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

August 16, 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 572nd FULL COMMITTEE MEETING HELD ON MAY 6-8 2010 IN ROCKVILLE, MARYLAND

The minutes of the subject meeting were certified on August 6, 2010 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

August 6, 2010

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 572nd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), MAY 6-8, 2010

I certify that based on my review of the minutes from the 572nd 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/Sunsi
NAME	SMeador	CSantos/sam
DATE	08/ 06 /10	08/ 06 /10

OFFICIAL RECORD COPY

CERTIFIED Date Certified: 08/06/2010

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During its 572nd meeting, May 6-8, 2010, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following letters and memoranda:

LETTERS

Letters to R. W. Borchardt, Executive Director for Operations, NRC, from Said Abdel-Khalik, Chairman, ACRS:

- Revised Standard Review Plan for Spent Fuel Dry Storage Systems (NUREG-1536), dated May 17, 2010

- Draft Guidance on Crediting Containment Accident Pressure in Meeting the Net Positive Suction Head Required to Demonstrate that Safety Systems Can Mitigate Accidents as Designed, dated May 19, 2010

MEMORANDA

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

- Comments Regarding Licensed Operator Training Standards, dated May 12, 2010

- Proposed Standard Review Plan, Branch Technical Position 7-19, dated May 12, 2010

- Final Interim Staff Guidance ISG-25, dated May 12, 2010

- Proposed Revisions to Regulatory Guides 3.67 and 7.3, dated May 12, 2010

MINUTES OF THE 572nd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MAY 6-8, 2010
ROCKVILLE, MARYLAND

The 572nd meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B1, Two White Flint North Building, Rockville, Maryland, on May 6-8, 2010. Notice of this meeting was published in the *Federal Register* on April 22, 2010 (72 FR 21046-21047) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting schedule and outline (Appendix II). The meeting was open to public attendance.

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

ATTENDEES

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

I. Chairman's Report (Open)

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

Dr. Said Abdel-Khalik, 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. AbdelKhalik 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.

In addition, it was announced that after 38 and a half years of working at the NRC, Ms. Michele Kelton will be retiring in early June. Ms. Kelton started with the Atomic Energy Commission in August of 1971 as a secretary in the Division of Reactor Licensing, now the Office of Nuclear Reactor Regulation. In May of 1975, Ms. Kelton joined the ACRS as secretary to a senior staff engineer. Ms. Kelton has remained in ACRS since 1975, holding positions of increasing responsibility, including secretary to the Deputy Director of ACRS, technical secretary to the ACRS and ACNW, and at the current position as management analyst. Her dedication, hard work, professionalism, attention to details, and exceptional work ethics are very much appreciated. We thank her so much and wish her the best in her future endeavors.

II. Revision 1C to NUREG-1536, "Standard Review Plan for Spent Fuel Storage Systems at a General License Facility"

[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 Revision 1C to NUREG-1536, "Standard Review Plan (SRP) for Spent Fuel Dry Storage Systems at a General License Facility." The SRP provides guidance to the NRC staff for reviewing applications for a dry storage system Certificate of Compliance. The document promotes a consistent regulatory review of a vendor's application. It also provides a framework that identifies acceptable approaches to meeting the regulatory requirements for a spent fuel dry storage system in accordance with 10 CFR Part 72.

In the SRP the staff describes a new prioritization methodology that ranks review areas as high, medium, and low. The staff discussed the basis for performing burnup credit measurements in the SRP. In order to prevent misloading of assemblies that do not meet the criteria for burnup credit, the SRP states that the applicant should perform supplementary pool-side measurements to confirm assembly burnup values. Acceptable measurement methods are not described in the SRP. ACRS Members noted that it was unclear why plant burnup records alone would not be acceptable to confirm assembly burnup once the assemblies are explicitly verified.

The Committee issued a letter to the Executive Director for Operations on this matter dated May 17, 2010, recommending that NUREG-1536 be issued after the discussion of burnup credit is removed and reference is made to Interim Staff Guidance (ISG)-8, Revision 2, "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks." The Committee also recommended that the staff reconsider the guidance in ISG-8, Revision 2, regarding the need for supplementary measurements to verify and adjust plant burnup records prior to granting burnup credit.

III. BWR Owners Group (BWROG) Topical Report NEDC-33347P, "Containment Overpressure (COP) Credit for Net Positive Suction Head (NPSH)" and the Staff's Proposed Guidance for the Use of COP (Open/Closed)

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

The Committee met with representatives of the NRC staff to discuss the draft staff guidance regarding the acceptability of crediting containment accident pressure (CAP) in extended power uprates (EPUs) and other applications. In a January 8, 2009, Staff Requirements Memorandum, the Commission directed the staff to continue working to resolve the differences of opinion between the Committee and the staff concerning CAP. In 2009, the Committee reviewed the staff's White Paper on crediting CAP. As a result of this review, the Committee issued a March 18, 2009, letter, which delineated the Committee's concerns and proposed some alternate approaches for the staff's consideration. The current draft staff guidance incorporates some of the recommendations in the March 18, 2009, ACRS letter.
The guidance provides criteria that applications crediting CAP should meet for different

scenarios such as, design-basis loss-of-coolant accidents (LOCAs) and special events (e.g., Appendix R fires, Anticipated Transients Without Scram, and Station Blackout). The guidance also describes the types of analyses and pump tests that should be performed.
The staff re-examined the behavior of pumps operating with available NPSH near the required NPSH. They consulted with pump experts to help estimate the uncertainties associated with the required NPSH. The staff concluded in their draft guidance that: (1) uncertainties associated with the determination of the required NPSH during a LOCA should be accounted for; (2) appropriate modifications to the pump mechanical seal designs should be considered; and (3) the duration of pump operation within the maximum wear rate region should be limited to 100 hours.

Although the draft guidance relies on deterministic licensing-basis analyses that assume containment integrity, the staff described a generic assessment of the change in risk due to containment leakage large enough to reduce containment pressure below that needed for operation of the pumps. For containment leakage test intervals believed to be representative of BWRs with Mark I containments (approximately once per week), the predicted change in core damage frequency is less than $1x10^{-7}$/yr. The staff's risk model assumed that the failure rate of containment isolation is the same before and after the accident. The model did not include the possibility that head loss due to sump strainer blockage is greater than the predicted value. The generic risk study also included some initial seismic risk estimates, but the risk associated with other significant contributors such as fires and operator actions were not included.
The draft staff guidance includes an explicit expectation that licensees demonstrate the impracticality of plant modifications that would eliminate the need for CAP credit. The Committee strongly supports that position.

The Committee issued a letter to the Executive Director for Operations on this matter dated May 19, 2010, concluding that the draft guidance provides an improved framework for a more comprehensive assessment of the acceptability of crediting CAP, but that these analyses should be complemented by plant-specific PRA analyses. The Committee agrees with the staff that before considering analyses to justify credit for CAP, licensees must first demonstrate that it is impractical to make plant modifications that eliminate this need. However, the Committee disagrees with the staff that a generic waiver of this requirement is appropriate for BWRs with Mark I containments. The Committee concluded that if no CAP credit is needed for special events licensing-basis analyses, and the 95/95 lower tolerance bound for LOCAs calculated using an acceptable methodology shows that no CAP credit is needed, then CAP credit can be deemed to be small enough that it is acceptable without the need for hardware modifications or additional risk studies. This conclusion is consistent with the intent of the previous ACRS position that if the CAP associated with the licensing-basis analysis is sufficiently "short" and "small," then it can be assumed to be largely due to conservatism in the calculation and does not represent a significant challenge to the independence of barriers and the associated risk is small.

Although the Committee's position is documented in the May 19, 2010, letter, four Members disagreed with the specific conclusion that supplemental risk assessments or plant modifications are not needed for those cases in which special events licensing-basis analyses and the 95/95 lower tolerance bound for LOCAs show that the CAP credit is not needed. These Members provided additional comments that are documented in the May 19, 2010, letter and

recommend that granting of CAP credit should include a thorough evaluation of potential safety system modifications that eliminate the need for CAP credit and, in the event that these modifications are impractical, a plant-specific full-scope PRA be performed to demonstrate that the increase in risk is small.

VI. MEETING WITH THE NRC CHAIRMAN

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

Chairman Jaczko addressed the Committee and provided his perspectives on nuclear safety and technical matters of significant interest to the NRC. These included the proposed revisions to 10 CFR 50.46a, implementation of risk-informed performance-based fire protection (NFPA 805), safety culture, the reactor oversight process, and plant license extensions beyond 60 years. The Chairman also entertained ACRS Member questions on these and other matters of interest to the Committee. This was an information exchange. No Committee action was necessary.

V. 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 Commitments

- The Committee considered the EDO's response of April 20, 2010, to comments and recommendations included in the March 18, 2010, ACRS report on the status of staff rulemaking efforts for depleted uranium and other unique waste streams. The Committee decided that it was satisfied with the EDO response. In the EDO response, the staff committed to include a discussion of how early comments, including those of the ACRS, were addressed in the Statements of Consideration.

- The Committee considered the EDO's response of April 16, 2010, to comments and recommendations included in the February 22, 2010, ACRS report on draft final NUREG-1520 Revision 1, "Standard Review Plan for Review of a License Application for a Fuel Cycle Facility." In its report, the ACRS recommended that when developing and reporting the results of Integrated Safety Analyses (ISAs), the staff should consider fire-induced "hot shorts" and their potential to place systems in conditions other than a fail-safe condition. The EDO response stated that the staff will evaluate the potential for fuel cycle events related to hot shorts; and if the staff determines that these are common events or have significant risk for fuel cycle facilities, the staff will develop guidance utilizing the results of ongoing research on this topic. The Committee will evaluate the adequacy of the EDO's response to this issue and will consider preparing a response during its June 9-11, 2010, meeting.

- The Committee considered the EDO's response of March 25, 2010, to comments and recommendations included in the November 12, 2009, ACRS report on draft final Regulatory Guide 5.71, "Cyber Security Programs for Nuclear Facilities." The Committee decided that it was satisfied with the EDO's response.

B. Report of the Planning and Procedures Subcommittee Meeting

Anticipated Workload for ACRS Members

The anticipated workload for the ACRS members through July 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

Regulatory Guides, Interim Staff Guidances, and Branch Technical Positions

a) Draft Final Regulatory Guides

The staff plans to issue the following Draft Final Regulatory Guides (RGs) and would like to know whether the Committee wants to review these Guides prior to being issued as final.

- Draft Final Revision 2 to Regulatory Guide 1.174 (DG-1226), "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"

- Draft Final Revision 1 to Regulatory Guide 1.177 (DG-1227), "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications"

  Regulatory Guide 1.174, Revision 2, was issued as DG-1226 and Regulatory Guide 1.177, Revision 1, was issued as DG-1227. The changes include editing the terminology to be consistent with Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities;" the American Nuclear Society/American Society of Mechanical Engineers' Probabilistic Risk Assessment (PRA) Standard RA-Sa-2009, "Standard for Level 1 Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications;" and Standard Technical Specifications (RG 1.177 only). Other changes include deletion of outdated information (e.g., steam generator tube rupture activities), providing guidance for temporary technical specification changes, including new quantitative criteria that align with NUMARC-93-01, Revision 2 (RG 1.177 only), and updating the discussion of uncertainty to incorporate NUREG-1855, "Guidance on Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making."

  Based on his review of these Regulatory Guides, Dr. Bley recommends that the Committee review the draft final revisions to these Guides.

b) Proposed Regulatory Guides

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

- Proposed Revision 1 to Regulatory Guide 3.67 (DG-3039), "Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities"

  DG-3039 is the proposed Revision 1 to Regulatory Guide 3.67. The regulatory guide is being revised in order to reflect the changes in the regulations that have been promulgated since 1992 and the experience gained with the implementation of material and fuel cycle emergency plans.

  Based on his review of this Proposed Regulatory Guide, Mr. Sieber recommends that the Committee reconsider review of the draft final Guide after reconciliation of public comments.

- Proposed Revision 1 to Regulatory Guide 7.3 (DG-7006), "Procedures for Picking up and Receiving Packages of Radioactive Material"

  DG-7006 is the proposed Revision 1 to Regulatory Guide 7.3. The regulatory guide is being revised because it is outdated and no longer accurately reflects the regulations in 10 CFR 20.1906, "Procedures for Receiving and Opening Packages," or 10 CFR 71.89, "Opening Instructions." The draft guide directs the reader to the methods and procedures in NUREG/CR-4775, "Guide for Preparing Operating Procedures for Shipping Packages."

  Based on his review of this Proposed Regulatory Guide, Dr. Ryan recommends that the Committee reconsider review of the draft final Guide after reconciliation of public comments.

ACRS Meeting With the Commission

The ACRS is scheduled to meet with the Commission between 1:30 and 3:30 p.m., on Wednesday, June 9, 2010, to discuss items of mutual interest. The following list of proposed topics has been approved by SECY.

1. Overview (Abdel-Khalik)
   - Major Accomplishments
   - Update on Solicitation for New ACRS Members
   - New Reactors Review Activities
   - License Renewal/Power Uprates
   - Major Areas of Ongoing/Future Activities

2. Risk-Informed Performance-Based Fire Protection (RG 1.205) (Stetkar)

3. NRC Safety Research Program (Powers)

4. BWROG Containment Overpressure Methodology (Shack)

5. Rulemaking for Disposal of Depleted Uranium (Ryan)

Item 4, BWROG COP Methodology, is being considered by the Full Committee during the May meeting. A letter report will be discussed and approved at that time and so will be the set of slides supporting this presentation.

4) Annual Visit to a Nuclear Plant and Meeting with the Regional Administrator

A site visit at the Columbia Generating Station has been scheduled for Tuesday, July 27. If the members choose to also visit PNNL, this activity would have to be scheduled for Wednesday, July 28. This would mean traveling to Texas on Thursday, July 29 and meeting with Region IV on Friday, July 30. If the members decide not to visit PNNL, the meeting with Region IV could be held on Thursday, July 29.

Future Presentation by NEI on Reactor Issues

NEI has suggested a presentation to the ACRS on reactors issues and any other initiatives the ACRS may suggest for discussion. The following topics are being proposed:

- Security
- Fire Protection
- Safety Culture
- PWR Sumps
- Groundwater Contamination (Mike Ryan's suggestion)

Post-Accident AP1000 Containment Leakage

In a letter dated April 21, 2010, addressed to Dr. Abdel-Khalik, Mr. John Runkle (Counsel for the AP1000 Oversight Group) requested that the ACRS initiate a special investigation of a potentially unreviewed safety issue regarding the AP1000 design (Copies were distributed to all the ACRS members during the week of April 21.) A response was issued to Mr. Runkle on April 23, 2010 [p. 12]. We have invited the AP1000 Oversight Group to make a presentation regarding these concerns during an upcoming AP1000 Subcommittee meeting. NRC staff has also been tasked to respond to similar correspondence. We are coordinating with NRC staff as to the proposed timing of their response.

Concern with Industry Standards in Licensed Operator Training

In an April 8, 2010, letter addressed to Dr. Abdel-Khalik, Mr. Kenneth W. Norris (Licensed Operator Instructor at Braidwood Station) raised concerns regarding licensed operator training standards. Mr. Norris raises the following specific issues:

- "Industry focus on rule-based operation has resulted in a cultural shift that no longer values understanding of the fundamental scientific principles underlying plant operation. Combined with the mass exodus of experienced nuclear industry workers to retirement, this will set the stage for eventual repeat events (some resulting from procedure revisions based on inadequate knowledge level of the writer, and an operator who implements it without the knowledge base to challenge the procedure, and in fact, is discouraged from such a challenge)."

- "The nuclear industry has defaulted to the NRC GFE (Generic Fundamentals Examination) as the sole basis for a standard of understanding the scientific principles related to nuclear power plant operation. This is a result of the open market place for electricity."

- "The NRC GFE bank of questions has become obsolete and is gradually losing relevance to modern nuclear plant operation."

To address these concerns Mr. Norris recommends specific actions for both INPO and the NRC.

Status of Selection of New Members

The solicitation for new members closed on April 13, 2010. The review panel met on April 27, 2010. Some of the prospective members will be meeting with the ACRS members on Friday, May 7.

Status of Resolution of Generic Safety Issue 191 (Assessment of Debris Accumulation on PWR Sump Performance)

On April 15, 2010, the staff briefed the Commission on the status of the resolution of GSI-191. One of the options that was discussed for resolving GSI-191 was the revision to 10 CFR 50.46a. The SRM from this meeting has not yet been finalized. An information briefing on the status of the resolution of GSI-191 is scheduled for the June 2010 meeting.

10) Digital I&C DAC Pilot Inspection Procedure

On May 6, 2010, the ACRS P&P Subcommittee will meet with NRO management and staff to informally discuss NRO's DAC Working Group efforts and their upcoming delivery (tentatively for May 15) to the ACRS of a pilot Digital I&C Inspection Procedure. The staff plans to apply this procedure in a pilot ("test case") inspection of the South Texas Digital I&C DAC, which is currently scheduled for June 21.

Length and Scheduling of Subcommittee Weeks

Recent interactions among members have indicated serious concerns with the way future Subcommittee activities are being scheduled. At issue is the availability of key members to attend concurrent briefings, especially when it comes to support new reactor activities, which often involve several and diverse chapters being brought for ACRS review.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 573rd ACRS Meeting, June 9-11, 2010.

A list of documents that were provided to the Committee during the 572nd ACRS Meeting is listed in Appendix V.

The meeting was adjourned at 7:00 pm on May 7, 2010.

Biology, University of North Carolina, Wilmington, NC 28403.

Activity for Which Permit is Requested: Take and Import into the U.S.A. The applicant plans to salvage sediments from abandoned and active penguin colonies by excavation of small pits, no larger than 1x1 meter, in each area. In addition, the applicant will collect 10 each organic remains (bones, tissue, feathers, eggshell fragments, otoliths, squid beaks, and other prey remains) from sediments in abandoned colonies of Adelie, Chinstrap, Gentoo, Emperor, and Macaroni penguins, Southern Giant Petrel, Antarctic Petrel, Cape Petrel, Snow Petrel, Blue Petrel, Antarctic Fulmar, White-chinned petrel, Sooty shearwater, Wilson's Storm-petrel, Black-bellied storm-petrel, Blue-eyed shag, Greater sheathbill, South Polar Skua, Brown Skua, Kelp gull, and Antarctic Tern.

The applicant also plans to capture 100 each of adult or juvenile Adelie, Chinstrap and Gentoo penguins to collect some breast feathers and blood samples for analysis of carbon and nitrogen isotope values to examine diets, and for mercury (Hg). All capture bird will be released.

Location:

ASPA 102–Rookery Islands, Holme Bay, ASPA 103–Ardery and Odbert Islands, ASPA 104–Sabrina Island, Balleny Island, ASPA 105–Beaufort Island, ASPA 106–Cape Hallett, Victoria Land, ASPA 107–Dion Islands, ASPA 108–Green Island, Berthelot Islands, ASPA 109–Moa Island, South Orkneys, ASPA 110–Lynch Island, South Orkneys, ASPA 111–Southern Powell Island and adjacent islands, South Orkneys, ASPA 112–Coppermine Peninsula, Robert Island, ASPA 113–Litchfield Island, Arthur Harbor, Palmer Archipelago, ASPA 114–North Coronation Island, ASPA 115-Lagotellerie Island, Marguerite Bay, ASPA 116–New College Valley, Caughley Beach, Cape Bird ASPA 117–Avian Island, northwest Marguerite Bay, ASPA 121–Cape Royds, Ross Island, ASPA 124–Cape Crozier, Ross Island, ASPA 125–Fildes Peninsula, King George Island, South Shetland Islands, ASPA 126–Byers Peninsula, Livingston Island, ASPA 127–Haswell Island, ASPA 128-Western shore of Admiralty Bay, King George Island, ASPA 129–Rothera Point, Adelaide Island, ASPA 132–Potter Peninsula, King George Island, ASPA 133–Harmony Point, Nelson Island, ASPA 134–Cierva Point, Danco Coast, ASPA 135–Bailey Peninsula, Budd Coast, ASPA 136–Clark Peninsula, Budd Coast, ASPA 139–Biscoe Point, Anvers Island, Palmer Archipelago, ASPA 143–Marine Plain, Mule Peninsula, Vestfold Hills, ASPA 149–Cape Shirreff, Livingston Island and ASPA 150–Ardley Island, King George Island.

Dates: October 1, 2010 to September 30, 2012.

Nadene G. Kennedy,

Permit Officer, Office of Polar Programs.

[FR Doc. 2010–9319 Filed 4–21–10; 8:45 am]

BILLING CODE 7555–01–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 May 6–8, 2010, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Monday, October 14, 2009, (74 FR 52829–52830).

Thursday, May 6, 2010, Conference Room T2–B1, Two White Flint North, Rockville, Maryland

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

8:35 a.m.–9:30 a.m.: Revision 1C to NUREG–1536, "Standard Review Plan for Spent Fuel Storage Systems at a General License Facility" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding proposed Revision 1C to NUREG–1536, "Standard Review Plan for Spent Fuel Storage Systems at a General License Facility," and the NRC staff's resolution of public comments.

9:45 a.m.–11:15 a.m.: Preparation for Meeting with the Commission on June 9, 2010 (Open)—The Committee will discuss the topics for meeting with the Commission on June 9, 2010.

11:30 a.m.–12 p.m.: Meeting with the NRC Chairman (Open)—The Committee will hold discussions with the NRC Chairman to discuss topics of mutual interest.

1 p.m.–4 p.m.: Boiling Water Reactor (BWR) Owners Group (BWROG) Topical Report NEDC–33347P, "Containment Overpressure (COP) Credit for Net Positive Suction Head (NPSH)" and the Staff's Proposed Guidance for the Use of COP (Open/Closed)—The Committee will hold discussions with representatives of the NRC staff and the BWROG regarding the staff's review of Revision 0 to Topical Report NEDC–33347P, "COP Credit for Net Positive Suction Head," and the NRC staff's proposed guidance for the use of COP.

Note: A portion of this session may be closed to protect information that is proprietary to General Electric-Hitachi pursuant to 5 U.S.C. 552b(c)(4).

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

Friday, May 7, 2010, Conference Room T2–B1, Two White Flint North, Rockville, Maryland

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

8:35 a.m.–10 a.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding items proposed for consideration by the Full Committee during future ACRS meetings, and Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS Business, including anticipated workload and member assignments.

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

10 a.m.–10:15 a.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

10:30 a.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

Saturday, May 8, 2010, Conference Room T2–B1, Two White Flint North, Rockville, Maryland

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

12:30 p.m.–1 p.m.: Miscellaneous (Open)—The Committee will 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 14, 2009, (74 FR 52829–52830).

In accordance with those procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Persons desiring to make oral statements should notify Mr. Derek Widmayer, Cognizant ACRS Staff (Telephone: 301–415–7366, e-mail: *Derek.Widmayer@nrc.gov*), five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. 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.

Thirty-five hard copies of each presentation or handout should be provided 30 minutes before the meeting. In addition, one electronic copy of each presentation should be emailed to the Cognizant ACRS Staff one day before meeting. If an electronic copy cannot be provided within this timeframe, presenters should provide the Cognizant ACRS Staff with a CD containing each presentation at least 30 minutes before the meeting.

In accordance with Subsection 10(d) Public Law 92–463, and 5 U.S.C. 552b(c), certain portions of this meeting may be closed, as specifically noted above. 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. Electronic recordings will be permitted only during the open portions of the meeting.

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: April 16, 2010.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 2010–9299 Filed 4–21–10; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[NRC–2010–0002]

Sunshine Act; Meeting Notice

AGENCY HOLDING THE MEETINGS: Nuclear Regulatory Commission [NRC–2010–0002].

DATES: Weeks of April 19, 26, May 3, 10, 17, 24, 2010.

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

STATUS: Public and Closed.

Week of April 19, 2010

There are no meetings scheduled for the week of April 19, 2010.

Week of April 26, 2010—Tentative

Thursday, April 29, 2010

9:30 a.m. Briefing on the Fuel Cycle Oversight Process Revisions, (Public Meeting), (Contact: Michael Raddatz, 301–492–3108).

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

Week of May 3, 2010—Tentative

Tuesday, May 4, 2010

9:30 a.m. Briefing on Human Capital and Equal Employment Opportunity, (Public Meeting), (Contact: Kristin Davis, 301–415–2673).

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

10:30 a.m. Discussion of Management Issues (Closed—Ex. 2).

Week of May 10, 2010—Tentative

Tuesday, May 11, 2010

9:30 a.m. Briefing on Federal and State Materials and Environmental, Management Programs (FSME) Programs, Performance, & Future Plans (Public Meeting), (Contact: George Deegan, 301–415–7834).

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

Week of May 17, 2010—Tentative

There are no meetings scheduled for the week of May 17, 2010.

Week of May 24, 2010—Tentative

Thursday, May 27, 2010

9:30 a.m. Briefing on the Results of the Agency Action Review Meeting, (AARM) (Public Meeting), (Contact: Nathan Sanfilippo, 301–415–3951).

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

* * * * *

* The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings, call (recording)—(301) 415–1292. Contact person for more information: Rochelle Bavol, (301) 415–1651.

* * * * *

The NRC Commission Meeting Schedule can be found on the Internet at: *http://www.nrc.gov/about-nrc/policy-making/schedule.html.*

* * * * *

The NRC provides reasonable accommodation to individuals with disabilities where appropriate. If you need a reasonable accommodation to participate in these public meetings, or need this meeting notice or the transcript or other information from the public meetings in another format (*e.g.,* braille, large print), please notify Angela Bolduc, Chief, Employee/Labor Relations and Work Life Branch, at 301–492–2230, TDD: 301–415–2100, or by e-mail at *angela.bolduc@nrc.gov.* Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

* * * * *

This notice is distributed electronically to subscribers. If you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 20555 (301–415–1969), or send an e-mail to *darlene.wright@nrc.gov.*

Dated: April 14, 2010.

Rochelle C. Bavol,

Office of the Secretary.

[FR Doc. 2010–9436 Filed 4–20–10; 11:15 am]

BILLING CODE 7590–01–P

RAILROAD RETIREMENT BOARD

Sunshine Act; Notice of Public Meeting

Notice is hereby given that the Railroad Retirement Board will hold a meeting on April 28, 2010, 10 a.m. at the Board's meeting room on the 8th floor of its headquarters building, 844 North Rush Street, Chicago, Illinois

Appendix II

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

Revised AGENDA
572nd ACRS MEETING
MAY 6-8, 2010

THURSDAY, MAY 6, 2010, CONFERENCE ROOM T-2B1, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1)	8:30 – 8:35 A.M.	Opening Remarks by the ACRS Chairman (Open) (SAK/EMH) 1.1) Opening statement 1.2) Items of current interest
2)	8:35 – ~~9:30~~ 9:50 A.M.	Revision 1C to NUREG-1536, "Standard Review Plan for Spent Fuel Storage Systems at a General License Facility" (Open) (MTR/CLB) 3.1) Remarks by the Subcommittee Chairman 3.2) Briefing by and discussions with representatives of the NRC staff regarding proposed Revision 1C to NUREG-1536, "Standard Review Plan for Spent Fuel Storage Systems at a General License Facility", and the NRC's staff resolution of public comments.
	9:50 - 10:00 **~~9:30~~ – ~~9:45~~ A.M.**	***** BREAK *****
3)	~~9:45~~ 10:00 – 11:15 A.M.	Preparation for Meeting with the Commission on June 9, 2010 (Open) (SAK, et al. /EMH, et al.) Discussion of the topics for meeting with the Commission on June 9, 2010.
	11:15 – 11:30 A.M.	***** BREAK *****
4)	11:30 – ~~12:00~~ 12:05 P.M.	Meeting with the NRC Chairman (Open) (SAK/EMH) 4.1) Remarks by the ACRS Chairman 4.2) Discussions with NRC Chairman Gregory B. Jaczko on topics of mutual interest.
	12:05 **~~12:00~~ – 1:00 P.M.**	***** LUNCH *****
5)	1:00 – ~~4:00~~ 5:00 P.M.	BWR Owners Group (BWROG) Topical Report NEDC-33347P, "Containment Overpressure (COP) Credit for Net Positive Suction Head (NPSH)" and the Staff's Proposed Guidance for the Use of COP (Open/Closed) (WJS/ZA) 5.1) Remarks by the Subcommittee Chairman 5.2) Briefing by and discussions with representatives of the

NRC staff and the BWR Owners Group regarding the staff's review of Revision 0 to Topical Report NEDC-33347P, "COP Credit for Net Positive Suction Head," and staff's proposed guidance for the use of COP.

[NOTE: A portion of this session may be closed to protect information that is proprietary to General Electric-Hitachi pursuant to 5 U.S.C. 552b (c)(4).]

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

6) ~~4:15~~ – 7:00 P.M. Preparation of ACRS Reports (Open)
5:45
Discussion of proposed ACRS reports on:

6:44-7:12p 6.1) Revision 1C to NUREG-1536, "Standard Review Plan for Spent Fuel Storage Systems at a General License Facility" (MTR/CLB)

5:49-6:43p 6.2) BWROG Topical Report NEDC-33347P, "COP Credit for Net Positive Suction Head (NPSH)" and the Staff's Proposed Guidance for the Use of COP (WJS/ZA)

FRIDAY, MAY 7, 2010, CONFERENCE ROOM T-2B1, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

8) 8:35 – ~~9:30~~ A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed) (SAK/EMH)
10:00

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.

10:00 a- 3:00p
Interviews with ACRS membership applicants

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.]

4:00
9) 3:00 – ~~3:15~~ P.M. Reconciliation of ACRS Comments and Recommendations (Open) (SAK/CS/AFD)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

3:15 – 3:30 P.M. ***** BREAK *****

10) 3:30 – 7:00 P.M. Preparation of ACRS Reports (Open)
Continue discussion of the proposed ACRS reports listed under Item 6. 6.1 – 4:00-4:45p
6.2 – 5:00-7:20p

SATURDAY, MAY 8, 2010, CONFERENCE ROOM T-2B1, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

11) 8:30 – 12:30 P.M. Preparation of ACRS Reports (Open)
Continue discussion of the proposed ACRS reports listed under Item 6. There may be 15 minutes breaks at some points during this activity. 6.2 – 8:30-10:53a

12) 12:30 – 1:00 P.M. Miscellaneous (Open) (SAK/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.
Prep for Commission Meeting 10:54-11:56

NOTES:

- When appropriate, members of the public and representatives of the nuclear industry may provide their views during the briefings
- 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 III

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
571st FULL COMMITTEE MEETING

May 6-8, 2010

PLEASE PRINT

TODAY'S DATE: May 6, 2010

	NAME	AFFILIATION
1	Steven J. Hammer	Xcel Energy - MNGP
2	John M. Freeman	Exelon Nuclear
3	Allen L Williams	Xcel Energy - MNGP
4	William M. Bentley	TVA-Browns Ferry
5	Alan Wojchouski	Xcel Energy - MNGP
6	FRED EMERSON	GE-HITACHI - BWR06
7	~~George Bacuta~~	
8	Jim RILEY	NEI
9	~~Zeechung Wang~~	~~NRC~~
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Appendix III

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
572nd FULL COMMITTEE MEETING

May 6-8, 2010

PLEASE PRINT

TODAY'S DATE: May 6, 2010

	NAME	NRC ORGANIZATION
1	Michel Call	NMSS/SFST
2	Jim Peterson	NMSS/SFST
3	David Tang	NMSS/SFST
4	Ron Parkhill	" "
5	Elizabeth Thompson	" "
6	Larry Campbell	NMSS/SFST
7	Ryan Nolan	NRO/SBPA
8	Diego Saenz	NRO/DSRA/SRSB
9	Andrew Barto	NMSS/SFST
10	Dennis Damon	NMSS/FCSS
11	GEOFF HORNSETH	NMSS/SFST
12	Jorge Solis	NMSS/SFST
13	Yvette Mabry	OIG
14	Ray Wsn	NMSS/SFST
15	AHSAN SALLMAN	NRR/DSS/SCVB
16	Steve Smith	NRR/DSS/SSIB
17	Clint Ashley	NRO/DSRA/SPCV
18	George Bacuta	NRR/DPR/PLPB
19	Andrea Russell	NRR/DPR/PGCB
20	Bob Dennis	NRR/DSS/SCVB
21	Donnie Harrison	NRR/DRA/APLA
22	Bruce Heida	NRR/DSS/SCVB
23	Tina Ghosh	NRR/DRA/APLA
24	Russ Haskell	NRR/DIRS/IOEB
25	Rick Jervey	RES/DE
26	Zeechung Wang	RES/DE
27	Margie Kotzalas	OEDO
28		

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

May 12, 2010

AGENDA
573rd ACRS MEETING
JUNE 9-11, 2010

WEDNESDAY, JUNE 9, 2010, CONFERENCE ROOM T-2B1, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 – 10:30 A.M. Draft Final Regulatory Guide (RG) 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings above Design-Basis Pressure" (Open) (WJS/ZA)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff regarding draft final RG 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings above Design-Basis Pressure," and the staff's resolution of public comments.

10:30 – 10:45 A.M. ***** BREAK *****

3) 10:45 – 12:00 P.M. Discussion of Topics for Meeting with the Commission (Open) (SAK, et al. /EMH, et al.)
   Discussion of the following topics for meeting with the Commission:
   3.1) Overview
   3.2) Risk-Informed Performance-Based Fire Protection
   3.3) NRC Safety Research Program
   3.4) Draft Guidance for Use of Containment Accident Pressure
   3.5) Rulemaking for Disposal of Depleted Uranium

12:00 – 1:30 P.M. ***** LUNCH *****

4) 1:30 – 3:30 P.M. Meeting with the Commission (Open) (SAK, et al. /EMH, et al.)
   Meeting with the Commission, Commissioners' Conference Room, One White Flint North, to discuss topics listed under Item 3.

3:30 – 3:45 P.M. ***** BREAK *****

5) 3:45 – 5:00 P.M. Proposed Rulemaking on Distribution of Source Materials to Exempt Persons and to General Licensees and Revision of General License and Exemptions (Open) (MTR/NMC)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding a proposed rule on distribution of source materials to exempt persons and to general licensees and revision of general license and exemptions.

5:00 – 5:15 P.M. * BREAK ***

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

Discussion of proposed ACRS reports on:

6.1) Draft Final RG 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings above Design-Basis Pressure" (WJS/ZA)

6.2) Proposed Rulemaking on Distribution of Source Materials to Exempt Persons and to General Licensees and Revision of General License and Exemptions (MTR/NMC)

6.3) Response to the NRC Executive Director for Operations (EDO) regarding NUREG-1520, Revision 1, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility" (MTR/DAP/NMC)

THURSDAY, JUNE 10, 2010, CONFERENCE ROOM T-2B1, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

8) 8:35 – 10:00 A.M. Proposed Interim Staff Guidance (ISG) DC/COL-ISG-013, "Assessing the Consequences of an Accidental Release of Radioactive Materials from Waste Tanks," and Proposed DC/COL-ISG-014, "Assessing Groundwater Flow and Transport of Accidental Radionuclide Releases" (Open) (MTR/DAW)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff regarding proposed ISG-013, "Assessing the Consequences of an Accidental Release of Radioactive Materials from Waste Tanks," and proposed ISG-014, "Assessing Groundwater Flow and Transport of Accidental Radionuclide Releases."

10:00 – 10:15 A.M. * BREAK ***

9) 10:15 – 12:00 P.M. Status of Risk-Informing Guidance for New Reactors (Open) (JWS/JCL)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussions with representatives of the NRC staff regarding the current status of risk-informing guidance for new reactors.

12:00 – 1:00 P.M. ***** LUNCH *****

10) 1:00 – 2:30 P.M. Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Performance" (Open) (SB/WW)

10.1) Remarks by the Subcommittee Chairman

10.2) Briefing by and discussions with representatives of the NRC staff regarding the current status towards resolution of GSI-191.

2:30 – 2:45 P.M. ***** BREAK *****

11) 2:45 – 4:15 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed) (SAK/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) 4:15 – 4:30 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (SAK/CS/AFD)

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

4:30 – 4:45 P.M. ***** BREAK *****

13) 4:45 – 7:00 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

13.1) Draft Final RG 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings above Design-Basis Pressure" (WJS/ZA)

13.2) Proposed Rulemaking on Distribution of Source Materials to Exempt Persons and to General Licensees and Revision of General License and Exemptions (MTR/NMC)

13.3) Response to the EDO regarding NUREG-1520, Revision 1, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility" (MTR/DAP/NMC)

13.4) Proposed ISG-13, "Accidental Release of Radioactive Materials from Waste Tanks" and proposed ISG-14, "Assessing Groundwater Flow and Transport of Accidental Radionuclide Releases" (MTR/DAW)

FRIDAY, JUNE 11, 2010, CONFERENCE ROOM T-2B1, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 12:00 – 2:00 P.M. Preparation of ACRS Reports (Open)
Continue discussion of the proposed ACRS reports listed under Item 13.

15) 2:00 – 2:30 P.M. Miscellaneous (Open) (SAK/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:

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APPENDIX V

LIST OF HANDOUTS
572nd ACRS MEETING
MAY 6-8, 2010

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

II. Revision 1C to NUREG-1536, "Standard Review Plan for Spent Fuel Storage Systems at a General License Facility"
3. Proposed Schedule
4. Status Report
5. SRP
6. ISG – 25

V. BWR Owners Group (BWROG) Topical Report NEDC-33347P, "Containment Overpressure (COP) Credit for Net Positive Suction Head (NPSH)" and the Staff's Proposed Guidance for the Use of COP
7. Table of Contents
8. Subcommittee Status Report – April 23, 2010
9. Budris-Pump Specialist Reports
10. Draft Staff Guidance for COP Credit

Updating the Standard Review Plan for Dry Storage Systems (NUREG-1536)

Briefing for the ACRS Full Committee

Division of Spent Fuel Storage and Transportation
Office Of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission

- May 6, 2010 -

Opening Remarks

Raymond Lorson, Deputy Director
Meraj Rahimi, Acting TCB Branch Chief
Technical Review Directorate
Division of Spent Fuel Storage and Transportation (SFST)

Purpose

- Brief the ACRS on the update to the Standard Review Plan (SRP) for Spent Fuel Dry Storage Systems at a General License Facility (NUREG-1536)

- Obtain ACRS comments and concurrence to publish

Briefed Subcommittee on February 17, and April 21, 2010

- Revision to NUREG-1536, Jan 1997
- Dry Cask Storage Background
  - Division of Spent Fuel Storage and Transportation
  - Typical Dry Cask Storage Operations
  - Regulatory Basis and Design Basis
- Std. Review Plan (SRP) Update Project
  - Overall Project Approach
  - Prioritization Method
  - Key Revisions to SRP per Chapter
  - Key Stakeholder Comments per Chapter

Incorporated ACRS Subcommittee Comments to Improve the SRP

- Replaced “Risk Informed” with Prioritized
- Identified Polymeric Neutron Shielding Materials as Important-To-Safety
- Changed prioritization method to reevaluate ‘Low’ or ‘Very Low’ ratings for Question #2
- Changed ‘catastrophic’ consequence to significant consequence from rating criteria of prioritization method
- Clarified reason for burnup measurements associated with granting burnup credit

Today's Presentation

- SRP Update Project & Public Comments (Ron Parkhill)
- Prioritization Methodology (Dennis Damon/Ron Parkhill)
- Radiation Protection (Elizabeth Thompson)
- Spent Fuel Oxidation (Ron Parkhill)
- Burnup Credit Measurements (Meraj Rahimi)

SRP Update Project

- SRP (NUREG 1536) issued 13 years ago (1/97)
- The update project started 4 years ago to
  - Incorporate applicable interim staff guidance documents & other necessary changes
  - Prioritize the review procedures section of each SRP chapter to better utilize staff resources
  - Develop a new materials chapter
  - Enhance knowledge transfer

Public Comments

- Only received comments from industry (NEI 192 and NAC 30 - mostly duplicated)
- Listed & Dispositioned in Appendix D to SRP
- Staff agreement on over 60% of the comments
- Resulted in better SRP

Prioritization Methodology

Dennis Damon, Sr. Level Advisor for Risk Assessment, FCSS

Ron Parkhill, Sr. Mechanical Engineer, SFST

Prioritization Methodology - focuses staff resources

- Focuses staff resources by assigning high, medium or low to items in the Review Procedures
- Standard Review Plan Chapter Structure
  - Review Objective
  - Areas of Review
  - Regulatory Requirements
  - Acceptance Criteria
  - Review Procedures (Prioritized)
  - Evaluation Findings

Prioritization Methodology

(1) Likelihood that requirement will not be met	**VH**=4, likely to occur, P>.5 **H**=3, Probably will occur, 0.1< P < 0.5 **M**=2, May occur, 0.03< P< 0.1 **L**=1, Unlikely to occur, 0.01<P< 0.03 **VL**=0, Occurrence improbable P< 0.01
(2) Likelihood that staff review will find the discrepancy	Same as (1)
(3) Risk if requirement not met	**H**=3, Likely to occur or significant consequences, >10^{-3}/yr or 25 rem to worker or 1 rem to public **M**=2, may occur or moderate consequences, <10^{-3}/yr but >10^{-5}/yr or 5-25 rem to worker or 0.1-1 rem to public **L**=1, Occurrence improbable or marginal consequences, < 10^{-5}/yr or less than 10CFR 20 dose limits for workers & public
(4) Add scores from (1), (2) & (3) to get combined Risk score	**High** is 9 to 11 **Medium** is 6 to 8 **Low** is 1 to 5

Prioritization Methodology (Cont'd.)

(5) Determine Defense in Depth- a review procedure impacts DinD if it provides a back up to the first line of defense (e.g. confinement is back up to cladding integrity)	If failure to perform a review procedure could impact DinD (assuming front line safety measure has failed) and has -a low, medium or high likelihood and/or consequence, then the item should be a low, medium or high, respectively -Same as (3) for low, medium or high Note most SRP review procedure items don't have DinD
(6) Determine which controls (or is more important) DinD (Step 5) or Risk (Step 4)	Assign controlling rating from DinD or Risk

Summary of Prioritization Results

Chapter	HIGH	MEDIUM	LOW	Total
1) General Info	----	4	----	4
2) Principle Design Criteria	----	4	1	5
3) Structural	6	13	7	26
4) Thermal	6	7	5	18
5) Confinement	----	5	2	7
6) Shielding	5	3	----	8
7) Criticality	11	3	1	15
8) Materials	7	13	7	27
9) Operating Procedures	2	5	4	11
10) Acceptance tests and Maintenance Program	5	3	7	15
11) Radiation Protection	----	4	----	4
12) Accident Analyses	----	1	----	1
13) Technical Specifications and Operational Controls & Limits	1	----	----	1
14) Quality Assurance	1	----	----	1
TOTALS	**44**	**65**	**34**	**143**

Radiation Protection in Dry Storage System Licensing and Operations

Elizabeth Thompson, CHP
Sr. Health Physicist, SFST

Radiation Protection

- SRP Revision
  - Incorporated ISGs
  - Few other changes
- Typical Doses
  - Per cask loaded
  - Per campaign
  - Trends

Fuel Oxidation

Dr. Robert Einziger,
Sr. Materials Engineer, SFST
Ron Parkhill,
Sr. Mechanical Engineer, SFST

Fuel Oxidation

- Fuel Integrity
  - Clad Splitting (unzipping) when spent fuel is exposed to air
  - >30% expansion upon oxidation of UO_2 to U_3O_8
- Confinement of Radioactive Material
  - Open grain boundaries in the fuel can release fission products
- To date all casks have been licensed for storage in an inert gas.

Fuel Oxidation

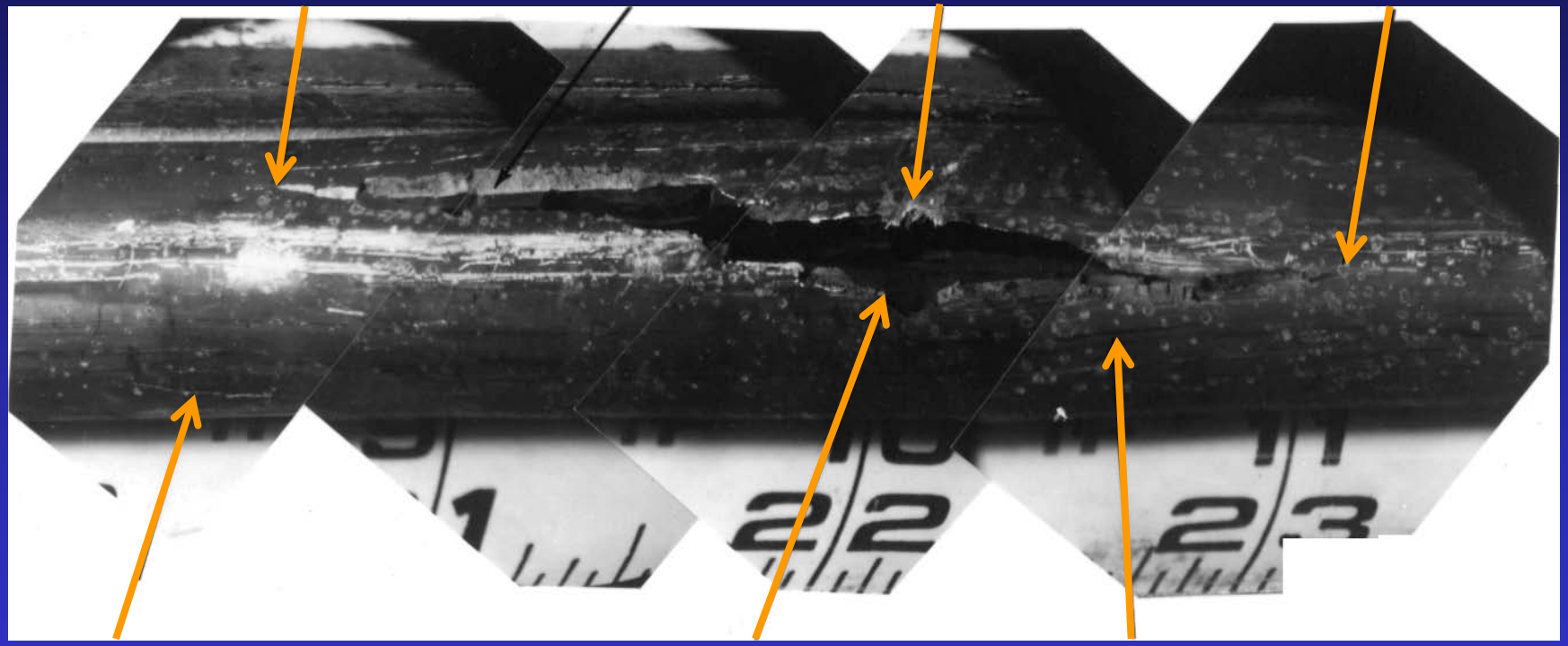
“Effects of an Oxidizing Atmosphere in a Spent Fuel Packaging Facility”,
EPRI NP-4524 R.E. Einziger, Sept 1991.

Fuel Oxidation

UPPER CRACK TIP

ORIGINAL DEFECT

LOWER CRACK TIP



SECONDARY CRACK

POINT OF LARGEST CRACK WIDTH

SECONDARY CRACK

"The Performance of Defected Spent LWR Fuel Rods in Inert and Dry Air Storage Atmospheres", Nuclear Engineering and Design, C. S. Olsen, Nov 1985.

Burnup Credit Measurements

Meraj Rahimi,
Acting TCB Branch Chief
Technical Review Directorate, SFST

Burnup Credit Measurements

- NEI Comment #162 - delete performing measurements to confirm assembly burnup values

  -Changed Response to clarify SRP that a confirmatory measurement, which would be based on reactor records, is needed to prevent misloading of assemblies that do not meet the loading criteria for burnup credit

Summary

- Incorporated several ISG's
- Updated to reflect current review practices
- Added new materials chapter
- Prioritized the review procedures
- Resolved public (industry) comments
- Improved safety focus of certification reviews

Path Forward

- Issue Final SRP Revision 1 (June)
- Continue Work on SRP for Storage Facilities (NUREG-1567)

USE OF CONTAINMENT ACCIDENT PRESSURE IN DETERMINING THE AVAILABLE NET POSITIVE SUCTION HEAD (NPSH) OF EMERGENCY CORE COOLING SYSTEM AND CONTAINMENT HEAT REMOVAL PUMPS

May 6, 2010

ACRS

Sher Bahadur

Deputy Director,

Division of Safety Systems

NRR

Introduction-1

- Recent staff/ACRS discussions of containment accident pressure:
  - November 15-16, 2005 Vermont Yankee EPU (Power Uprate Subcommittee)
  - December 7, 2005 Vermont Yankee EPU
  - February 1, 2007 Browns Ferry Unit 1 5% power uprate
  - December 4, 2008 staff white paper
  - March 18, 2009, ACRS letter with recommendations
  - Suspended review of use of containment accident pressure in Browns Ferry (September 18, 2009) and Monticello (October 1, 2009) EPU license amendment requests
  - April 23, 2010 draft staff guidance (Power Uprate Subcommittee)

Introduction-2

- Staff study of use of containment accident pressure in response to ACRS March 18, 2009, letter
- Major work accomplished:
  - Review of BWROG topical report on use of containment accident pressure
  - Consulted with experts in cavitation, NPSH and pump hydraulics
  - Performed calculations to study sensitivities and margins
  - Performed risk assessment
  - Prepared draft guidance on use of containment accident pressure based on the above work
- April 14, 2010 meeting with BWROG

Introduction-3

- Presentations:
  - Risk Aspects
  - Containment Integrity
  - Draft Guidelines
- Staff requests letter
- Future Actions
  - Finalize guidance with BWROG and PWROG and other stakeholders
  - Resume two EPU reviews
  - Document guidance

ACRS / BWROG - Containment Accident Pressure

Alan Wojchouski
Xcel Energy
BWROG CAP Committee Chair

May 6, 2010

Summary

I. Tech Specs and operator monitoring would detect a loss of containment integrity that affects NPSH margin. Tech Specs require prompt actions

II. Realistic analysis shows that for loss of containment integrity with a DBA LOCA, credit for containment accident pressure is not needed

III. For design basis and beyond design basis events, loss of primary containment defense barrier can be mitigated prior to loss of the fuel defense barrier

- Operator actions to mitigate loss of containment integrity can be taken to assure adequate NPSHa for ECCS pump operation and core cooling

I. Containment Integrity

Programs exist to assure containment leakage does not approach the level needed to lose NPSH margin

- Containment designed to maintain its integrity
- Operating license requirements assure containment integrity will be maintained:
  - Primary Containment Isolation System including valve alignment verification
  - Appendix J Test Program
  - Containment Boundary In-service Inspection Program
  - Operational containment leakage monitoring
  - Technical Specifications require prompt shutdown if containment integrity lost during operation

I. Appendix J and Technical Specifications

Appendix J Program requires testing of containment leakage paths at each refueling outage

- Tech Spec requirement: containment leakage is 1.0 La (leak acceptance criteria)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Primary containment inoperable.	A.1	Restore primary containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours

I. Operational Containment Leakage Monitoring for Pre-Existing Leaks

Gross containment leakage of magnitude sufficient to affect NPSH margin would be monitored by:

- Drywell pressure
- Containment oxygen concentration
- Nitrogen consumption

I. Operational Containment Leakage Monitoring for Pre-Existing Leaks

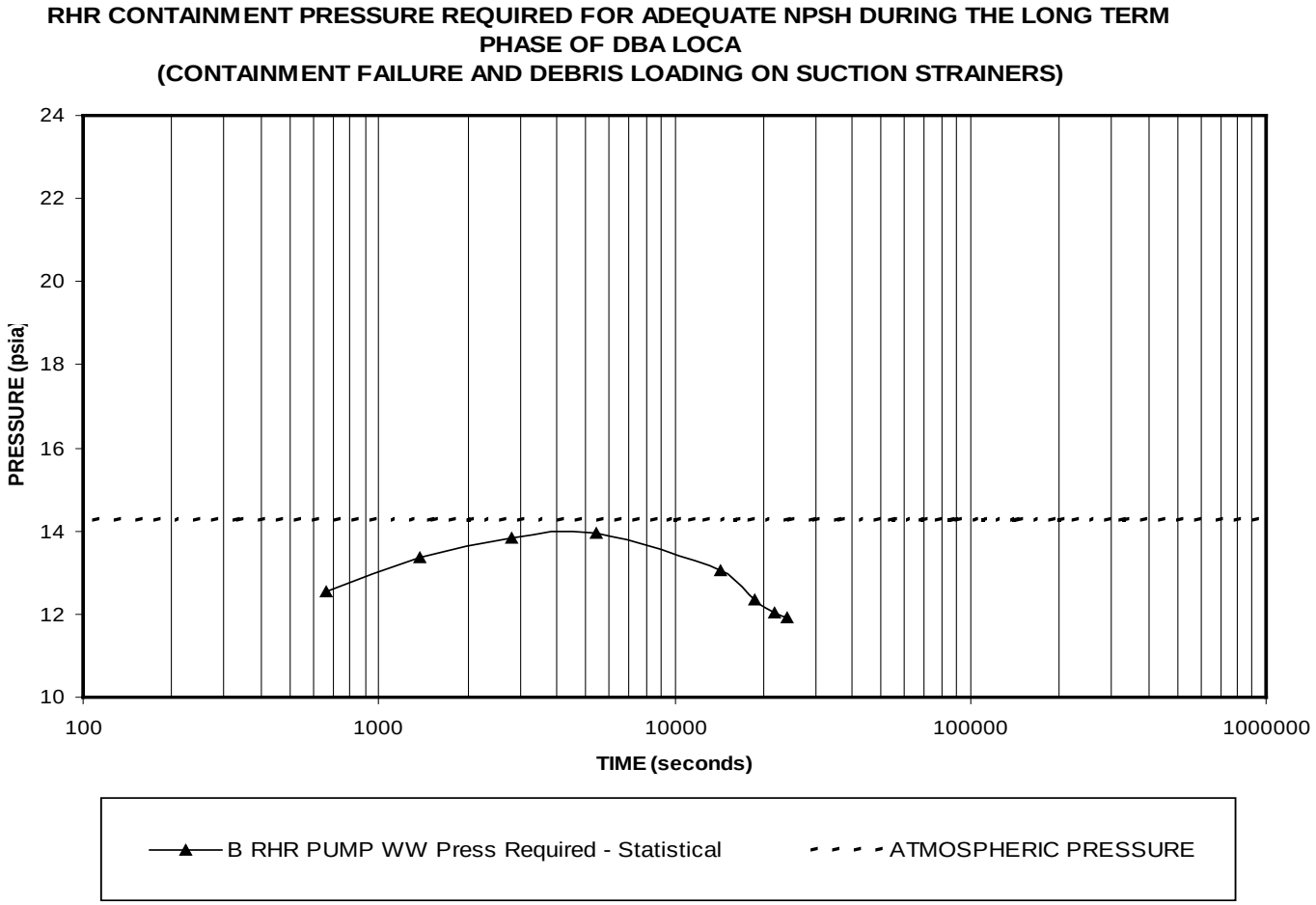
Drywell pressure and Excess Nitrogen Make-up

- Drywell pressure maintained between 0.1 psig and 1.5 psig
- Alarms for values outside the allowable range
- Excess use of nitrogen make-up indicates a possible containment leakage problem
- Operators would confirm and initiate an investigation per corrective action program

Containment oxygen concentration

- Technical Specifications requires that primary containment oxygen concentration shall be < 4.0 %
- Inability to maintain oxygen within limits indicates a possible containment leakage problem
- Operators would confirm and initiate an investigation per corrective action program

II. Realistic Analysis with Loss of Containment Integrity (NEDC-33347P)



III. Operator Actions Can Mitigate Loss of Containment Integrity During Events

Long-term NPSH can be maintained by operator actions by increasing suppression pool water level when there is no accident pressure available

Alternate injection systems can increase suppression pool water level and maintain reactor vessel water level

- Backup to ECCS following initial core flooding
- EOPs at example plant provide instructions to use alternate water sources such as RHRSW, Condensate Service Water, Fire Water, Service Water as available

Summary

I. Tech Specs and operator monitoring would detect a loss of containment integrity that affects NPSH margin. Tech Specs require prompt actions

II. Realistic analysis shows that for loss of containment integrity with a DBA LOCA, credit for containment accident pressure is not needed

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- Operator actions to mitigate loss of containment integrity can be taken to assure adequate NPSHa for ECCS pump operation and core cooling

USE OF CONTAINMENT ACCIDENT PRESSURE IN DETERMINING THE AVAILABLE NET POSITIVE SUCTION HEAD (NPSH) OF EMERGENCY CORE COOLING SYSTEM AND CONTAINMENT HEAT REMOVAL PUMPS

May 6, 2010

ACRS

Richard Lobel

Ahsan Sallman

Containment Accident Pressure vs. Overpressure

- Staff uses the term *containment accident pressure.*
- No system, structure or component is being overpressurized.
- *Overpressure* has been used several different ways:
  - Pressure greater than atmospheric pressure
  - Pressure greater than saturation pressure
  - Pressure greater than containment pressure prior to postulated accident (containment accident pressure)
  - BWROG topical report uses the first definition
- Containment accident pressure is greater than the containment pressure prior to the postulated accident

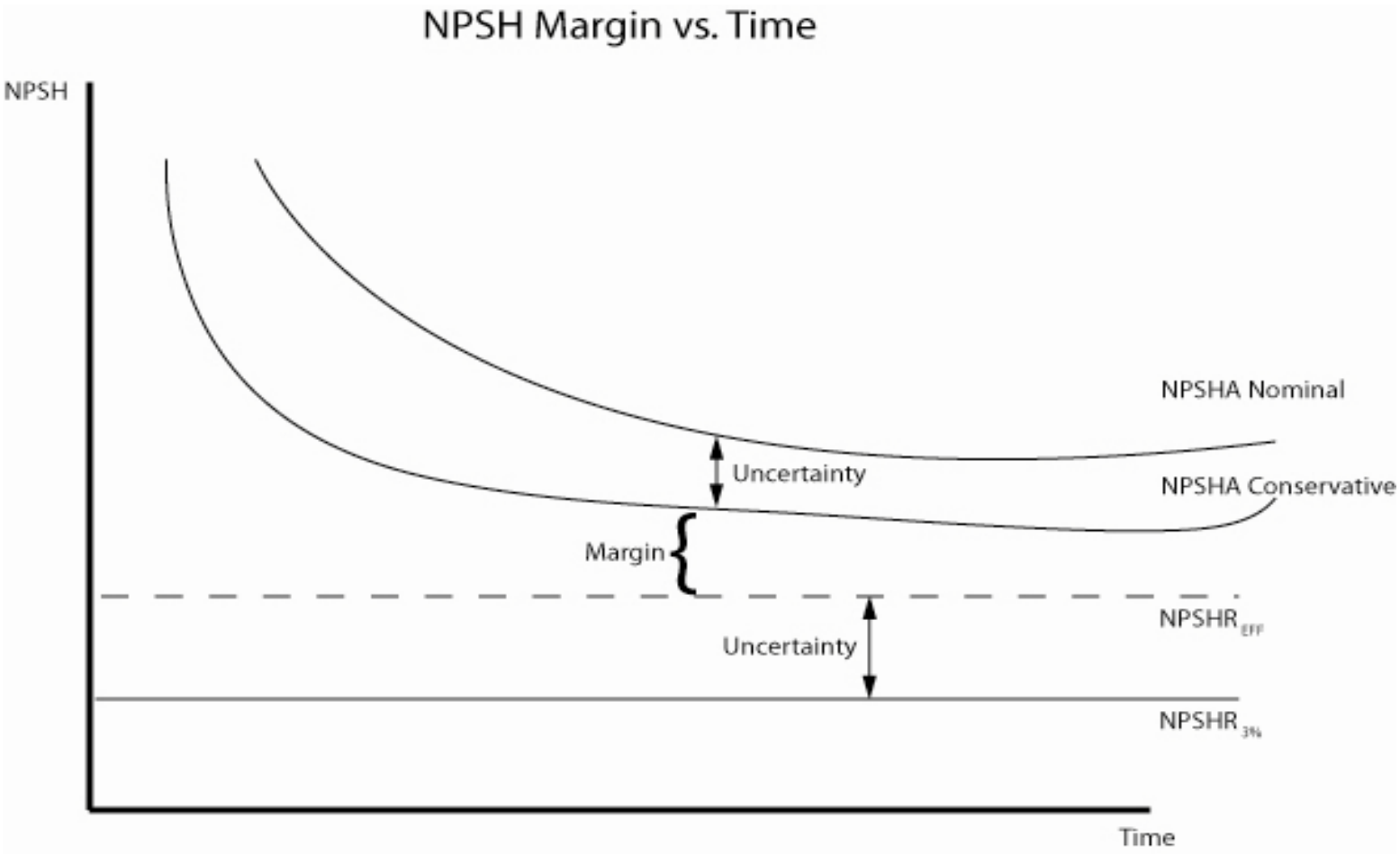
Status

- 27 operating reactors use containment accident pressure for determining available NPSH
  - 19 BWRs
  - 8 PWRs
- Extended power uprates:
  - Two EPU reviews on hold pending revised guidance on use of containment accident pressure for available NPSH

Proposed Guidance Key Ideas

- Design basis and “special events” are considered
- Containment integrity is assumed
  - New criterion proposed to ensure against pre-existing leak
  - Appendix R Fire associated circuits will not result in loss of containment integrity
  - Venting by procedure will not occur when containment accident pressure is needed
- Staff guidelines focus on pump performance (NPSH margin and adverse effects on pump)
- Quantitative guidelines are proposed

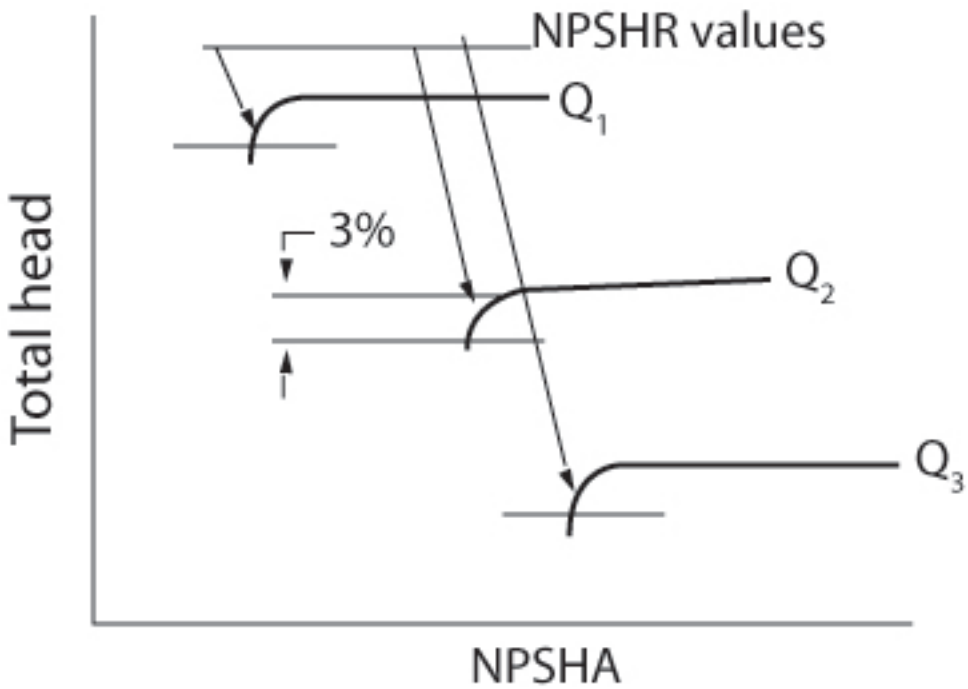
Margin and Uncertainty



Cavitation

- Cavitation is the formation of vapor in a liquid due to a decrease in the local static pressure followed by an increase in local static pressure which results in the sudden condensation of the vapor.
  - Occurs at constant liquid temperature.
- Excessive pump cavitation can result in:
  - Erosion of the pump impeller and other pump parts
  - Mechanical damage to seals, bearings, shaft, etc.
  - Decrease in pump flow rate
  - Decrease in pump discharge head
  - Vibration
- The degree to which any of these effects adversely affects pump performance depends on the amount of cavitation and its duration, the air/gas content of the liquid, the suction energy of the pump, the NPSH margin, and if the pump is operating in the low flow suction recirculation region.

Required NPSH



NPSHR DETERMINATION WITH SPEED AND FLOW RATE(Q) HELD CONSTANT

- Two natural NPSH values: incipient cavitation and break down.
- Others values of NPSH, such $NPSHR_{3\%}$, specify an arbitrary level of cavitation.

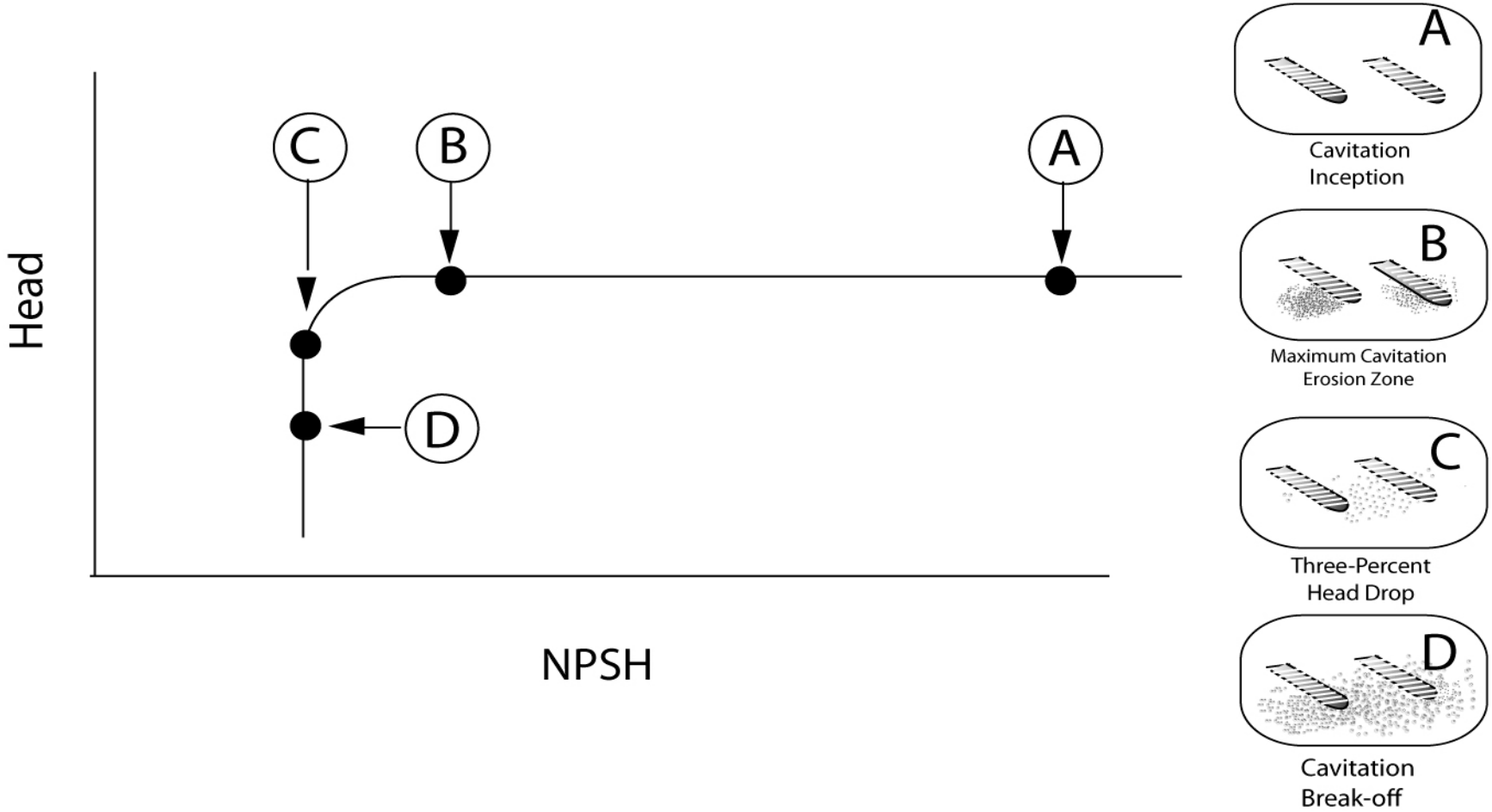
NPSHR Uncertainty

- Pump vendor test results for $NPSHR_{3\%}$ are most accurate. For best accuracy, test should be conducted at rated pump speed and impeller diameter with NPSHA controlled by vacuum pump.
- Additional uncertainty in $NPSHR_{3\%}$ due to field (installed) conditions.
- Installed uncertainty due to:
  - Pump speed
  - Water temperature
  - Suction piping layout
  - Air content of pumped water
  - Wear ring leakage
- Guidance defines a new parameter $NPSHR_{eff}$
  - $NPSHR_{eff}$ = (1.0 + uncertainty) $NPSHR_{3\%}$
  - $NPSHR_{eff}$ should be used in determining NPSHR margin
  - Value of NPSHR to be used in NPSH margin ratio not yet decided
  - For nondesign basis events, $NPSHR_{3\%}$ may be used for required NPSH

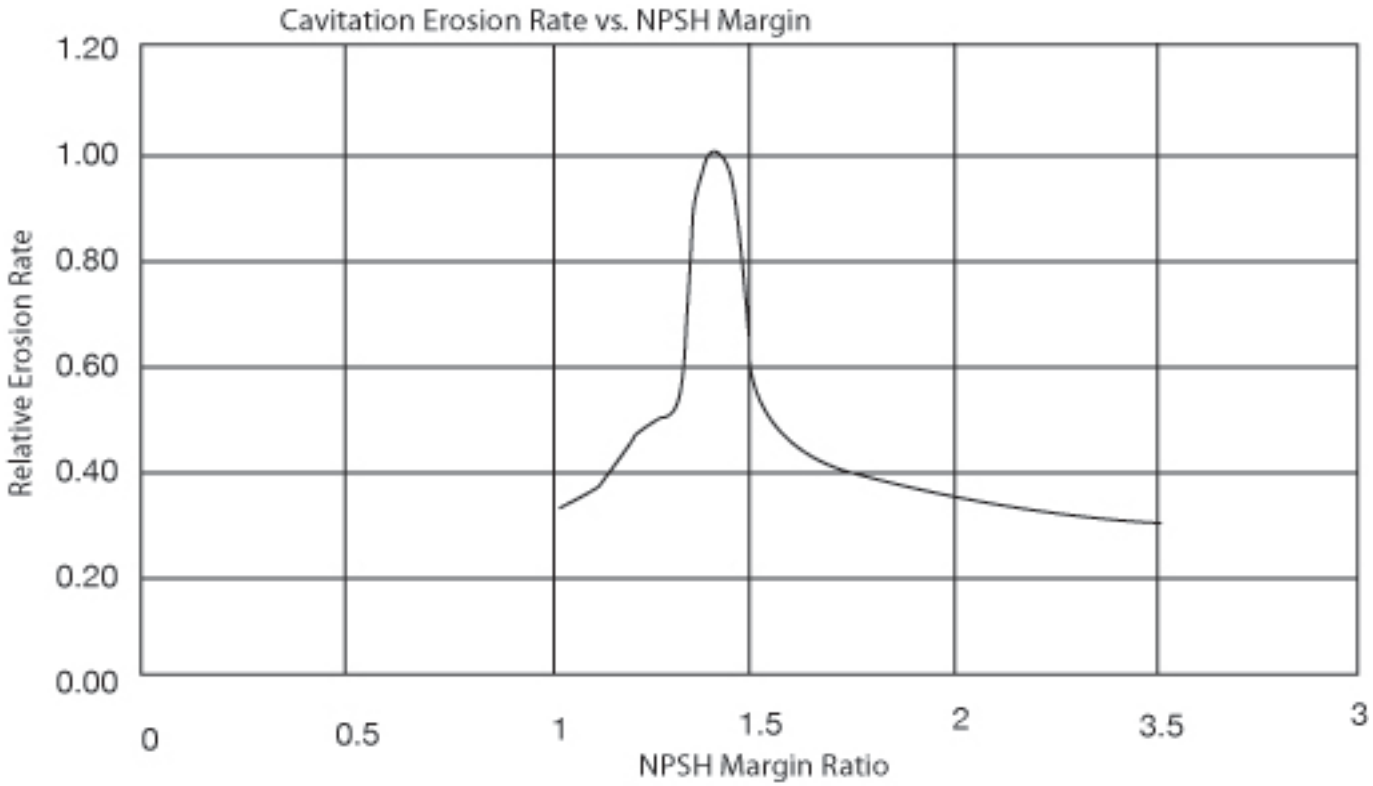
Cavitation Erosion

- Maximum cavitation erosion occurs between incipient cavitation and $NPSHR_{3\%}$.
- Only high and very high suction energy pumps will experience cavitation erosion damage in this zone.
- Using a typical curve of pressure transducer measurements of cavitation, the staff determined that the maximum erosion zone is between NPSH margin ratios of 1.2 and 1.6.
- The staff conservatively selected a limit of 100 hours for very high suction energy pumps as the maximum allowable time for operation in the maximum erosion zone.
  - Pump must continue to function for the remainder of its mission time (up to 30 days)
  - The pump would be at higher available NPSH values (out of maximum erosion zone) during post-accident operation

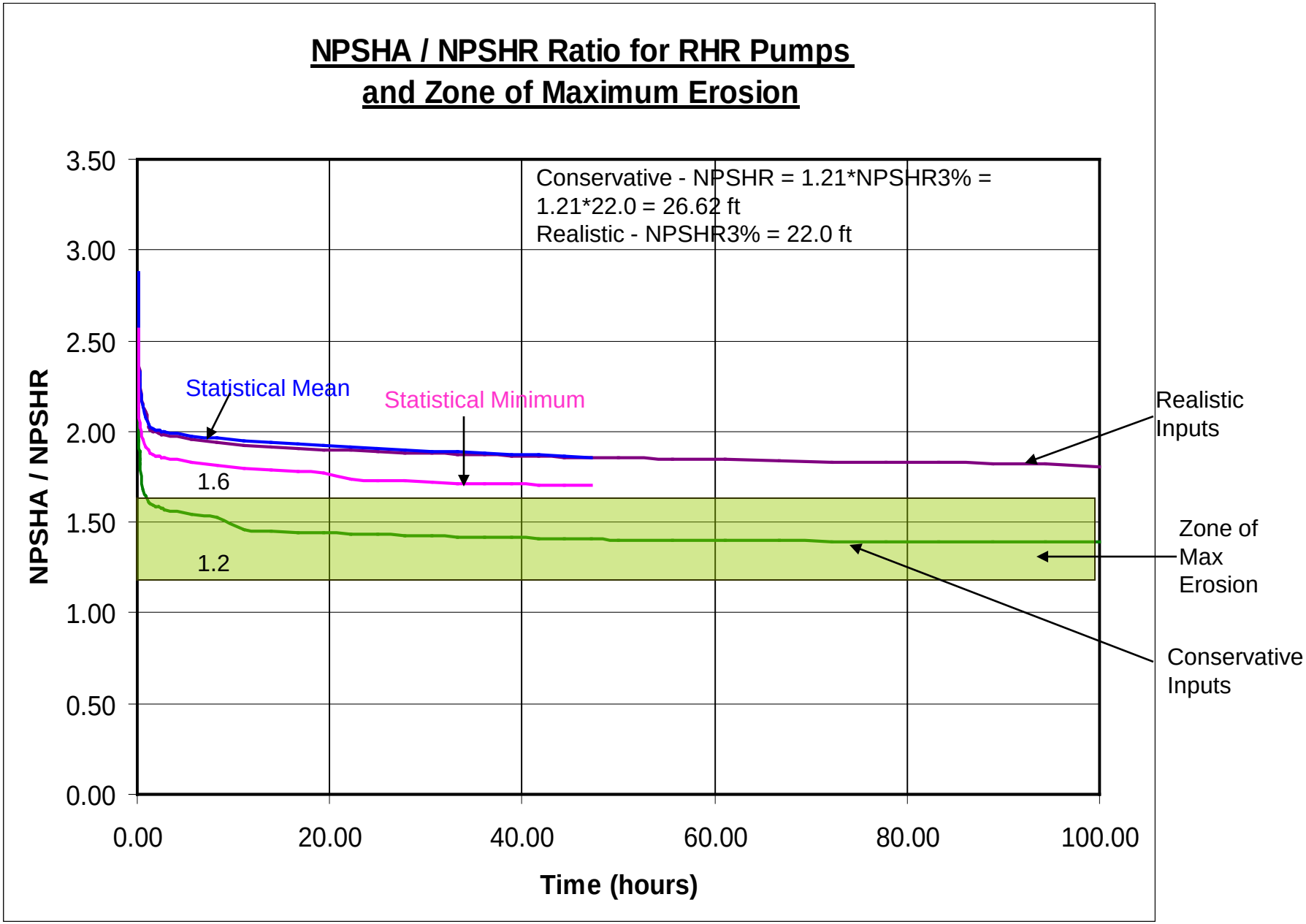
Cavitation Behavior Based on Impeller Voiding



Cavitation Acoustic Signal



Typical Relative Erosion Rate vs. NPSH Margin near BEP Flow Rate


NPSHA / NPSHR Ratio for RHR Pumps
and Zone of Maximum Erosion
Conservative - NPSHR = 1.21*NPSHR3% =
1.21*22.0 = 26.62 ft
Realistic - NPSHR3% = 22.0 ft
Statistical Mean
Statistical Minimum
1.6
1.2
Realistic Inputs
Zone of Max Erosion
Conservative Inputs
NPSHA / NPSHR
Time (hours)
3.50
3.00
2.50
2.00
1.50
1.00
0.50
0.00
0.00
20.00
40.00
60.00
80.00
100.00

NPSHA < NPSHR

- It is possible that the predicted available NPSH for an event may be less than the required NPSH.
- RG 1.82 Revision 3 states that predicted operation with NPSHA < NPSHR is acceptable if testing shows that the pump will continue to perform its safety function.
- Staff has developed the following conditions which should apply to testing:
  - Time of predicted operation in cavitation less than 100 hours
  - Tests conducted on actual pump or important pump hydraulic properties the same
    - e.g., same model, size, impeller diameter
  - Tests conducted at same field speed
  - Test conducted with predicted NPSHA
  - Test should be for predicted time that NPSHA < NPSHR
  - Flow rate and discharge head remain above values assumed in core and containment cooling analyses
  - No damage or excessive wear to pump components

Protection of Mechanical Seals

- A concern with operating a pump at or below the 3% NPSHR condition is the damage that the water vapor and/or entrained air could do inside the pump to the mechanical shaft seal faces, which could fail in a very short time if the seal faces run dry.

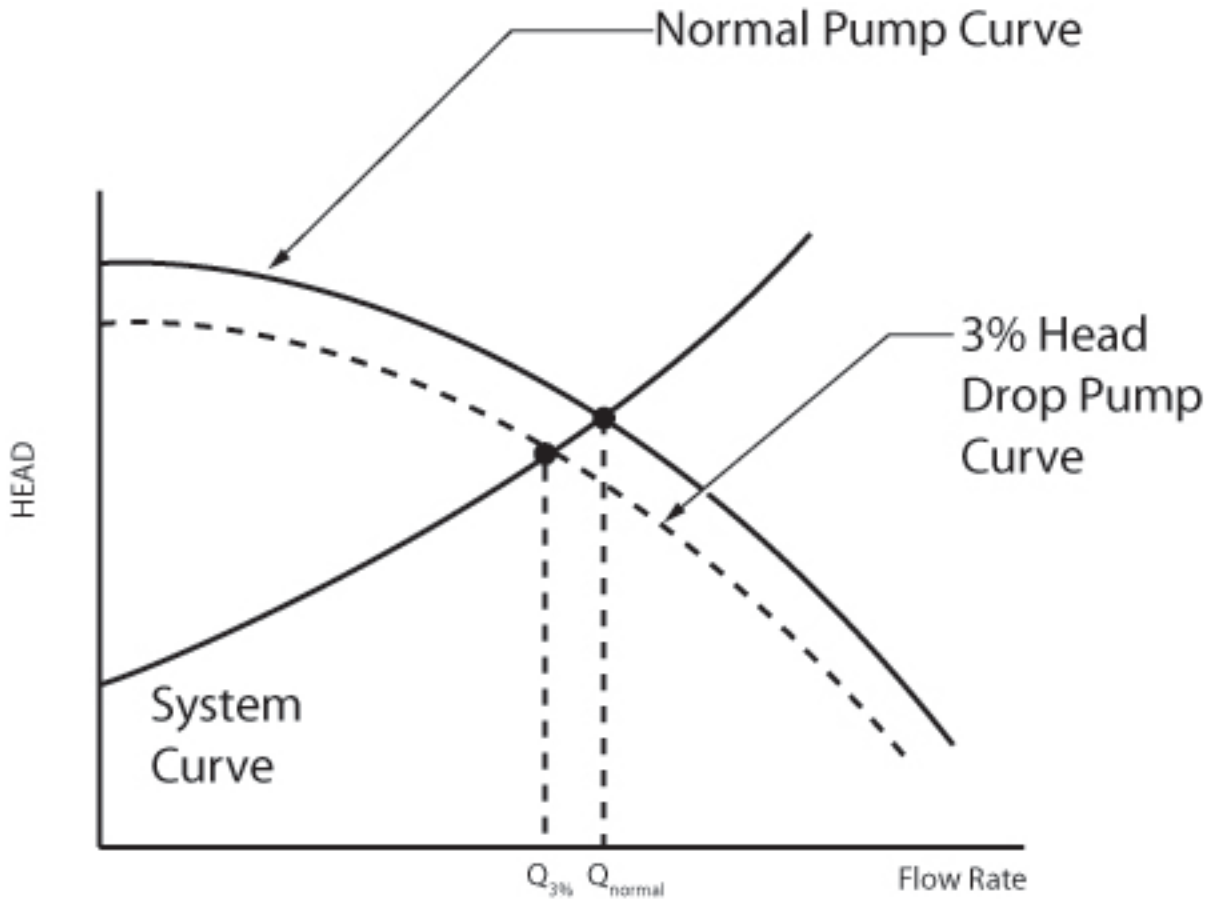
- Excessive entrained air may accumulate around the shaft, where the mechanical seal is housed.

- This additional entrained air comes from the dissolved air that comes out of solution as local static pressure drops to the vapor pressure and cavitation vapor bubbles are formed.

- To protect the mechanical seal faces from this excess entrained air (under operation at or below the 3% NPSHR condition), dual mechanical seals with an external cold water flush system (or equivalent) should be provided.
  - Necessity for this guideline may require individual pump evaluation

Pump Flow Rate

- The flow rate chosen for the NPSH analysis should be greater than or equal to the flow rate assumed in the safety analysis that demonstrates adequate core and containment cooling
- If the assumption that NPSHA = $NPSHR_{eff}$ or $NPSHR_{3\%}$ is used to determine the containment accident pressure used, then the pump flow rate used in the core and containment cooling analyses should be the flow rate resulting from a 3% decrease in pump total dynamic head


Normal Pump Curve
3% Head
Drop Pump
Curve
HEAD
System
Curve
$Q_{3\%}$
Q_{normal}
Flow Rate

Duration of Need for Containment Accident Pressure

- Staff concludes that a limit on the duration of use of containment accident pressure is not needed.
  - Not supported by risk analysis
  - Would be arbitrary (no technical basis)
- Staff has proposed a time limit in zone of maximum erosion rate of 100 hours

Reducing Containment Pressure by Cooling the Containment Atmosphere

- Licensee analysis should demonstrate that operation of sprays or fan coolers will not cause containment accident pressure to be less than that needed for adequate available NPSH.
  - BWR procedures contain a caution to this effect.
  - BWR calculations assume spray operation during time containment accident pressure is used.

- Restarting of spray pumps and fan coolers: plant specific

Proposed Containment Leakage Rate Surveillance

- To reduce the likelihood of a preexisting leak, licensees proposing to use containment accident pressure in determining NPSH margin should:
  - (i) Determine the minimum containment leakage rate sufficient to lose the containment accident pressure needed for adequate NPSH margin.
    - Staff and BWROG calculations predict this would be 40La.
  - (ii) Propose a method to determine if the actual containment leakage rate exceeds the leakage rate determined in (i) above.
    - For inerted containments, this method could consist of a periodic quantitative measurement of the nitrogen makeup performed at an appropriate frequency to ensure that no unusually large makeup of nitrogen occurs.
    - Monitoring oxygen content is another method.
    - For subatmospheric containments, a similar procedure might be used.
  - (iii) Propose a limit on the time interval that the plant operates when the actual containment leakage rate exceeds the leakage rate determined in (i) above.

Available NPSH (NPSHA) and Containment Analysis

- To determine available NPSH we must know:
  - containment pressures (drywell and wetwell)
  - containment temperatures (BWR drywell and wetwell atmospheres and suppression pool water)
  - water level above the pump suction
- Calculated with containment thermal hydraulic analysis code
- Staff guideline: Two calculations should be done to demonstrate margin in available NPSH:
  - Conservative
  - Realistic

NPSHA Staff Calculations

- Purpose of staff calculations
  - Prepare calculation models for future reviews
  - Study sensitivity of available NPSH and NPSH margin to input parameters, assumptions and calculation models
- Staff performed realistic, conservative and Monte Carlo calculations
- Realistic, conservative and Monte Carlo calculations assumed worst single failure
- Staff calculations use GOTHIC
- Staff calculations are not meant to reflect expected licensing calculation results
  - Staff does not have complete plant data, especially for more realistic input parameters
    - BWROG and licensees have provided the information the staff requested
  - Some staff calculations performed in a different way than BWROG: methods (computer codes) and models (e.g., modeling of decay heat for statistical models, heat transfer coefficients)
  - However, conclusions are considered generally applicable to BWRs with Mark I containments so that guidance can be formulated

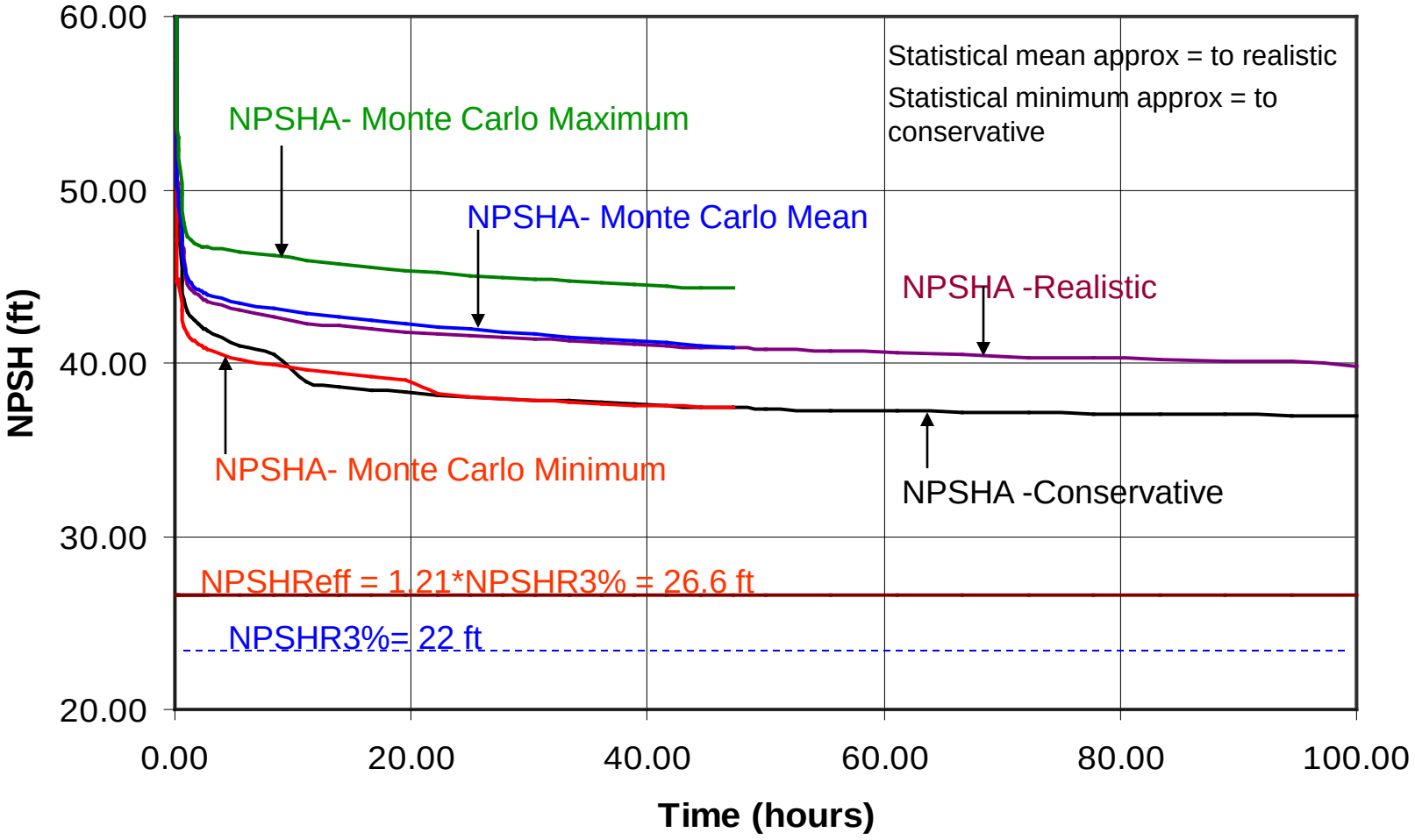
NPSHA Sensitivity Studies

- Staff performed sensitivity studies of input variables to NPSH calculation for a LOCA in a BWR/4 with a Mark I containment
- Most variables varied by 5%.
- Conservative inputs used for base case (except 100% power instead of 102%)

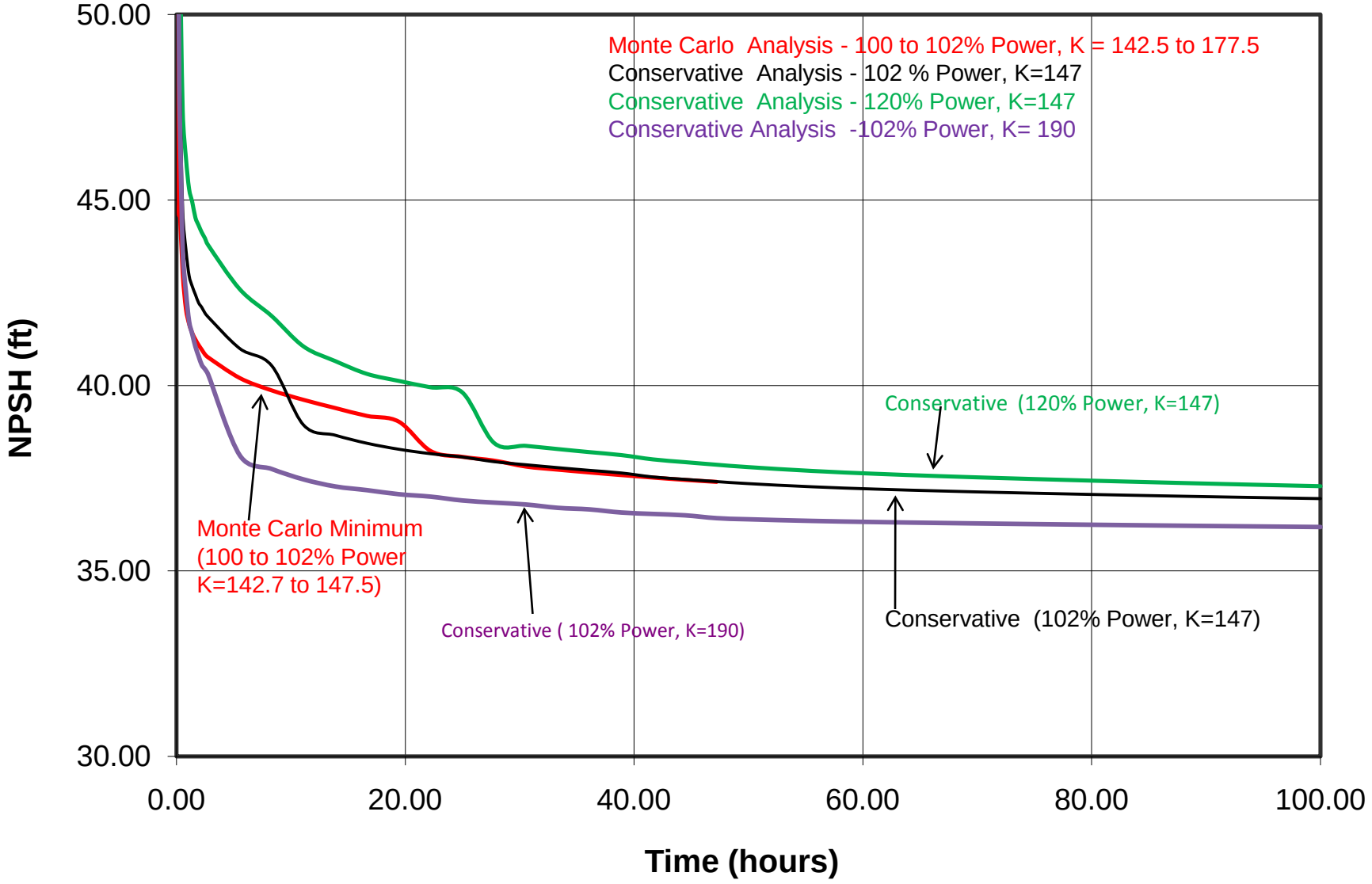
BWR-4 MARK I CONTAINMENT LONG TERM LOCA NPSH ANALYSIS SENSITIVITY STUDY RESULTS SUMMARY

No	Parameter	Base Value	Compared Value	Change in Parameter Value (%) (Note 1)	Maximum Change in Supp Pool Temp (%)	Maximum Change in Wetwell Pressure (%)	Maximum Change in Available NPSH (%)
1	Power (percent)	100	95	-5	-2.34	-5.47	-4.24
2	Core Spray Flow (gpm)	3027	2876	-5	-0.17	1.12	2.67
3	Initial Drywell Pressure (psia)	14.26	14.97	5	-0.1	2.02	2.53
4	Initial Wetwell Pressure (psia)	14.26	14.97	5	-0.2	2.16	2.32
5	Initial Supp Pool Temp (deg F)	90	85.5	-5	-2.93	-3.89	-2.27
6	Service Water Temperature (deg F)	90	85.5	-5	-2.63	-3.83	-2.26
7	RHR HX K-Value (Btu/sec deg F)	147	139.65	-5	2.76	4.89	2.14
8	Initial Drywell Temperature (deg F)	135	128.25	-5	-0.12	1.58	2.02
9	Initial Torus Liquid/Volume Ratio	0.3858	0.4051	5	-1.82	-3.67	1.29, -0.96
10	Reactor thermal conductors area reduced by 5%	100%	95%	-5	-0.38	-1.11	-0.98
11	Drywell Spray Flow	3800	3610	-5	-0.08	0.77	0.88, -0.22
12	Strainer & Piping Loss (ft)	5.79	5.5	-5	0	0	0.78
13	Initial Drywell Relative Humidity (%)	100	95	-5	-0.09	0.44, -0.72	0.67, -0.76
14	Wetwell Spray Flow	200	190	-5	-0.01	0.34, -0.08	0.54, -0.09
15	Decay Heat (sigma)	2	1.9	-5	-0.12	-0.22	0.21, -0.45
16	Containment Leakage (Weight%/day)	1.2	1.26	5	0.01, -0.03	0.12, -0.14	0.16, -0.17
17	Decay Heat (sigma)	2	0		-4.36	-8.14	-5.04
18	Containment Leakage (Weight%/day)	1.2	6.0	500	-0.02	-2.31	-2.86
19	Passive Heat Sinks	Present	Absent	-	1.31	2.12, -0.15	1.52, -0.03
20	Heat Transfer Coefficient for Containment Heat Sinks	Empirical	Heat & Mass Transfer Analogy	-	0.31, -0.01	0.08, -3.34	0.13, -3.65

NPSHA and NPSHR for RHR Pumps from LB LOCA Statistical, Conservative and Realistic Analyses



NPSHA for RHR Pumps from LB LOCA Conservative and Monte Carlo Analysis



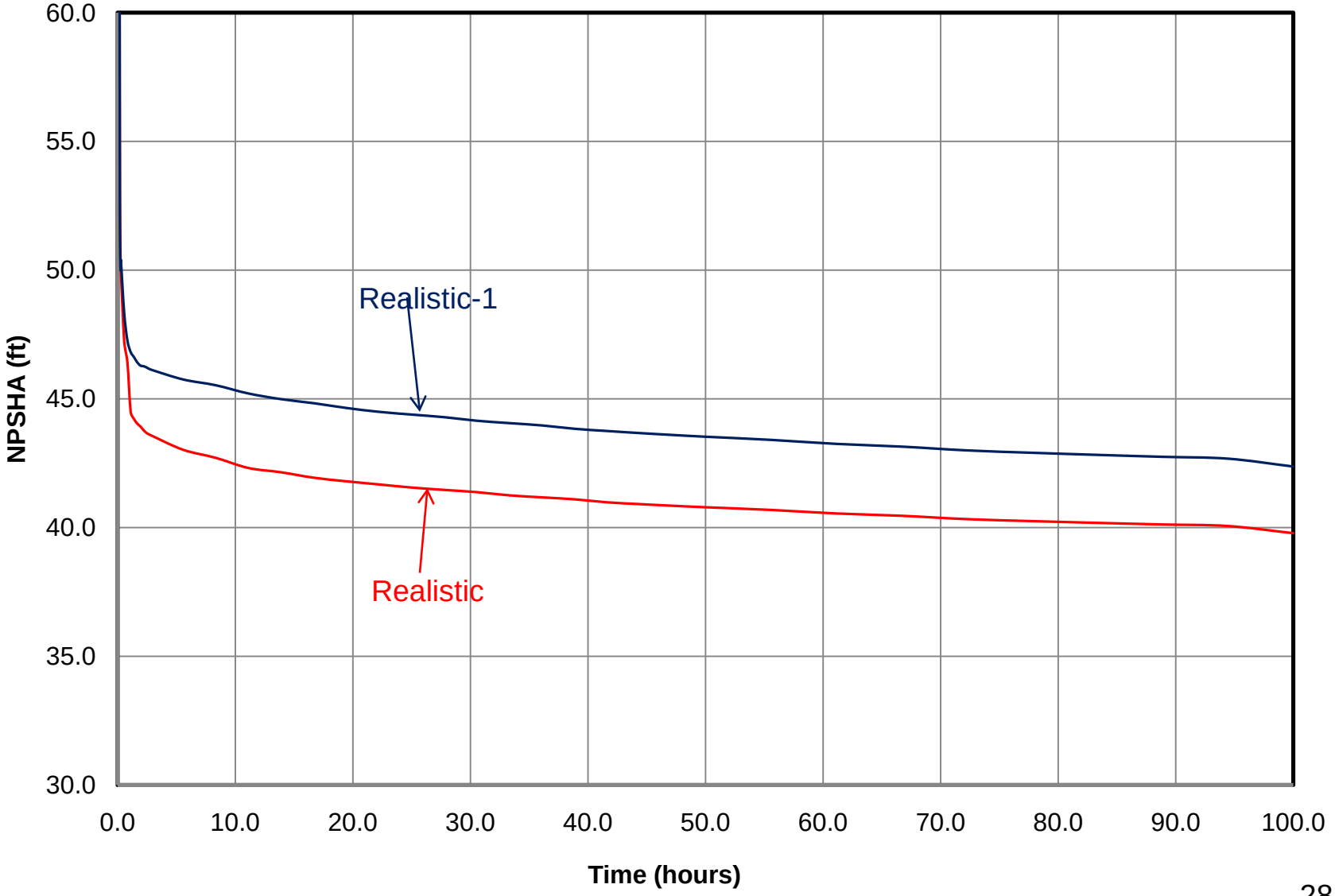
Comparison of Conservative, Realistic, and Statistical Inputs

Parameter	Conservative	Realistic	Statistical	Comment
Power (%)	102	100	100 to 102	
Decay heat	ANS 5.1-1979 +2σ	ANS 5.1-1979 +0σ	ANS 5.1-1979 +0σ to +2σ	Linear interpolation between 0 & 2σ for a randomly selected σ
DW Pressure (psia)	14.26 (atm press)	15.76	14.26 to 15.51	15.51 is based on plant data, 15.76 is based on ΔP DW to WW = 1.5 psi
WW Pressure (psia)	14.26 (atm press)	14.26 (atm press)	14.26 to 15.51	15.51 is based on plant data
DW Temp (deg F)	135 (max DW Temp)	135 (max DW Temp)	72.5 to 122.5	
WW Temp (deg F)	90	90	65 to 95	
Supp Pool Level	Minimum TS Limit	Minimum TS Limit	Minimum TS to Maximum TS Limit	
Service Water Temp (deg F)	90	90	32 to 90	
RHR Hx K-Value (Btu/s ^{o} F)	147	158	142.5 to 177.5	
DW Relative Humidity (%)	100	40	100	> 40 for may increase unidentified leakage
WW Relative Humidity (%)	100	100	100	
Mass & Energy	Based on reactor at 1040 psia and 549.5 ^{o}F	Based on reactor at 1040 psia and 549.5 ^{o}F	Based on reactor at 1040 psia and 549.5 ^{o}F	LB LOCA during reactor normal operation
Containment leakage	1.2 Weight% /day (La)	1.2 Weight% /day (La)	1.2 Weight% /day (La)	Tech Spec limit
Strainer and piping head loss	Based on maximum flow and pipe routing with maximum head loss	Based on maximum flow and pipe routing with maximum head loss	Based on maximum flow and pipe routing with maximum head loss	

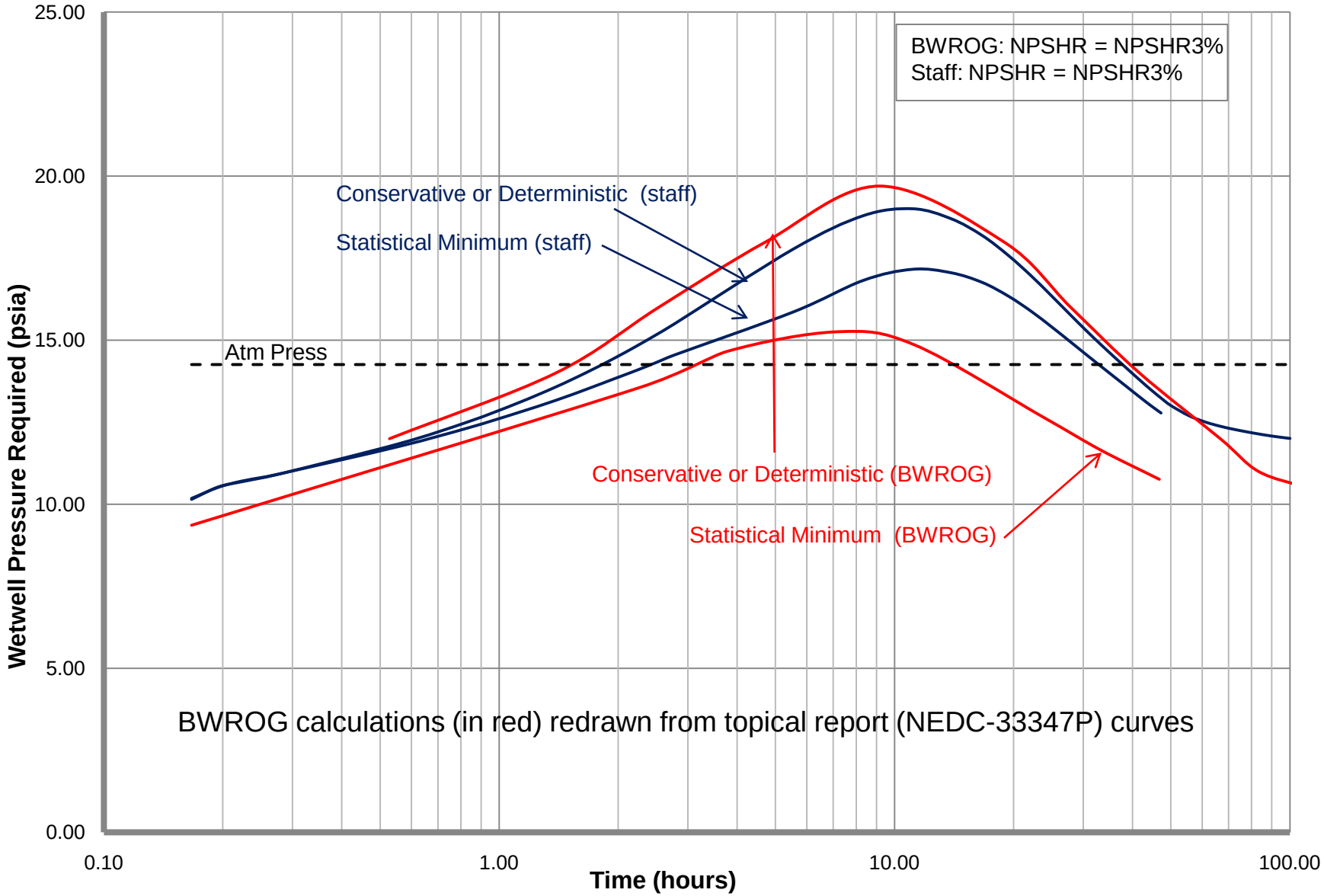
Comparison of Realistic and Realistic-1

Parameter	Realistic	Realistic-1	Comment
Power (%)	100	100	
Decay heat	ANS 5.1-1979 +0σ	ANS 5.1-1979 +0σ	
DW Pressure (psia)	15.76	15.76	15.76 is based on ΔP DW to WW = 1.5 psi
WW Pressure (psia)	14.26 (atm press)	14.26 (atm press)	
DW Temp (^{o}F)	135 (max DW Temp)	97.5	97.5 is the average of the range in statistical analysis
WW Temp (^{o}F)	90	80	80 is the average of the range in statistical analysis
Supp Pool Level	Minimum TS Limit	Average of Minimum and Maximum level	
Service Water Temperature (^{o}F)	90	90	
RHR Hx K-Value (Btu/s ^{o} F)	158	160	
DW Relative Humidity (%)	40	40	> 40 for may increase unidentified leakage
WW Relative Humidity (%)	100	100	
Mass & Energy	Based on reactor at 1040 psia and 549.5 ^{o}F	Based on reactor at 1040 psia and 549.5 ^{o}F	LB LOCA during reactor normal operation
Containment leakage	1.2 Weight% /day (La)	1.2 Weight% /day (La)	Tech spec limit
Strainer and piping head loss	Based on maximum flow and pipe routing with maximum head loss	Based on maximum flow and pipe routing with maximum head loss	

NPSHA for RHR Pumps from LB LOCA Realistic and Realistic-1 Analyses



CAP Required for RHR Pumps During DBA LOCA



Guidelines-1

- 1. Propose use of $NPSHR_{eff}$ defined as
  - $NPSHR_{eff}$ = (1.0 + uncertainty) $NPSHR_{3\%}$
- 2. ~~Either a conservative or 95/95 lower tolerance limit~~ A conservative analysis should be used in determining the available NPSH
- 3. A realistic calculation of available NPSH should be performed to compare with available NPSH determined from a conservative ~~or Monte Carlo 95/95~~ calculation
- 4. Pump operation in the maximum erosion zone should be limited to less than 100 hours.

Guidelines-2

- 5. Maximum flow rate chosen for NPSH analyses should be greater than flow rate used for core and containment cooling analyses
- 6. The mission time of the pump must consider any operation necessary to maintain stable core and containment cooling post-accident.
- 7. Containment isolation should not be lost due to an Appendix R Fire (associated circuit) or containment venting (required by procedures)
- 8. ~~Operator action to control containment pressure is acceptable if justified by human factors considerations and included in appropriate procedures~~
- 9. Operation for a limited time with NPSHA<NPSHR is acceptable, if justified by testing

Guidelines-3

- 10. To protect the mechanical seal faces from excess entrained air (released during operation at the 3% NPSHR condition), dual mechanical seals with an external cold water flush system (or equivalent) should be provided
- 11. Licensees should justify that use of containment accident pressure is necessary because the design cannot be practicably altered. (Not necessary for BWRs with Mark I containments.)

Summary

- Containment integrity is assumed.
  - based on the rigorous requirements for containment integrity (regulations, technical specifications, procedures) and based on risk insights
- Staff guidelines focus on pump performance to ensure that ECCS and containment heat removal pumps are capable of performing their safety function(s)
  - NPSHR uncertainty quantified
  - Staff guidelines used to indicate margin
    - Between realistic and conservative NPSHA
    - Between NPSHA and NPSHR or $NPSHR_{3\%}$
- Other effects (e.g., maximum cavitation erosion and air) included in guidance
- Some guidance details have not been finalized. Staff expects more discussion with owners' groups
  - 100 hour limit on time in maximum erosion zone
  - Values of uncertainty in required NPSH
  - Value of NPSHR for use in NPSH margin ratio for maximum, erosion zone calculations

Risk Evaluation of Using Containment Pressure to Prevent ECCS Pump Cavitation

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May 2010

Outline

- Overview
- Technical Approach
  - Preliminaries
  - Leak Probabilities
    - Pre-initiator
    - Upon-initiator
    - Post-initiator
  - Operator Actions
  - Seismic Risk Evaluation
- Risk Insights

Technical Approach

- Purpose: To estimate the increase in core-damage frequency (CDF) that results from relying upon containment accident pressure (CAP) to prevent ECCS pump cavitation.
- General approach:
  - Internal Events
    - Modify Standardized Plant Analysis of Risk (SPAR) models
    - Addresses all internal initiating events (LOCAs and transients)
    - Browns Ferry and Monticello
    - Assumes CAP credit is needed whenever the CS or RHR pumps are taking suction on the suppression pool
  - Seismic Events
    - Simplified method being used in GI-199
    - Added seismically induced loss of containment integrity (HCLPF = 0.3g)
    - Assumes loss of containment integrity directly causes core damage for all accident sequences
  - Internal Fires
    - Not assessed due to lack of detailed cable routing information to assess the impact on fire on containment integrity
    - Deterministic guidance developed

The Definition of “Loss of Containment Integrity”

- The event “loss of containment integrity” means that the containment is leaking enough to prevent adequate NPSH.
- The leak size needed to prevent adequate NPSH is plant-specific, and should be determined through containment thermal-hydraulic analyses (e.g., GOTHIC, MELCOR).
- Leak sizes used in previous license-performed risk evaluations:
  - Vermont Yankee EPU:
    - 27 La (calculated using 10 CFR 50 Appendix K requirements)
    - 60 La (using more realistic assumptions)
  - Browns Ferry EPU: 35 La (engineering judgment)
- Assumed 20 La in this analysis.

Three Timeframes Considered

- Pre-initiator: Containment may be leaking before an initiating event occurs.
- Upon-initiator: Containment may fail to isolate when an initiating event occurs.
- Post-initiator: Containment may start to leak after the initiating event occurs.

Pre-Initiator Leak Probability

- Previous risk evaluations used a pre-initiator (pre-existing leak) probability that only depended on the size of containment leakage.
  - Vermont Yankee EPU: 2.47×10^{-4} (from EPRI TR 1009325)
  - Browns Ferry: 9.86×10^{-4} (from EPRI TR 1009325)
- However, the probability of a pre-initiator containment leak should also depend on how the containment integrity is tested:
  - How often the test is performed, for example:
    - Integrated leak rate tests (ILRTs)
    - Oxygen concentration monitoring in BWR Mark I containments
  - Test efficiency (how good is the test at detecting leaks of the size needed to preclude adequate NPSH)
- The staff developed a semi-Markov model to represent the impact of containment integrity testing on the pre-initiator leak probability.

Technical Specification

CONDITION	REQUIRED ACTION	COMPLETION TIME
Containment leakage at or above [20] La	Reduce containment leakage below [20] La	T_{ST} [24h]
Required Action and Associated Completion Time not met	Shutdown plant	T_{SD} [8h]

SURVEILLANCE	FREQUENCY
Verify containment leakage less than [20] La	T_I [7 days]

Patterned after BWR/4 Standard Technical Specification 3.6.3.2, “Primary Containment Oxygen Concentration”

Parameters that Determine the Pre-Initiator Leak Probability

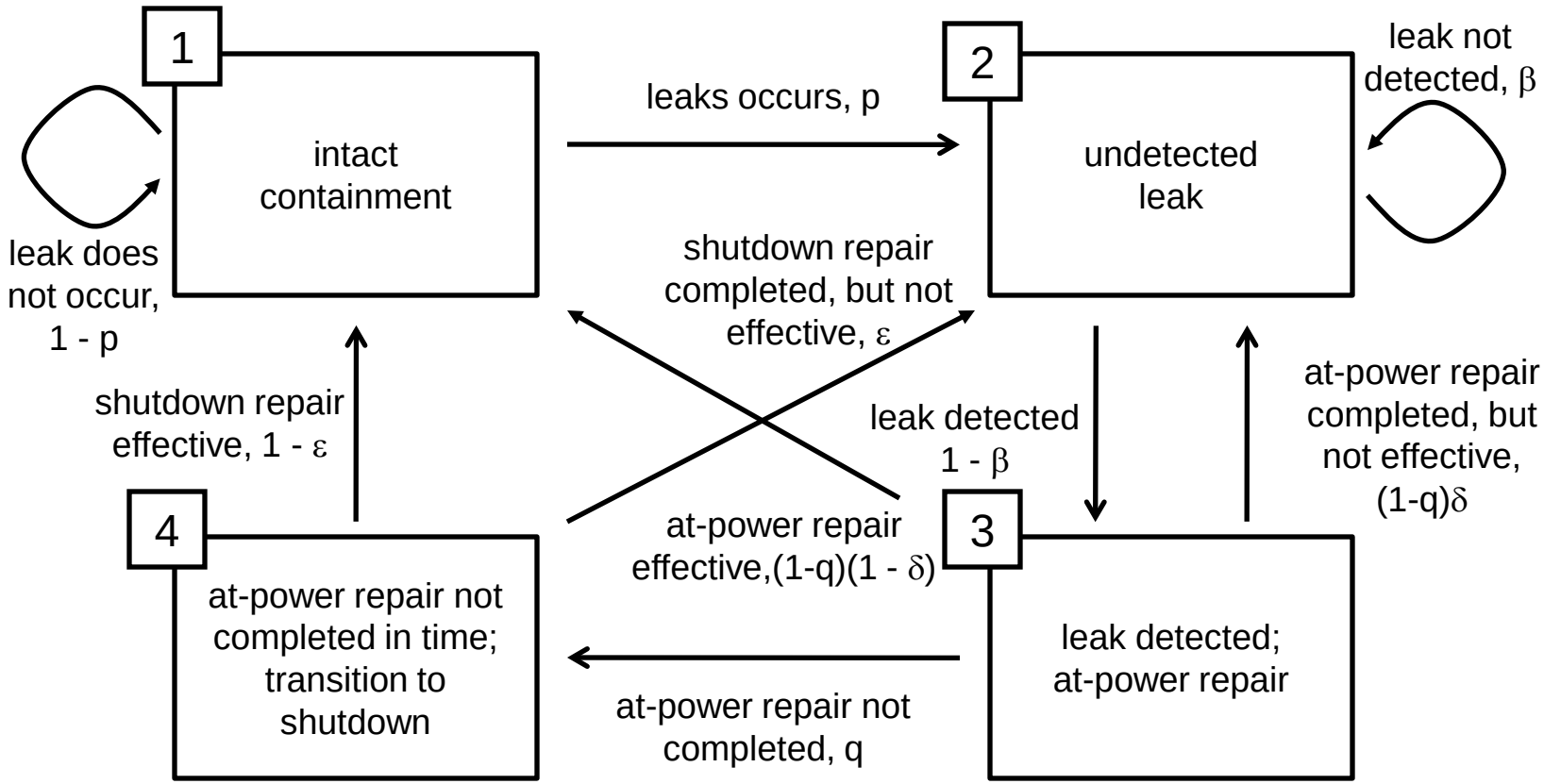
- Containment leakage failure rate, λ
- Mean time to repair, τ
- Surveillance test interval, T_I
- TS-allowed repair duration while at-power, T_{ST}
- TS-mandated shutdown time, T_{SD}
- Test sensitivity (probability of a Type II error), β, δ, and ε

Test Confusion Matrix
(Statistical Hypothesis Testing)

		Actual Containment Condition	
		Leak (true)	Intact (false)
Test Result	Leak (positive)	**true positive** $Pr\{T_P \mid C_T\} = 1 - \beta$ = sensitivity	**false positive** (Type I error) $Pr\{T_P \mid C_F\} = \alpha$
	Intact (negative)	**false negative** (Type II error) $Pr\{T_N \mid C_T\} = \beta$	**true negative** $Pr\{T_N \mid C_F\} = 1 - \alpha$ = specificity

- False negatives are important to plant safety.
- False positives are important to plant operations.

Semi-Markov Model



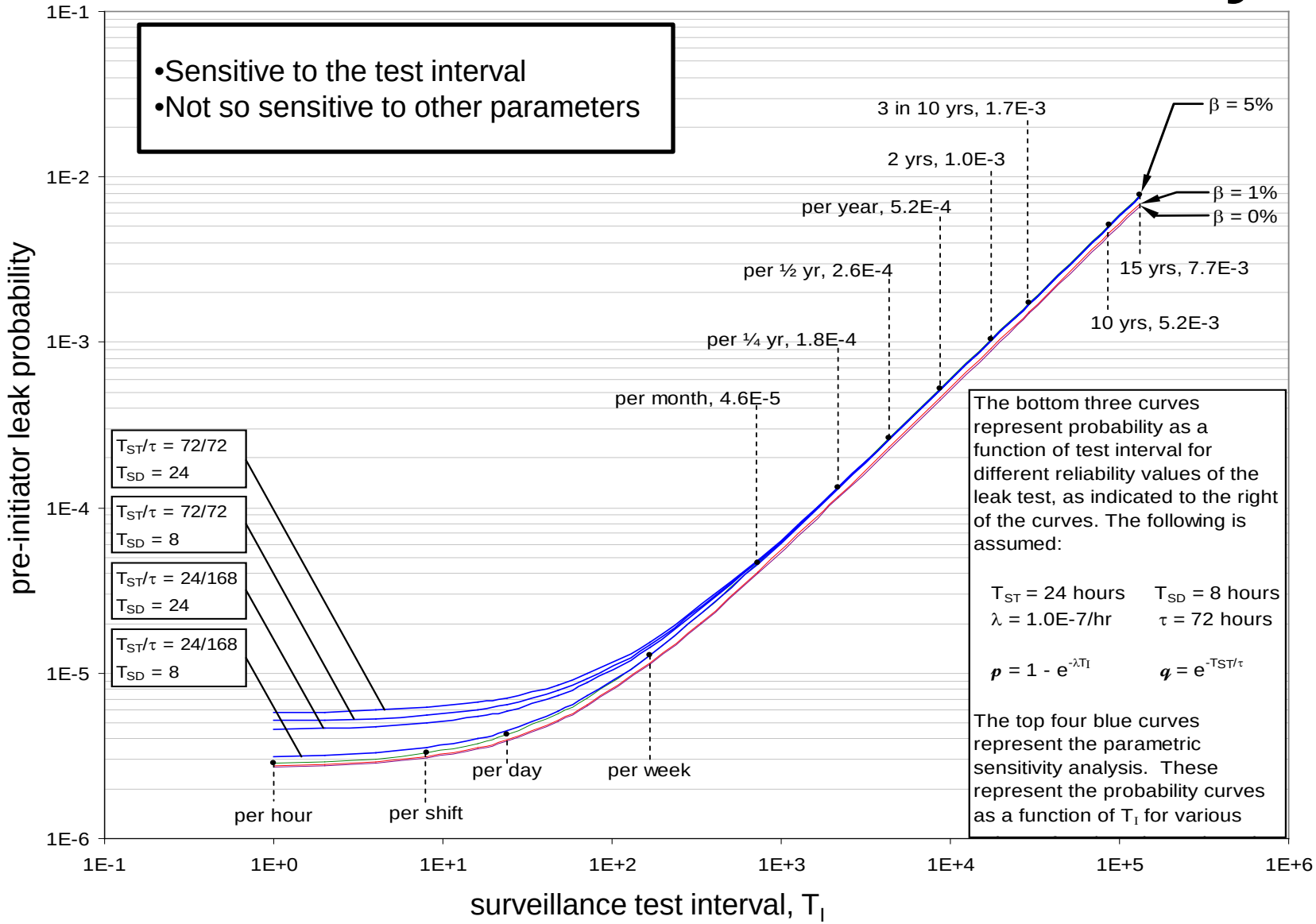
Pr{pre-initiator leak} = long-run fraction of time that the system is in States 2, 3 and 4

Containment Leakage Reliability Parameters

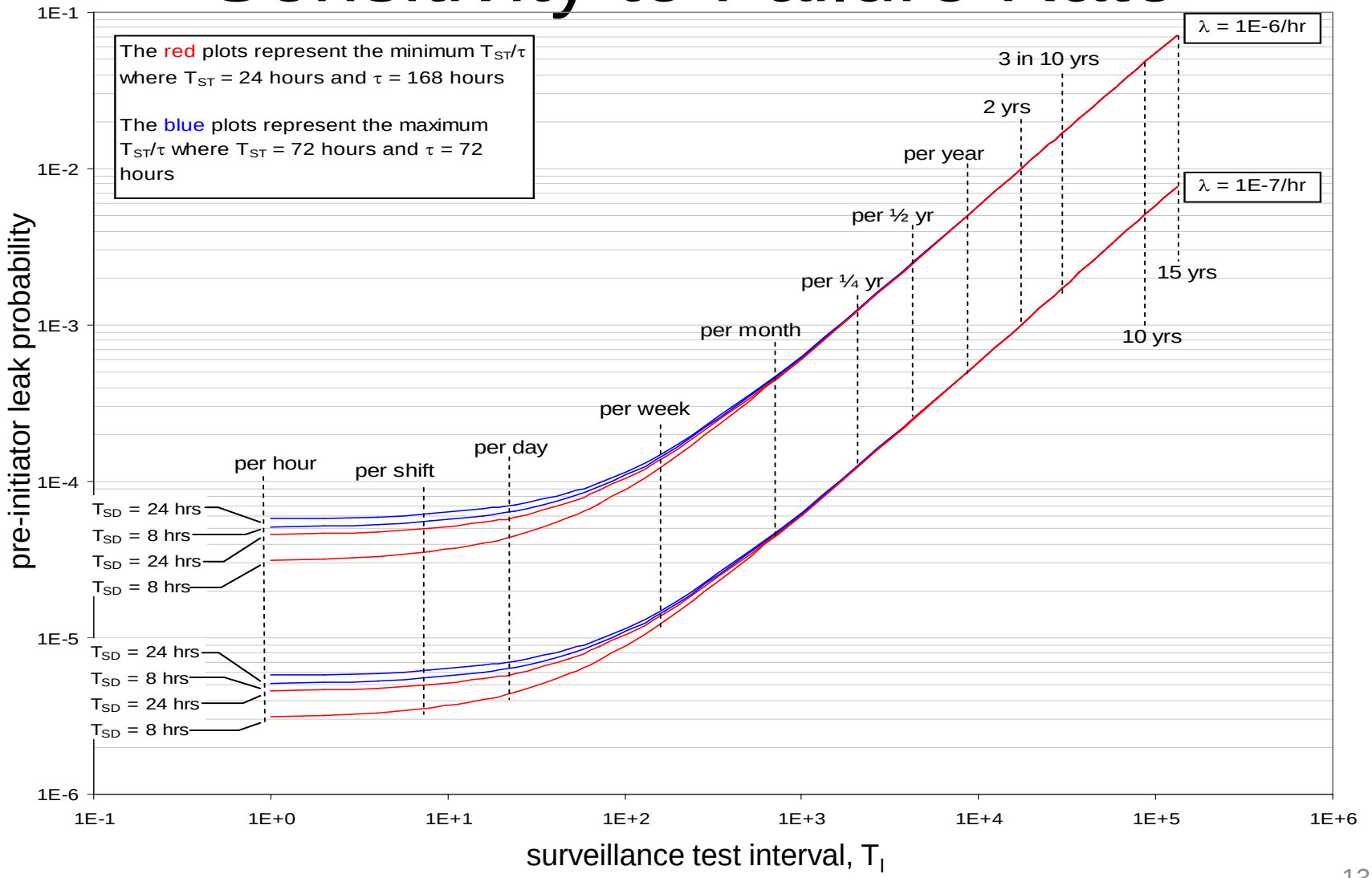
$$\lambda(20\,L_a) = \frac{\Pr\{\text{leak} \geq 20\,L_a \mid \text{leak}\}}{\Pr\{\text{leak} \geq 1\,L_a \mid \text{leak}\}} \times \lambda(1\,L_a) = 1 \times 10^{-7}\,/\text{h}$$

- λ(1 La) = 1.1E-2/RY = 1.3E-6/hour from NUREG-0933, “Resolution of Generic Issues,” Section 1 – TMI Action Plan Items, Item II.E.4, “Containment Integrity”
- Leak size distributions:
  - Pr{leak ≥ 20 L_a | leak exists}= 2E-3
  - Pr{leak ≥ 1 L_a | leak exists}= 3E-2
  - Source: Table D-1 of EPRI, “Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals,” Report No. 1009325, Rev. 1, October 2005 (ADAMS Accession No. ML053550424).
- Mean time to repair, τ = 72 hours (NUREG-0933)

Pre-Initiator Leak Probability



Pre-Initiator Leak Probability Sensitivity to Failure Rate



Containment Isolation

- During routine power operations, there are no pathways between the containment and the atmosphere.
  - Pathways exist during inerting (24h after plant startup) and deinerting (24 h prior to plant shutdown.)
- As a result, only necessary to model failure of the containment isolation system to close on demand when these pathways exist:

$$\Pr\{\text{upon-initiator leak}\} = \frac{2 \times 24\text{h}}{24\text{m}} \times \frac{1\text{m}}{730\text{h}} \times 10^{-3} = 3 \times 10^{-6}$$

- Different approach than previous licensee-performed risk evaluations, which assumed that containment isolation is always required.

Containment Isolation (Con't.)

- Following the occurrence of a LOCA (LLOCA, MLOCA, or SLOCA), failure to close the MSIVs introduces a pathway between the containment and the atmosphere.
- The probability that all MSIVS fail to close on demand is about 10^{-4} (including independent and common-cause failures).
- This pathway is a very small contribution to the change in core-damage frequency since LOCAs frequencies are relatively small.

Post-Initiator Leak Probability

- Used 72h mission time to account for the period that containment accident pressure is needed to provide adequate NPSH.

$$\Pr\{\text{post}-\text{initiator leak}\} = 1 - e^{-\lambda T} \approx \lambda T$$

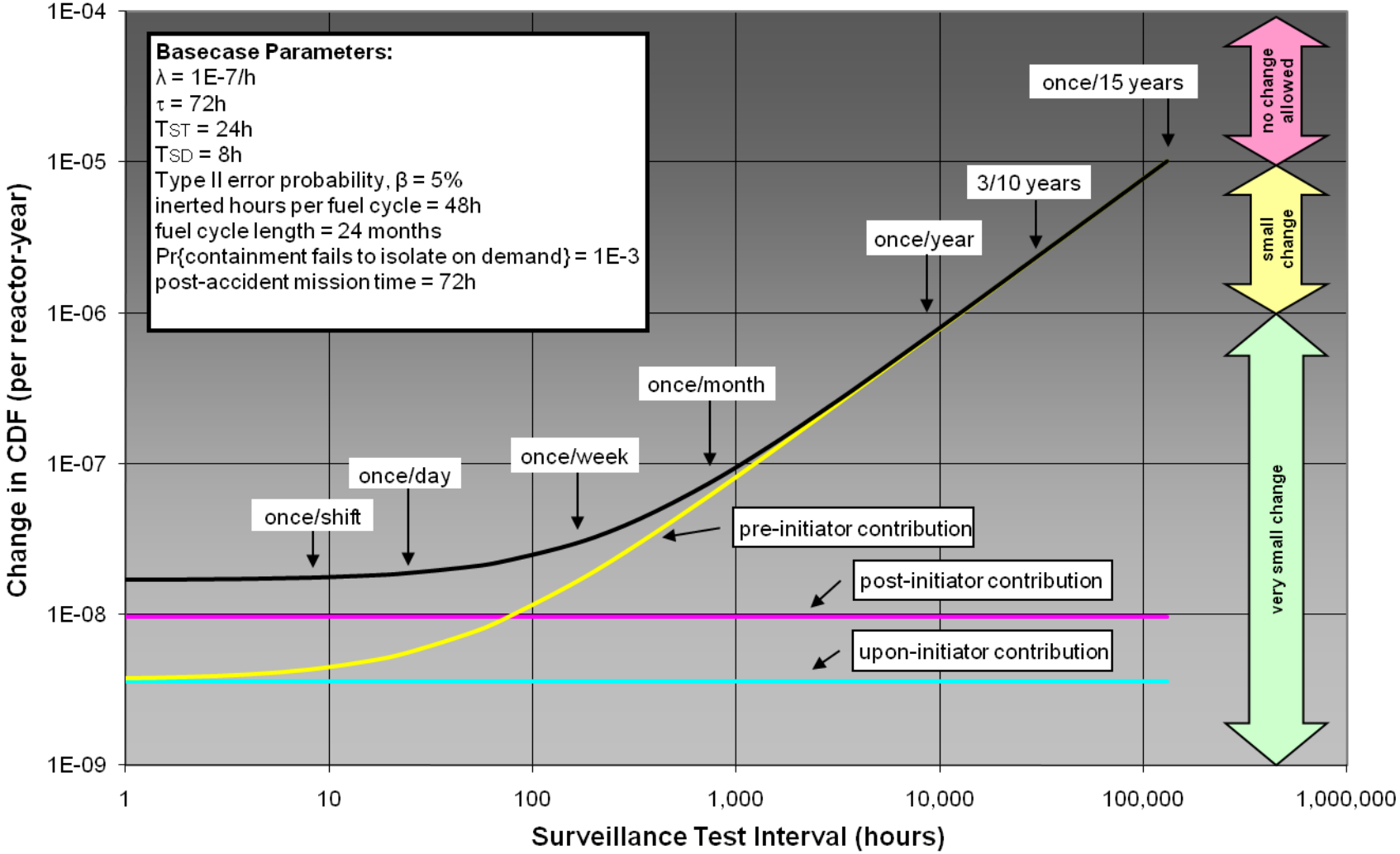
$$= (1\times 10^{-7}/\text{h})(72\text{h}) = 7\times 10^{-6}$$

The mission time is the only parameter in the model that pertains to the duration of the CAP credit

Operator Actions

- Current T/H analysis indicates that no operator actions are needed to implement the CAP credit (e.g., tripping the drywell coolers).
- Readily included in a risk evaluation, if needed:
  - Existing HRA methods adequate to estimate the human error probability (HEP).
  - Need to consider potential dependencies with other operator actions.
- May be a significant contribution to risk:
  - Human error probability >> pre-initiator, upon-initiator, and post-initiator leak probabilities.

Browns Ferry CAP Credit - Internal Events



Seismic Risk Evaluation

- Simplified method being used in GI-199:
  - Integration of the mean seismic hazard curve with the mean plant-level fragility curve (not a seismic PRA).
    - Hard rock seismic hazard: USGS 2008
    - Site-specific soil amplification factors: EPRI/SOG 1989
    - Plant-level fragility: IPEEE information or license renewal SAMA analysis (SPRA or SMA)
  - Added seismically induced loss of containment integrity:
    - HCLPF = 0.3g and $\beta_C = 0.4$
    - Assumes loss of containment integrity directly causes core damage for all accident sequences
- Evaluated all BWRs with Mark I containments.

Increase in Seismic Core-Damage Frequency Due to the CAP Credit

Plant Name	Docket Number	SCDF no CAP credit needed	SCDF CAP credit needed	ΔSCDF due to CAP credit
Browns Ferry 1	05000259	3.7E-06	5.7E-06	2.0E-06
Browns Ferry 2	05000260	5.4E-06	7.0E-06	1.6E-06
Browns Ferry 3	05000296	5.4E-06	7.0E-06	1.6E-06
Brunswick 1	05000325	1.5E-05	2.5E-05	9.8E-06
Brunswick 2	05000324	1.5E-05	2.5E-05	9.8E-06
Cooper	05000298	7.0E-06	1.1E-05	3.9E-06
Dresden 2	05000237	1.2E-05	1.5E-05	3.1E-06
Dresden 3	05000249	1.2E-05	1.5E-05	3.1E-06
Duane Arnold	05000331	3.8E-06	6.0E-06	2.2E-06
Fermi 2	05000341	4.2E-06	6.0E-06	1.8E-06
FitzPatrick	05000333	6.1E-06	6.9E-06	8.3E-07
Hatch 1	05000321	2.5E-06	3.8E-06	1.3E-06
Hatch 2	05000366	2.5E-06	3.8E-06	1.3E-06
Hope Creek 1	05000354	2.8E-06	8.9E-06	6.1E-06
Monticello	05000263	1.9E-05	2.0E-05	1.9E-07
Nine Mile Point 1	05000220	4.2E-06	5.4E-06	1.2E-06
Oyster Creek	05000219	1.4E-05	1.6E-05	2.4E-06
Peach Bottom 2	05000277	2.4E-05	2.7E-05	2.3E-06
Peach Bottom 3	05000278	2.4E-05	2.7E-05	2.3E-06
Pilgrim 1	05000293	6.9E-05	7.5E-05	5.9E-06
Quad Cities 1	05000254	2.7E-05	2.7E-05	5.3E-08
Quad Cities 2	05000265	2.7E-05	2.7E-05	5.3E-08
Vermont Yankee	05000271	8.1E-06	9.9E-06	1.8E-06

All plants meet the numerical risk acceptance guidelines in RG 1.174

Risk Insights

- There is only one minimal cut set where the loss of containment integrity leads directly to core damage (large LOCA).
- The increase in internal event CDF is very small ($<10^{-6}$/y, as defined in RG 1.174) when testing is conducted at least once/year (assuming a leak failure rate of 10^{-7}/h).
- The increase in seismic CDF meets the numerical risk acceptance guidelines in RG 1.174 for all plants.
- Contributions to containment leakage probability:
  - Pre-initiator (basecase): 55.9%
  - Post-initiator: 32.1%
  - Upon-initiator: 12.0%

Risk Insights (Con't)

- Importance measures for loss of containment integrity:
  - Fussell-Vesely (FV): 0.017
  - Risk achievement worth (RAW): 750
  - The loss of containment integrity is a "significant basic event," as defined in the ASME/ANS PRA Standard (FV > 0.005 and/or RAW > 2), over a wide range of model parameters.
- Sensitivity studies indicate that the pre-initiator contribution to the containment leakage probability mainly depends on:
  - The containment leakage failure rate
  - The surveillance test interval

Breakdown of Internal Event CDF Contributions

Initiating Event	No CAP Credit		With CAP Credit		Change in CDF	
	CDF	Percent	CDF	Percent	CDF	Percent
General Transient	3.4E-08	1.9%	3.4E-08	1.9%	2.9E-10	0.9%
Small LOCA	1.1E-09	0.1%	1.1E-09	0.1%	2.5E-12	0.2%
Steam Line Break Outside Containment	6.4E-08	3.7%	6.4E-08	3.6%	1.7E-11	0.0%
Medium LOCA	6.5E-08	3.8%	6.8E-08	3.8%	2.3E-09	3.6%
Loss of Service Water	9.1E-09	0.5%	1.8E-08	1.0%	9.2E-09	101.2%
Loss of Plant Control Air	7.1E-09	0.4%	1.1E-08	0.6%	3.5E-09	49.6%
Loss of Offsite Power	1.4E-06	81.4%	1.4E-06	80.2%	3.5E-09	0.2%
Loss of Main Feedwater	2.9E-08	1.7%	2.9E-08	1.7%	5.0E-11	0.2%
Large LOCA	7.8E-09	0.4%	8.0E-09	0.5%	2.3E-10	2.9%
Inadvertent Open Relief Valve	1.4E-08	0.8%	1.6E-08	0.9%	1.6E-09	11.8%
Loss of Condenser Heat Sink	9.3E-08	5.3%	1.0E-07	5.8%	9.4E-09	10.1%
Total	1.7E-06		1.8E-06		3.0E-08	1.7%

No significant change in the plant risk profile

Conclusions

- Major risk insights:
  - The loss of containment integrity is a “significant basic event,” as defined in the ASME/ANS PRA Standard, over a wide range of model parameters.
  - The increase in internal event CDF can be made very small ($<10^{-6}$/y, as defined in RG 1.174) with adequate testing of containment integrity.
  - The increase in seismic CDF meets the numerical risk acceptance guidelines in RG 1.174 for all plants.
  - Operator actions, if needed to implement a CAP credit, may be significant risk contributors.
- The increases in core-damage frequencies due to CAP credits are small and consistent with the intent of the Commission’s Safety Goal Policy Statement.

Backup Viewgraphs

Semi-Markov Processes

- System is in one of a number of discrete states.
- The probability that the system transitions to another state depends only on its current state:
  - Transitions are independent of the system's past history.
  - This characteristic is called the "Markovian property."
- The time that the system waits in a given state is random with an arbitrary probability distribution:
  - In general, each state has its own waiting time distribution.
  - Special cases:
    - Markov chain – all waiting times are fixed and equal.
    - Continuous time Markov process – all waiting times are exponentially distributed.

Average State Durations

M_1	$T_l\left(1-\frac{p}{2}\right)$
M_2	$\frac{T_l}{2}(1+\beta)$
M_3	$qT_{ST}+(1-q)\frac{\tau-(\tau+T_{ST})\exp(-T_{ST}/\tau)}{1-\exp(-T_{ST}/\tau)}$
M_4	T_{SD}

Semi-Markov Model Solution

$p = 1 - \exp(-\lambda T_I)$ Probability of leak over interval T_I

$q = \exp(-T_{ST} / \tau)$ Probability that leak is not repaired in interval T_{ST}

$$\left.\begin{aligned}
\pi_1 &= \frac{1-\beta}{pD}\left[1-(1-q)\delta - \varepsilon q\right] \\
\pi_2 &= 1/D \\
\pi_3 &= \frac{1-\beta}{D} \\
\pi_4 &= \frac{(1-\beta)q}{D} \\
D &= \frac{1-\beta}{p}\left[1-(1-p)\delta - \varepsilon q\right] + 1 + (1-\beta) + (1-\beta)q
\end{aligned}\right\}$$

Steady-state solutions to the embedded Markov chain

$$\boxed{P_{234} = \frac{M_2 + (1-\beta)M_3 + (1-\beta)qM_4}{\frac{1-\beta}{p}\left[1-(1-p)\delta - \varepsilon q\right]M_1 + M_2 + (1-\beta)M_3 + (1-\beta)qM_4}}$$

Using Containment Pressure in Determining Available NPSH

Applications of Risk Insights

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May 2010

Summary of Risk Insights

- There is only one minimal cut set where the loss of containment integrity leads directly to core damage (large LOCA).
- The increase in internal event CDF can be made very small (<10-6/y, as defined in RG 1.174) with adequate testing of containment integrity.
- The increase in seismic CDF, based on a simplified and conservative analysis, meets the numerical risk acceptance guidelines in RG 1.174 for all plants.
- Determanistic guidance is needed to address fire risk (no numerical risk results exist).

Regulatory Roadmap

SECY-99-246
(10/12/1999)

SRM on SECY-99-246
(1/5/2000)

RIS 00-07
(3/28/2000)

COMSECY-00-0038
(11/13/2000)

SRM on COMSECY-00-0038
(11/22/2000)

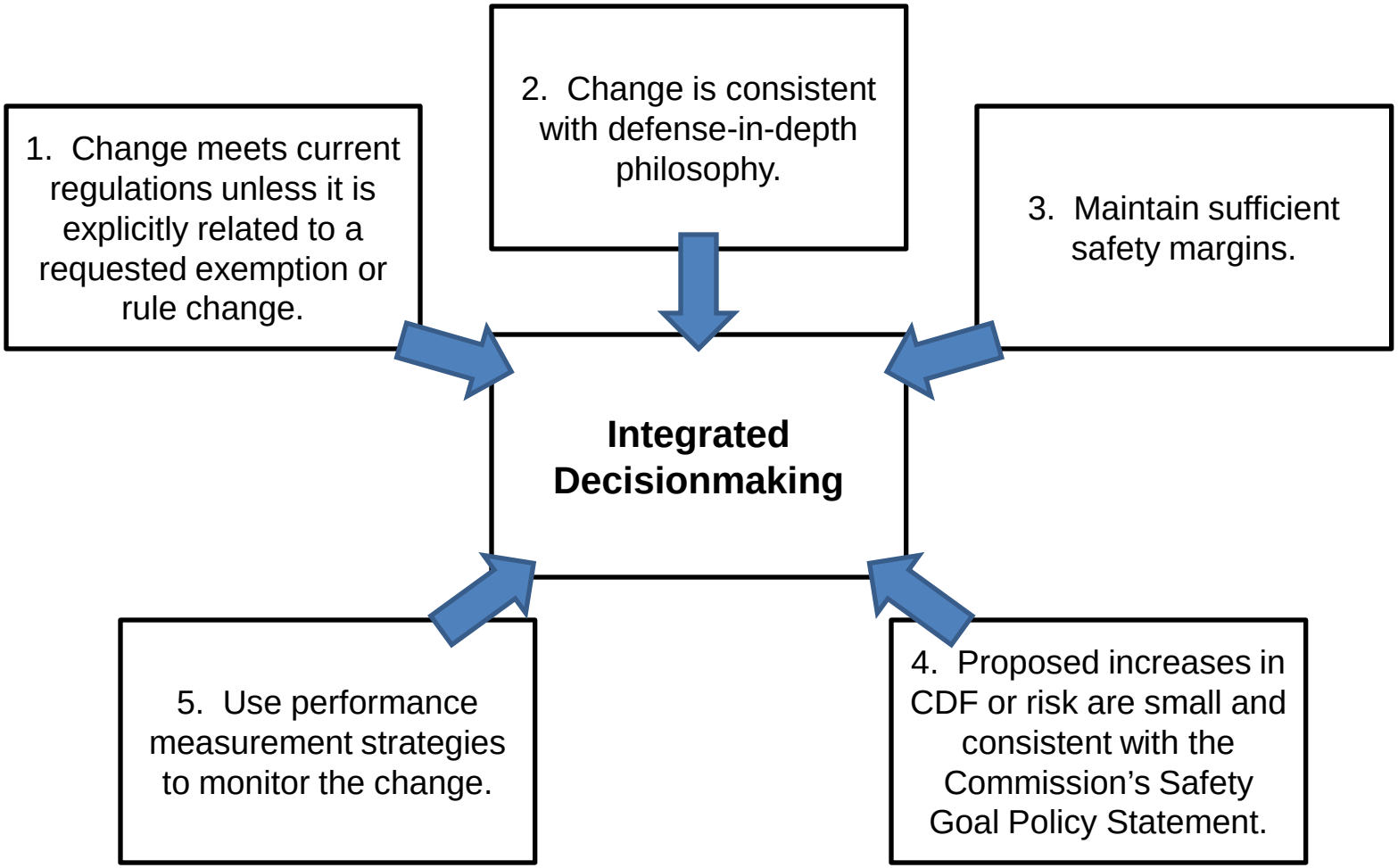
RIS 01-02
(1/18/2001)

RG 1.174

SRP Section 19.2, App. D
Use of Risk Information in Review of Non-Risk-Informed License Amendment Requests

The numerical guidance on CDF and LERF and the safety principles in RG 1.174 are intended to provide a basis for finding that there is reasonable assurance of adequate protection.

Five Key Principles of Risk-Informed Decisionmaking



Defense-in-Depth

- The Commission's White Paper on Risk-Informed and Performance-Based Regulation (SRM to SECY-98-144, March 1, 1999):
  - Defense-in-depth is an element of the NRC's Safety Philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally-caused event occurs at a nuclear facility.
  - The defense-in-depth philosophy ensures that safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of a nuclear facility.
  - Decisions on the adequacy of or the necessity for elements of defense should reflect risk insights gained through identification of the individual performance of each defense system in relation to overall performance.

Advisory Committee Views

- ACRS Letter, “The Role of Defense in Depth in and Risk-Informed Regulatory System,” May 19, 1999:
  - Our motivation for this report has arisen because of instances in which seemingly arbitrary appeals to defense in depth have been used to avoid making changes in regulations or regulatory practices that seemed appropriate in the light of results of quantitative risk analyses.
  - Unless defense-in-depth measures are justified in terms of necessity and sufficiency, the full benefits of risk-informed regulation cannot be realized.
- Joint ANCW and ACRS Letter, “Use of Defense in Depth in Risk-Informing NMSS Activities,” May 25, 2000:
  - Defense in depth is a strategy to ensure public safety given unquantified uncertainty in PRAs.
  - The nature and extent of compensatory measures should depend on:
    - The degree of uncertainty, and
    - The degree of risk posed by the activity.
  - How good each compensatory measure should be is a value judgment and, thus, a matter of policy.

RG 1.174, Section 2.2.1.1 Guidance

- Consistency with the defense-in-depth philosophy is maintained if:
  - A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.
  - Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.
  - System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).
  - Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed.
  - Independence of barriers is not degraded. CAP credit concern
  - Defenses against human errors are preserved.
  - The intent of the General Design Criteria in Appendix A to 10 CFR Part 50 is maintained.

SRP Section 19.2
Subsection III.2.1.1.1 Guidance

- The change does not result in a significant increase in the existing challenges to the integrity of barriers.
- The change does not significantly change the failure probability of any individual barrier.
- The proposal does not introduce new or additional failure dependencies among barriers that significantly increase the likelihood of failure compared to the existing conditions.
- The overall redundancy and diversity among the barriers is sufficient to ensure compatibility with the risk acceptance guidelines.

Defense in Depth and the Use of Containment Pressure in Determining Available NPSH

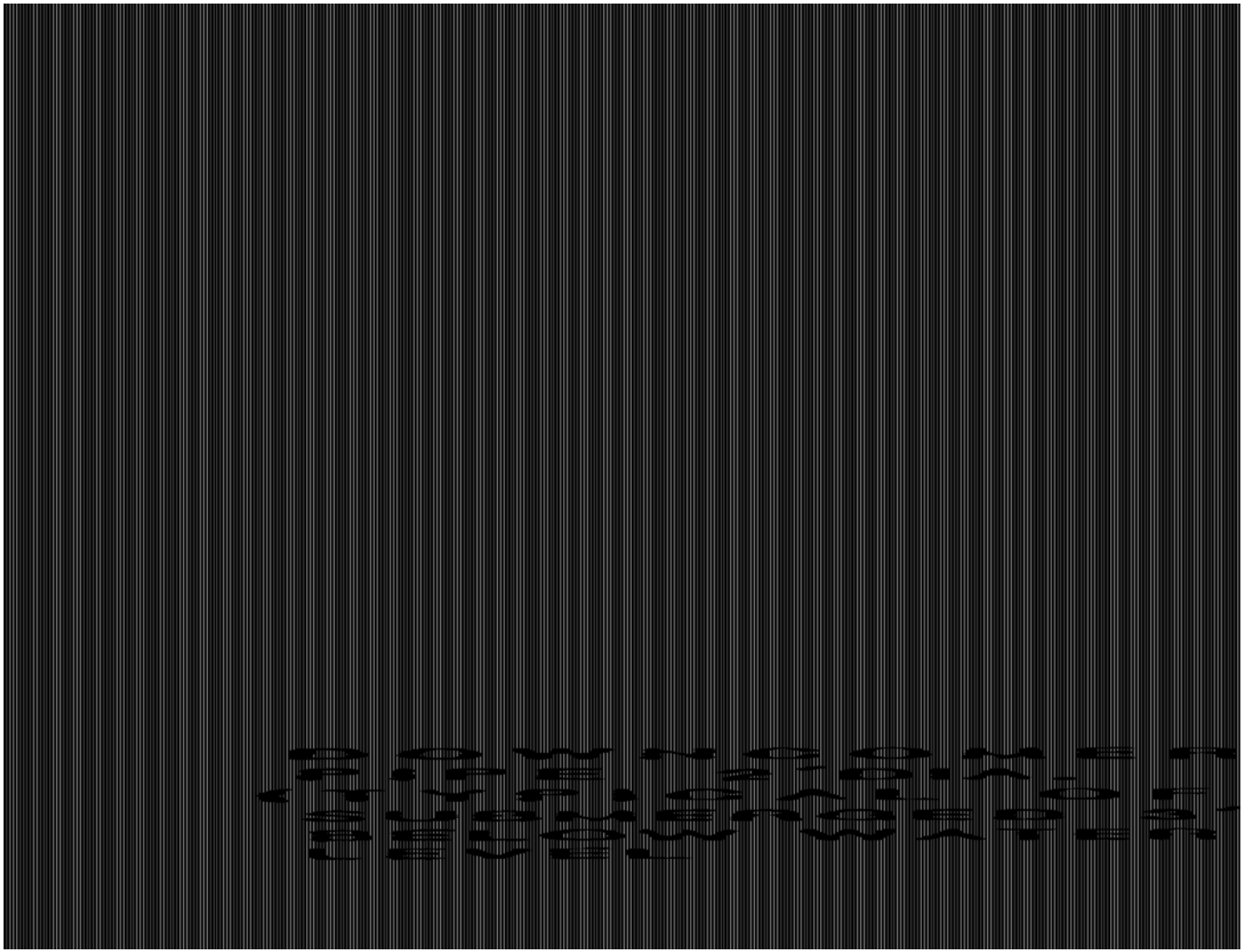
- No changes to any programmatic element (e.g., inspections) that provides defense-in-depth.
- Proposed staff guidance:
  - Considers T/H uncertainties in NPSHA and NPSHR.
  - Specifies adequate containment integrity monitoring.
- The size of a containment leak that causes loss of NPSH is smaller than the size associated with a large early release:
  - 20-40 La compared to > 100 La
  - Therefore: ΔCDF > ΔLERF

Risk Insights and “Practicably Altered”

- Staff guidelines state that licensees should justify that use of containment accident pressure is necessary because the design cannot be practicably altered.
- Numerical risk results may make this guideline moot for BWRs with Mark I containments:
  - The increase in internal event CDF can be made very small with adequate containment integrity surveillance.
  - Seismic risk may be a major contributor, depending on:
    - Site-specific seismic hazard curves
    - Containment seismic fragility
  - Based on the regulatory (value-impact) analysis guidelines provided in NUREG/BR-0184:
    - Assume $\Delta CDF = 10^{-6}/y$ due to the CAP credit.
    - $74K justified to avoid using the CAP credit.

Backup slides to BWROG May 6 presentation to ACRS on Containment Accident Pressure

BWR Mark I Containment



Pre-Existing Leak in Containment

Typically, the nitrogen makeup system is not in service. When the nitrogen makeup system is in service the maximum nitrogen makeup flow rate is

2 SCFM

Gross containment leakage that would affect NPSH margin is ~40 La or 40 times 458 SCFH = 305 SCFM

Therefore, the low Drywell pressure alarm could not be cleared. Containment leakage would be determined to exceed 1.0 La and TS actions entered.

Pre-Existing Leak in Containment

Technical Specifications requires that primary containment oxygen concentration shall be < 4.0 volume percent during power operations.

Primary containment oxygen concentration is required to be verified within limits every 7 days.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Primary containment oxygen concentration not within limit.	A.1 Restore oxygen concentration to within limit.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 15% RTP.	8 hours