular area
(1157)
(824)
(912)
(812)
(727)
(712)
(711)
(722)
(690)
(854)
(899)
(865)
(783)
(669)
(731)
(846)
(828)
(795)
Average of Red locations (696) was input to model
Gray locations were not input to model
Blue locations are show in picture
Reference
C-1302-187-5320-024

Finite Element Model

Finite Element Model

- ✓ ANSYS
  - Release 8.1 and ANSYS Release 11 used to generate the three dimensional model
  - Industry accepted, finite element analysis software
- ✓ Dimensions and material properties obtained from OEM drawings
- ✓ Model Size
  - Approximately 406,000 elements
  - Approximately 400,000 nodes
- ✓ Mesh Sensitivity
  - Up to approximately 1,000,000 elements

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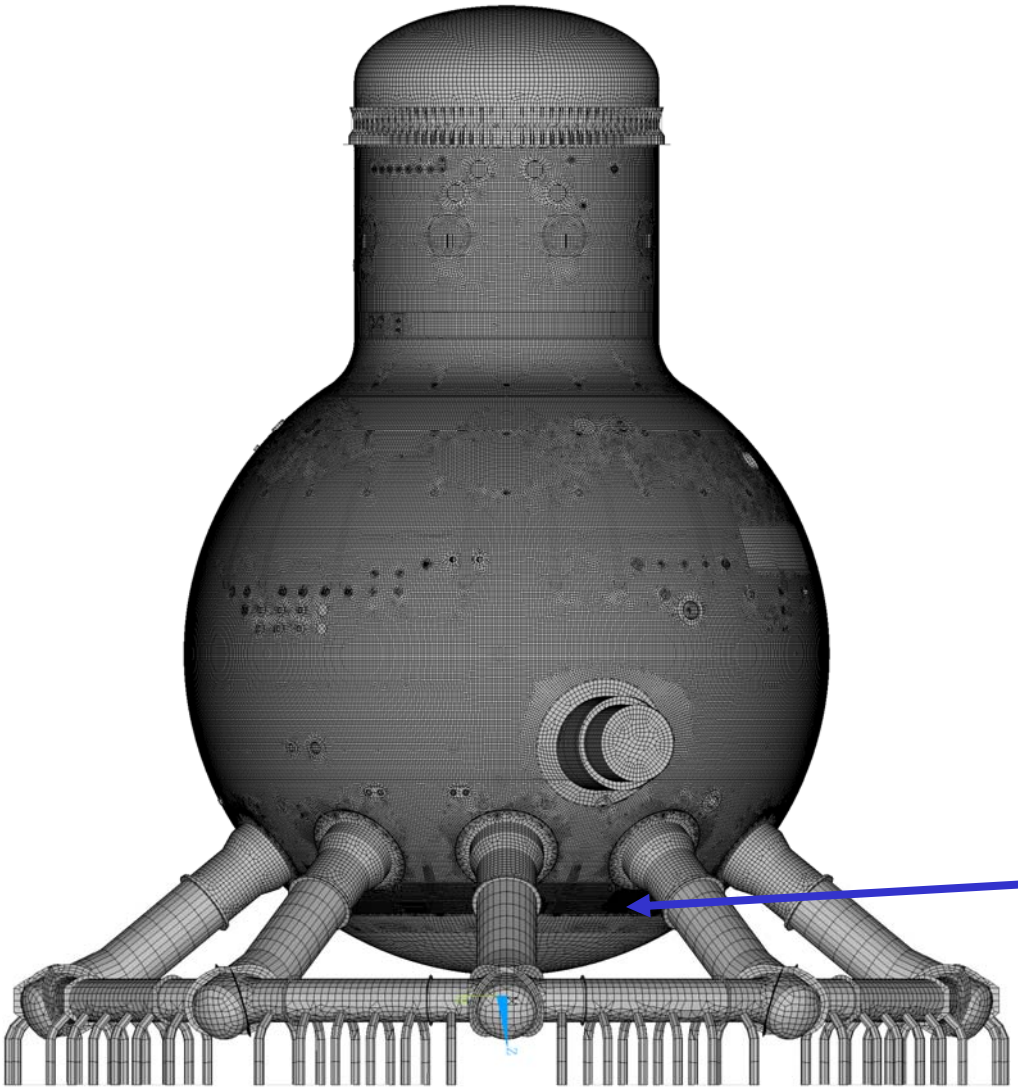
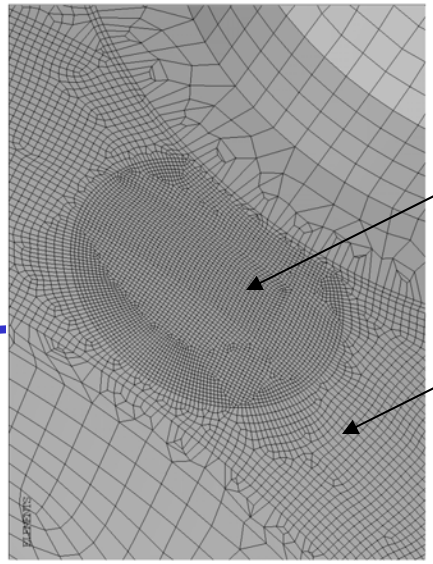
Finite Element Mesh



Elevation View of Finite Element Model

MESH SIZES	
Region	General Mesh Size (Inch)
Local Thinning Area	0.75
Sandbed Region	1.50
Mid Spherical Shell	3.00
Knuckle Region	3.00
Cylindrical Shell	3.00
Top Dome	6.00
6 inch or smaller penetrations	2.50
8 inch or larger penetrations	5.00
Vent Pipes	6.00
Vent Header	10.00
Equipment Hatch	6.00
Bottom Spherical Shell (within concrete)	12.00



Locally Thinned Area in Bay 19

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Subcommittee Question #3

- ✓ Describe specifically the modeling of the vent header boundary condition. Is this boundary condition appropriate as to the effect on the shell? Will the downcomer vents buckle?
- ✓ Response:
  - The vent header boundary condition modeling is appropriate and the boundary condition does not provided added benefit to the shell analysis
  - The vent pipes/header will not buckle before the shell

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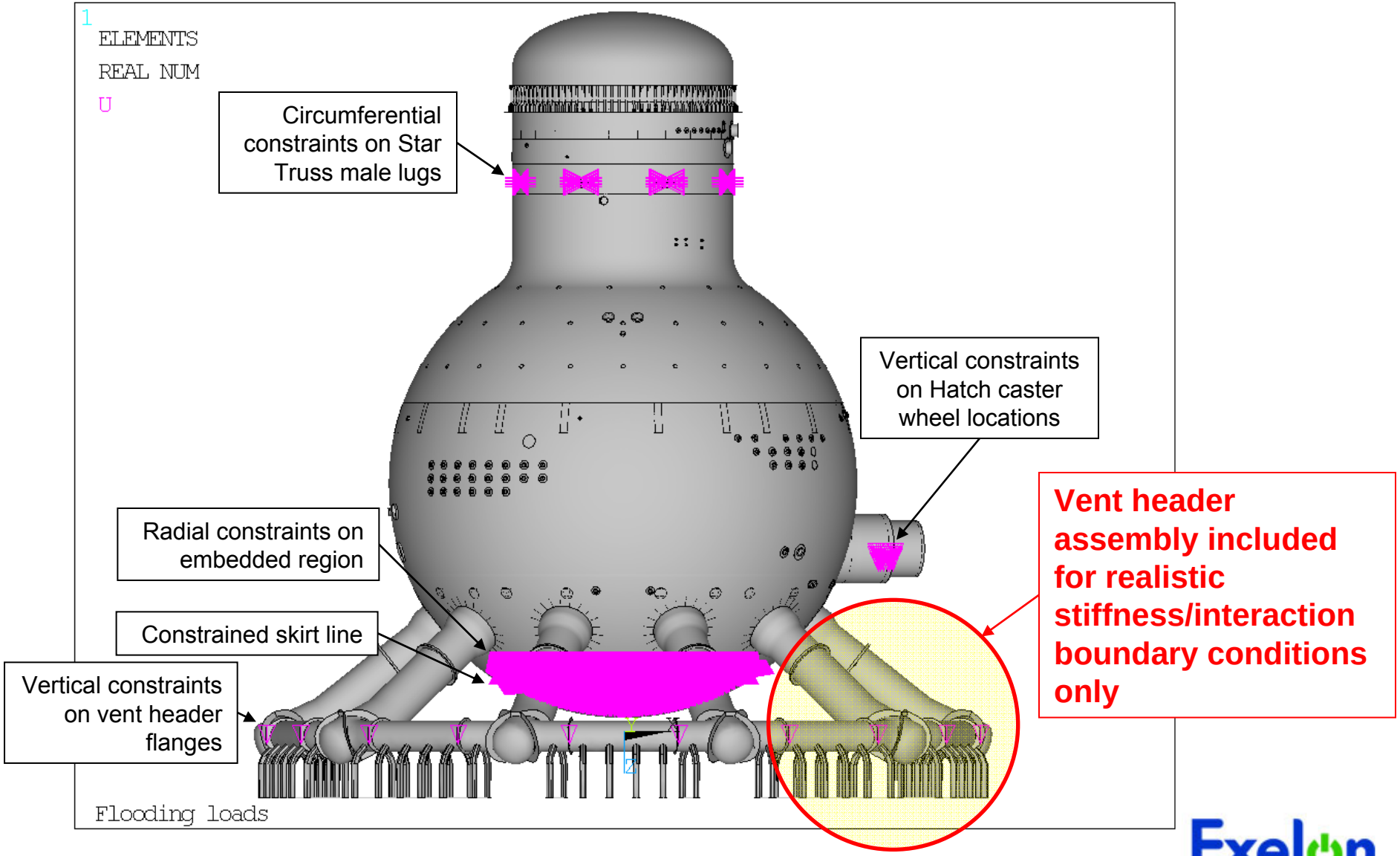
Boundary Conditions & Vent Header Interaction



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Drywell Stiffness Interface with Vent Header

- ✓ Detailed modeling of the vent headers/downcomers is included for realistic representation of stiffness/interaction of components attached to the drywell shell
  - Similar to the use of stiffness matrix as boundary conditions
- ✓ Detailed modeling reduces conservatism in stress/buckling results
  - Account for realistic load transfer
  - Alternate simplified approach is to model with estimated boundary conditions at vent pipes/vent header interface location

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Reaction Loads at Embedded Sandbed Region

- ✓ Loads going through the vent pipes are low
- ✓ Case: Flooding Loads
  - 93% of reaction force was at the embedded bottom head boundary conditions
  - 4% of reaction force was at the vent header boundary conditions
  - 3% went to other boundary conditions (equipment hatch and star truss)

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Vent Pipes will not Buckle before the Shell

- ✓ R/t: Vent pipe ≈ 150, Vent header ≈110, Downcomers ≈ 48
  - Cylindrical shell R/t ≈ 300
  - Spherical shell R/t ≈ 400 to 600
- ✓ Length between stiffeners: Vent Pipe ≈ 85”, Vent header ≈ 48”, Downcomers ≈ 75”
  - Cylindrical shell length between stiffeners = 103”
  - Spherical shell length between stiffeners = 235” to 332”
- ✓ In Code Case N-284-1, for cylinder, a smaller R/t gives a larger capacity reduction factor and higher buckling strength

Conclusion: Based on the comparison of R/t and the length between stiffeners for the vent pipes and the shell, the vent pipes would have higher buckling strength and larger safety factors than the shell.

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Sensitivity Studies

Sensitivity Studies

- ✓ Case 1: Reduce the wall thickness of the 51 inch defined locally thin area in Bay 1 by 100 mils (from 696 mils to 596 mils), keeping the general area of the thickness in Bay 1 constant (826 mils)

- ✓ Case 2: Reduce the wall thickness of the general area in Bay 19 by 50 mils (from 826 mils to 776 mils), keeping the locally thin area constant (720 mils)

Results (Sandbed Region)

Load Combination Case	Required Safety Factor	Base Case Safety Factor (Limiting)	Case 1 Safety Factor (Limiting)	Case 2 Safety Factor (Limiting)
		Based on 2006 data confirmed by 2008 data	100 mil local reduction in Bay 1	50 mil reduction of Bay 19
Refueling	2.0	3.54 Bay 3	3.21 Bay 3	3.46 Bay 3
Post Accident Flooding	1.67 (Service Level C)	2.02 Bay 19	2.01 Bay 19	1.98 Bay 19

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Subcommittee Question #4

- ✓ How is the selection of the 100 mil local reduction and 50 mil bay-wide reduction related to UT data?
- ✓ Response:
  - **Local Area Reduction**: The measurement difference between individual external UT points was -6/+1 mils. The 100 mil local reduction is over 15 times the observed external UT measurement difference and, therefore, is expected to bound data uncertainties.
  - **General Bay-Wide Reduction**: The internal UT Grid measurement standard errors range from +/- 2 to +/- 16.6 mils. The 50 mil bay-wide reduction is at least 3 times the standard error for the data used to determine the general bay-wide thickness input and, therefore, is expected to bound data uncertainties.
  - UT measurements demonstrate that the drywell shell is not experiencing corrosion; however, a hypothetical corrosion rate of 2 mils per year would yield a postulated bounding value of 8 mils over a 4 year measurement interval. The selected reductions of 100 mil local or 50 mil bay-wide are much greater.

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Overall Results/Conclusions

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Overall Results/Conclusions

- ✓ For normal operating conditions, the limiting condition is the refueling condition. For this condition, the current safety factor of the limiting sandbed bay is 3.54, which results in a safety margin greater than the ASME Code specified safety factor of 2.0.

- ✓ For emergency conditions, the limiting condition is the post-accident flooding condition. For this condition, the current safety factor of the limiting sandbed bay is 2.02, which results in a safety margin greater than the ASME Code specified safety factor for Service Level C conditions of 1.67.

- ✓ Sensitivity studies demonstrate that significant thickness changes could occur in the future, or measurement uncertainties could exist, without a significant reduction in margin to ASME Code specified safety factors.

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Questions?

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NRC Staff Review of Oyster Creek Drywell Shell 3-D Finite Element Analysis

Allen Hiser, Jr.
Hans Ashar
Division of License Renewal

Kamal Manoly
Division of Engineering

Timothy O'Hara
Region I

Advisory Committee on Reactor Safeguards
October 8, 2009

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TOPICS

- Overall Perspective
- Historical Perspective of Analyses
- NRC Inspections
- NRC Staff Review
- NRC Conclusions

OVERALL PERSPECTIVE

- Corrosion identified late 1980s
  - Narrow band of the lower spherical portion of the shell – “sandbed” region
  - Prevalent in half of bays

- Remediation
  - Removed source of corrosion (wet sand)
  - Protected with a robust coating
  - Ultrasonic test measurements indicate corrosion arrested; upper shell measurements indicate low general corrosion

OYSTER CREEK DRYWELL SHELL ANALYSES

- General Electric Analysis - 1992
  - Assumed uniform reduction in shell thickness (sandbed region) to minimum measurement (0.736 in.) and locally thinned area (0.536 in.)
  - Modified capacity reduction factor – refuel load
  - Current licensing basis analysis
- Sandia - 2007
  - Conservatively modeled degradation based on average of external UT measurements, locally thinned area set to bay lowest thickness
  - Conservative assumptions based on no access to proprietary design data
  - Unmodified capacity reduction factor – refuel load
- Structural Integrity Associates – 2009
  - Realistic analysis to compute margins in existing drywell shell above ASME Code
  - Performed base case analysis and sensitivity analyses to address measurement uncertainty
  - Access to GE proprietary design data
  - Modified capacity reduction factor – refuel load

All analyses demonstrate margins that exceed ASME Code

INDEPENDENT REVIEWS

- Brookhaven National Laboratory review of GE analysis for NRC
- ACRS review of Sandia analysis as a part of license renewal review
- Becht Nuclear Services review of SIA analysis for State of New Jersey

All three reviews point to the conclusion that the Oyster Creek drywell shell can perform its intended function without compromising the ASME Code margins

NRC STAFF REVIEWS

- January 22, 2009, submittal
  - Base case and sensitivity cases
  - Detailed rigorous review of submittal
  - Part of license renewal inspection
    - Inspection report issued (ML091380379)
  - Staff assessment issued (ML091310413)

- September 9, 2009, submittal
  - Two revised modeling approximations
  - Detailed review of submittal
  - Performed audit to supplement review
  - Audit report to be issued

NRC INSPECTIONS

ACTIVITIES

- Region I inspectors observed Exelon's drywell shell inspections in 2006 and 2008
- Noted by sample observation that external measurement locations appeared to be chosen in areas of significant relative corrosion

OBSERVATIONS

- Drywell exterior coating was, in general, in good condition
- UT inspections were conducted and reported in a competent, accurate manner
- Sandbed general conditions were good, some minor repairs to seals and coating were needed

NRC REVIEW FINDINGS

CONSERVATIVE ASSUMPTIONS

- Bounding seismic response spectra
- In post accident flooding case, all the water in included as added mass in the drywell seismic analysis
- Conservative 2% damping under OBE
- Post accident Flooding Case evaluated against ASME Service Level C limits
- Support provided at locations of Star Truss/bioshield wall conservatively neglected from the analyses, especially in buckling evaluation
- Sizes of locally thinned areas in sandbed region conservatively mapped

NRC REVIEW FINDINGS

FINITE ELEMENT MODEL

- Realistic estimate analysis of current condition of the drywell shell including realistic simulation of locally-thinned areas identified in 1992 and confirmed in 1994, 2006 & 2008
- Extensive model with general element size of 3”, refined to 0.75” in locally-thinned areas

STRESS EVALUATION

- Refueling case limiting load combination in Levels A and B service conditions, while post-accident flooding case limiting for Level C service condition

NRC REVIEW FINDINGS

BUCKLING EVALUATION

- Where applicable, CRF modified to account for benefit of hoop tensile stresses in the shell as a result of the weight of the water in the reactor cavity pool
- Refueling and post-accident flooding load cases are controlling for buckling evaluation
- In refueling load case, minimum FS 3.39 in the cylindrical region compared against a min FS of 2.0
- In flooding load case, minimum FS 2.0 in the sandbed region, compared against a min FS of 1.67

NRC REVIEW FINDINGS

SENSITIVITY ANALYSIS CASES

- Two sensitivity analysis cases considered to capture potential uncertainties in location and degradation of locally thinned areas
- Stresses in regions above the sandbed area marginally affected and remain essentially the same as in the baseline analysis … less than 8% increase in stress intensities well below allowable ASME Code limits

NRC CONCLUSIONS

- Overall, the 3-D FEA analysis performed utilizing widely accepted engineering practices consistent with ASME Code, good engineering judgment, and applied conservatively biased realistic assumptions
- All components of the drywell shell show adequate margins against instability under refueling loads and post-accident flooding loads
- In all loading cases, stresses are less than allowable ASME Code limits
- For refueling load case, margin against buckling is the lowest in the cylindrical shell region, an area that has experienced little corrosion
- Evaluations in all cases (baseline and sensitivity cases) confirm the Oyster Creek drywell shell complies with the ASME Code limits, and provide reasonable and realistic quantification of the available safety margin of the drywell shell for the postulated loading conditions

Sunil D. Weerakkody

Deputy Division Director – Fire Protection

Daniel M. Frumkin

Team Leader, Fire Protection Branch

Division of Risk Assessment

Office of Nuclear Reactor Regulation

Regulatory Guide 1.189, Fire Protection for Nuclear Power Plants

ACRS

October 9, 2009

Purpose

- Inform the Committee of Background and Regulatory Context for Regulatory Guide
- Inform the Committee that the guide is prepared and inter-office concurrences have been received
- Discuss Technical Topics
- Request Committee endorsement to issue the regulatory guide

Background and Context

- Background
  - A rule that has key words which are undefined
  - Issuance of a rule without a regulatory guide
  - Tests that showed the need for regulatory action
  - Implications of backfit
- Proposed resolution to multiple spurious actuations in SECY 06-0196, “Issuance of Generic Letter 2006-xx, “Post-Fire Safe-Shutdown Circuits Analysis Spurious Actuations”
  - The Commission disapproved proposal and issued directions to staff via SRM-06-0196
- The staff has fulfilled the Commission direction in SRM-06-0196

The Guide is Ready

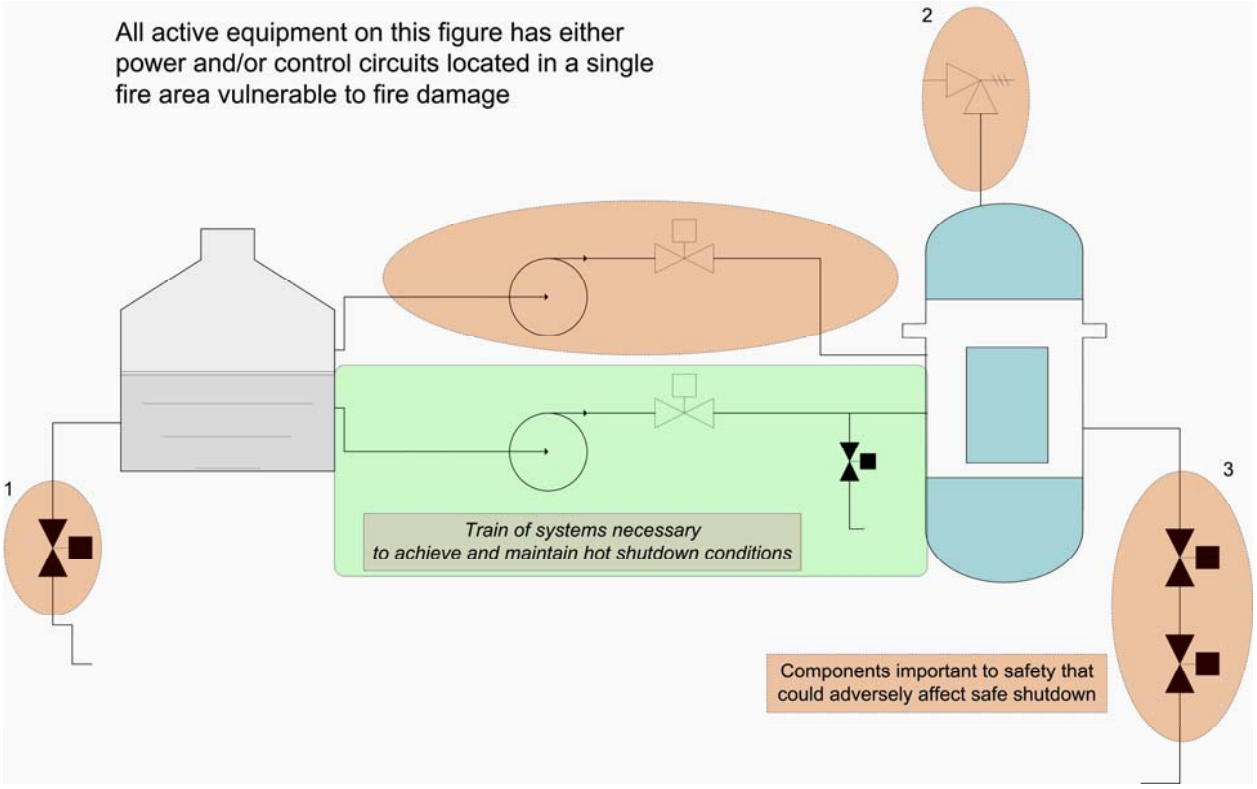
- NRC staff has prepared a Regulatory Guide that has received concurrences from all offices (NRO, OGC, RES).
  - Minor changes have been made to the regulatory guide to address comments from NRO, RES, OGC, and Technical Editor
- The staff has developed an update to the Regulatory Guide 1.189 that enables the staff to disposition circuit issues in a manner that assures plant safety and adequate protection

Safe Shutdown Success Path Components vs. Components Important to Safe Shutdown (1)

- SECY 08-0093 describes two categories of equipment:
  - Safe Shutdown Success Path
    - Also “Green Box” or “Components Required for Hot Shutdown”
  - Components Important to Safe Shutdown
    - Also “Orange Box”
- Although both require protection to assure safety – only Safe Shutdown Success Path Components require Appendix R, III.G.2 protection

Safe Shutdown Success Path Components vs. Components Important to Safe Shutdown (2)

- SECY 08-0093, “Resolution of Issues Related to Fire-Induced Circuit Failures.”



Concurrent Hot Shorts in Separate Cables for Components Important to Safe Shutdown

- Language in guide:
  - For circuits not sealed-in or latched for equipment important to safe shutdown, licensees should consider multiple fire-induced circuit failures in at least two separate cables. For circuits not sealed-in or latched for equipment important to safe shutdown that involves high-low pressure interfaces, licensees should consider circuit failures in at least three cables. This applies where defense-in-depth features, such as automatic suppression and limits on ignition sources and combustibles, are present. Where defense-in-depth features are not present, the number of cables to consider should not be limited to two or three as described above. In addition, for multiconductor cables, all circuit faults that could occur within the cable should be assumed to occur.

Request for Endorsement

- NRC staff requests the Committee to issue a letter endorsing the guide
- Commission priority requires the NRC staff to issue guidance to licensee so that they can begin dispositioning circuit issues.
- Issuance of the Final Regulatory Guide 1.189 is planned for the fourth quarter of 2009
- Issuance of R.G. 1.189, will start the “clock” on Enforcement Guidance Memorandum (EGM) 09-002:
  - Licensees will have six months to identify noncompliances
  - And an additional 30 months to resolve those noncompliances

Backup Slide

- Rule Language – 10 CFR 50, Appendix R, III.G.2
  - "where cables or equipment . . . of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area . . ., one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided:"
    - 3 hour fire barrier
    - 20' and suppression and detection
    - 1 hour barrier and suppression and detection
- To summarize – only equipment necessary to achieve and maintain hot shutdown conditions is required to have III.G.2 protection provided

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New Plant Licensing Process

An Overview

Jerry N. Wilson, PE
Office of New Reactors

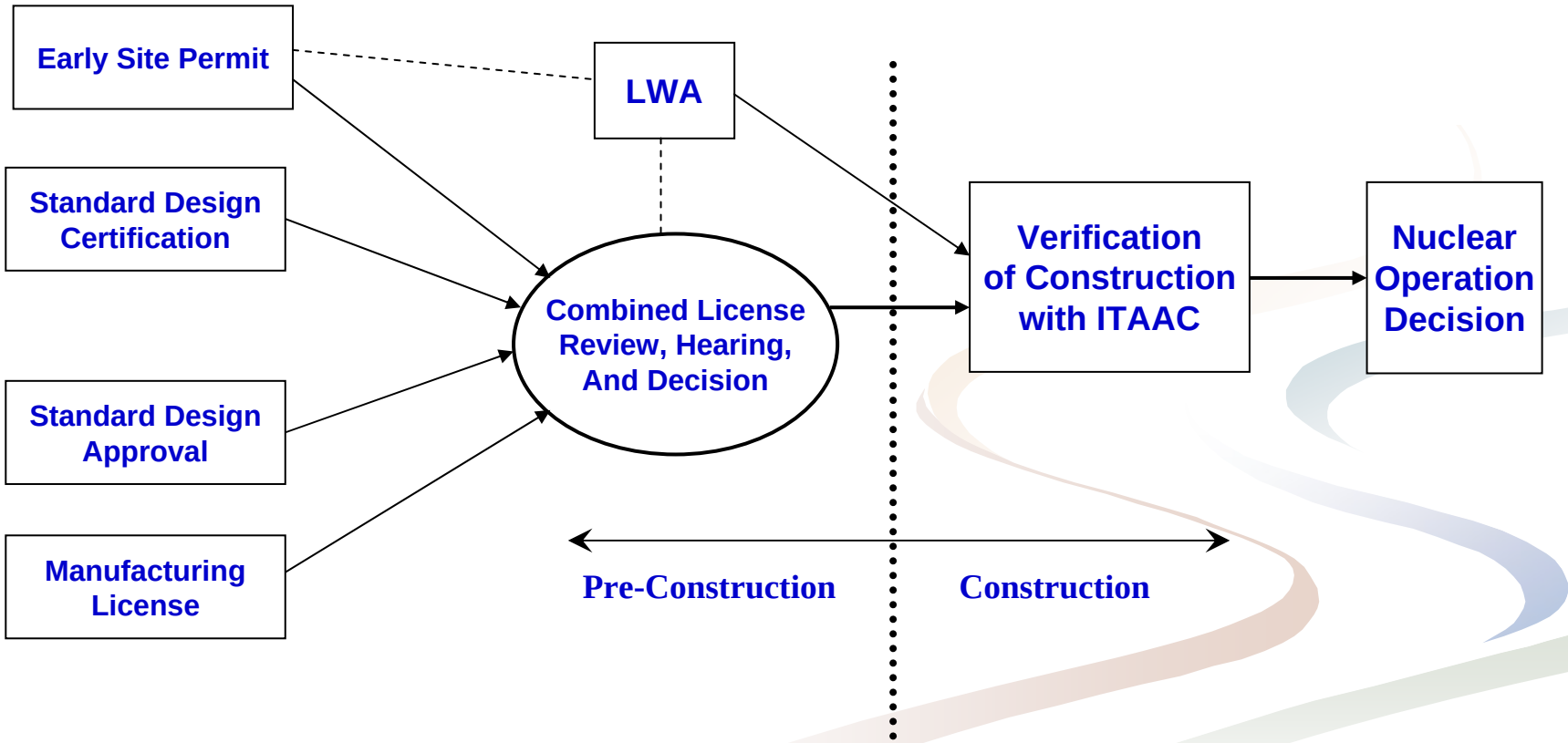
Required for Licensing

- Applicant Qualifications
- Design Acceptability
- Environmental Impacts
- Operational Programs
- Site Safety
- Verification with ITAAC

Part 52 Licensing Processes

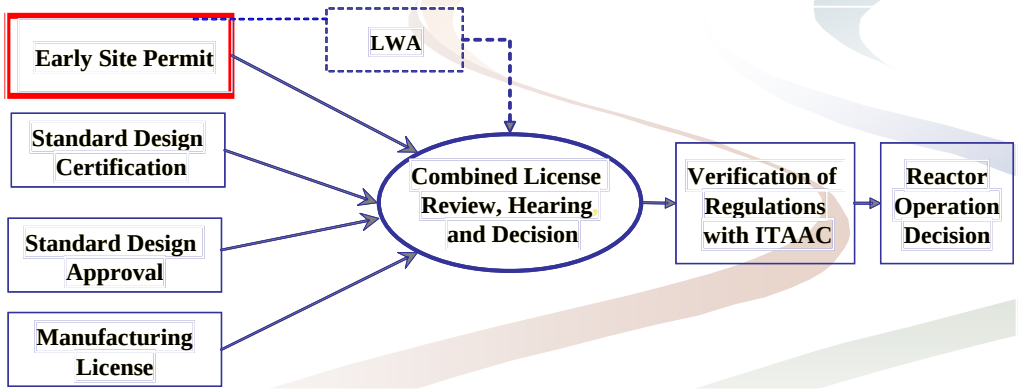
- Licensing Processes:
  - Early Site Permit (ESP)
  - Design Certification (DCR)
  - Combined License (COL)
  - Standard Design Approval
  - Manufacturing License (ML)
- Provide a more predictable licensing process
- Resolve safety and environmental issues before authorizing construction
- Provide for timely & meaningful public participation
- Encourage standardization of nuclear plant designs
- Reduce financial risk to nuclear plant licensees

Part 52 Licensing Process



Early Site Permits (Subpart A)

- Allows applicant to “bank” a site
- Are licenses (partial construction permits)
- Good for 10-20 yrs [52.27] + renewal
- Review Scope [52.18] :
  - Site Safety
    - Site characteristics that affect the design
  - Environmental Impact
    - What the design can do to the environment
  - Emergency Preparedness



Early Site Permits

- 4 early site permits (ESPs) issued
- 3 ESPs referenced in COL applications

- Challenge: applicant's use of plant parameter envelopes

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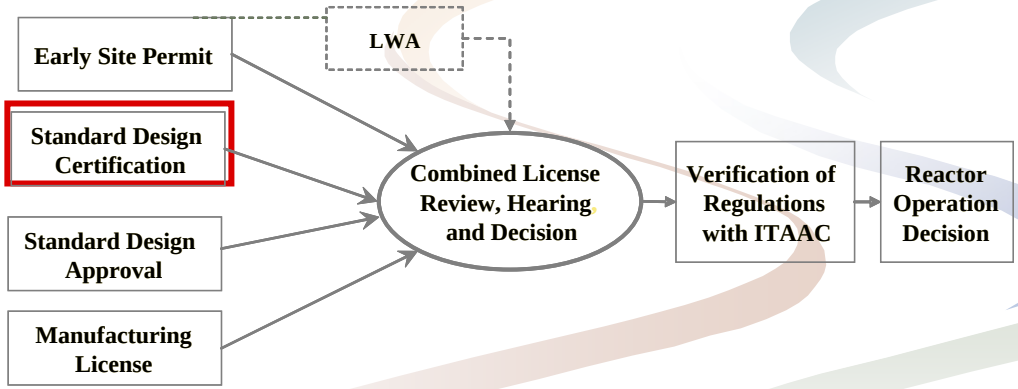
Standard Design Certifications (Subpart B)

- Allows NSSS vendor/applicant to obtain pre-approval of design (certified design becomes an Appendix to Part 52)
  - Reduces licensing uncertainty by resolving design issues early
  - Facilitates standardization
  - Facilitates regulatory finality through rulemaking
- Certification good for 15 yrs [52.55] + renewal



Standard Design Certifications

- 4 designs certified
- 4 designs under review

- <u>Challenges</u>:

  - 1 certified design referenced

  - Incomplete designs (DAC)

  - Level of design detail

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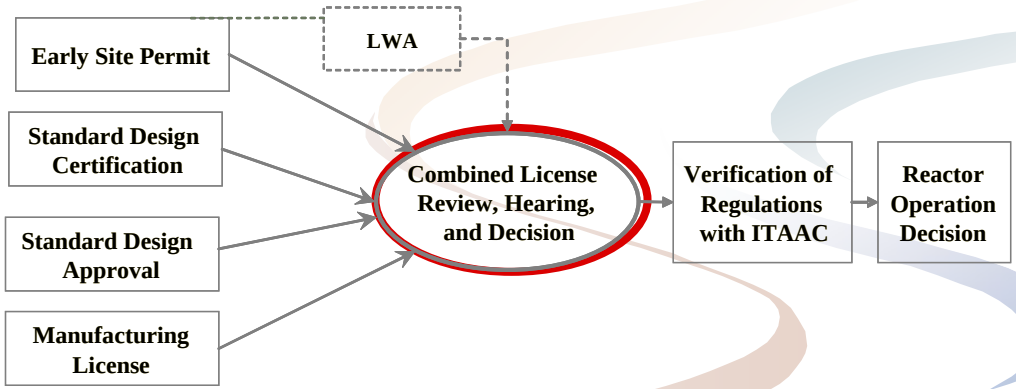
Combined License (Subpart C)

- Combined construction permit and operating license with conditions [52.1(a)]
- Fundamental licensing process in Part 52 for reducing financial risk of applicants/licensees
- Can reference ESP, Certified Design, Design Approval, Manufacturing License, or none
- Lasts 40 yrs [52.104] + renewal



Combined Licenses

- 17 applications for a combined license
- Successes:

  - Design center review approach

  - RCOL - one issue, one review

  - SCOLs - maximize standardization

<u>Challenge</u>: multiple applications for parallel COL & DCR reviews

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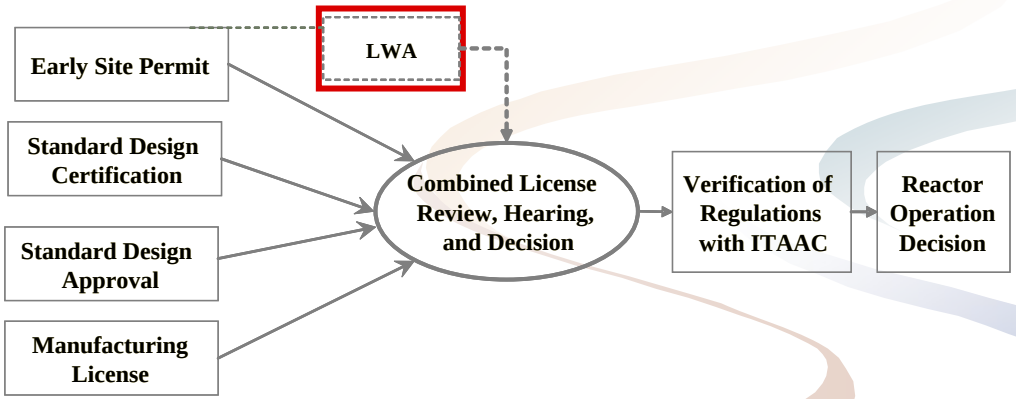
Limited Work Authorization (LWA)

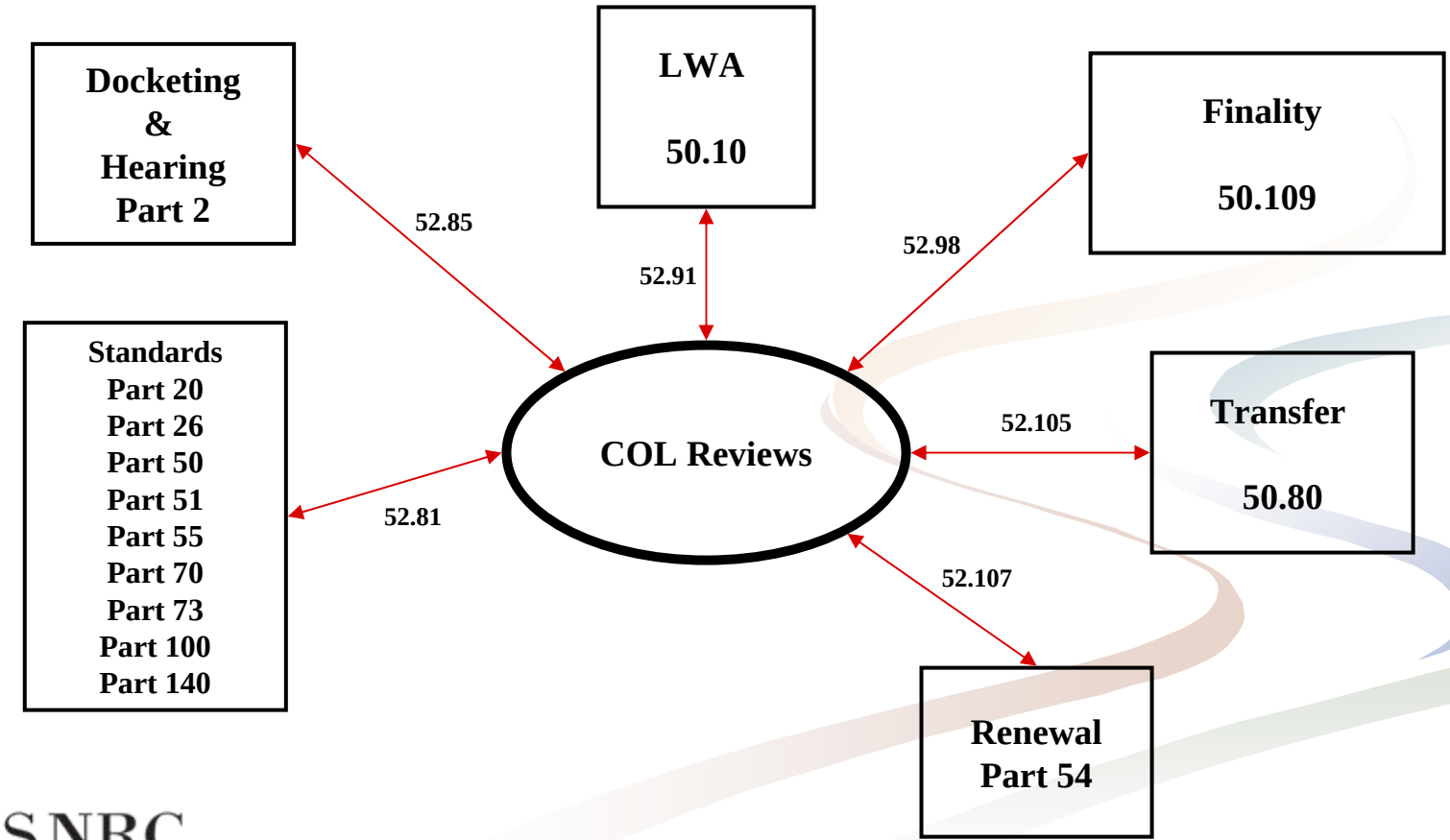
- May request LWA in advance of COL
- Safety review of requested activities
- EIS for requested activities
- Site Redress Plan
- Bifurcated hearing on LWA activities



LWA Activities

- Activities that may be authorized under an LWA:
  - Driving of piles
  - Subsurface preparation
  - Placement of backfill, concrete, or permanent retaining walls
  - Installation of foundation

Part 52 Uses other Regulations



Revisions to Part 52

- Reorganized subparts w/common format & content
- Clarified applicability of technical requirements to each licensing process, e.g. Part 21 and App. B to Part 50
- Provided a process for amending design certifications
- Required COLs to provide schedule for completing ITAACs and notify NRC of scheduled fuel load date
- Required COLs to address operational programs
- Required COLs to maintain & upgrade PRAs

Key Terms in Part 52

(a) *Early site permit* means a Commission approval, issued under subpart A of this part, for a site or sites for one or more nuclear power facilities.

(b) *Standard design certification or design certification* means a Commission approval, issued under subpart B of this part, of a final standard design for a nuclear power facility. A design so approved may be referred to as a certified standard design.

(c) *Combined license* means a combined construction permit and operating license with conditions for a nuclear power facility issued under subpart C of this part.

Key Terms in Part 52 (cont.)

(e) *Standard design approval or design approval* means an NRC staff approval, issued under subpart E of this part, of a final standard design for an entire nuclear power facility or a major portion thereof.

(f) *Manufacturing license* means a license, issued under subpart F of this part, authorizing the manufacture of nuclear power reactors but not their construction, installation, or operation at the sites on which the reactors are to be operated.

ITAAC – the inspections, tests, and analyses that the licensee shall perform, and the acceptance criteria that, if met, are necessary and sufficient to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license, the provisions of the Act, and the Commission's regulations.