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

July 13, 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 556th FULL COMMITTEE MEETING HELD ON OCTOBER 2-4, 2008, IN ROCKVILLE, MARYLAND

The minutes of the subject meeting were certified on November 19, 2008 as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment:
As stated

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

November 19, 2008

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

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

SUBJECT: MINUTES OF THE 555th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), OCTOBER 2-4, 2008

I certify that based on my review of the minutes from the 552nd ACRS Full Committee meeting, and to the best of my knowledge and belief, I have observed no substantive errors or omissions in the record of this proceeding subject to the comments noted below.

OFFICE	ACRS	ACRS:RSB
NAME	SMeador	CSantos/sam
DATE	11/ 19 /08	11/ 19 /08

OFFICIAL RECORD COPY

CERTIFIED Date Certified: 11/19/2008

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IV. List of Documents Provided to the Committee

During its 556th meeting, October 2-3, 2008, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports, letters, and memoranda:

REPORTS

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

- Report on the Safety Aspects of the License Renewal Application for the Shearon Harris Nuclear Power Plant, Unit 1, dated October 16, 2008

- Status of Resolution of Generic Safety Issue-191, "Assessment of Debris Accumulation on PWR Sump Performance," dated October 22, 2008

LETTERS

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

- Interim Letter 5: Chapters 19 and 22 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Certification of the ESBWR Design, dated October 29, 2008

Letter to Dr. Brian Sheron, Director, Office of Nuclear Regulatory Research, NRC, from William J. Shack, Chairman, ACRS:

- ACRS Assessment of the Quality of Selected NRC Research Projects - FY 2008, dated October 22, 2008

MEMORANDA

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

- Proposed Interim Staff Guidances COL/ESP-ISG-004 and DC/COL-ISG-07, dated October 8, 2008

- Draft Final Revision to Regulatory Guides 1.114 and 3.11, dated October 8, 2008

- Withdrawal of Regulatory Guide 6.8, "Identification Plaque For Irretrievable Well-Logging Sources," dated October 8, 2008

MINUTES OF THE 556th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS OCTOBER 2-4, 2008 ROCKVILLE, MARYLAND

The 555th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on May 8-10, 2008. Notice of this meeting was published in the *Federal Register* on September 22, 2008 (72 FR 54635-54636) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting schedule and outline (Appendix II). The meeting was open to public attendance.

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

ATTENDEES

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

I. <u>Chairman's Report</u> (Open)

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

Dr. William J. Shack, Committee Chairman, convened the meeting at 8:30 a.m. In his opening remarks he announced that the meeting was being conducted in accordance with the provisions of the Federal Advisory Committee Act. He reviewed the agenda items for discussion and noted that no written comments or requests for time to make oral statements from members of the public had been received. Dr. Shack also noted that a transcript of the open portions of the meeting was being kept and speakers were requested to identify themselves and speak with clarity and volume. Dr. Shack announced that Mr. Charles Brown, Jr. is an official member of the committee with an expertise in digital instrumentation and control.

II. License Renewal Application for the Shearon Harris Nuclear Power Plant, Unit 1

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

The Committee met with representatives of the Carolina Power & Light Company (CP&L) (the applicant) and the NRC staff to discuss the license renewal application (LRA) for the Shearon Harris Nuclear Power Plant (HNP), Unit 1, and the associated NRC staff's final Safety Evaluation Report (SER). The operating license for HNP, Unit 1 expires on October 24, 2026. The applicant has requested approval for continued operation for a period of 20 years beyond the current license expiration date.

In the final SER, the staff documented its review of the license renewal application and other information submitted by CP&L and obtained during the audits and an inspection conducted at the plant site. The staff reviewed the completeness of the applicant's identification of structures, systems, and components (SSCs) that are within the scope of license renewal; the integrated plant assessment process; the applicant's identification of the plausible aging mechanisms associated with passive, long-lived components; the adequacy of the applicant's aging management programs; and the identification and assessment of time-limited aging analyses requiring review. The applicant addressed the open item, which is related to the scoping classification of feedwater regulating valves and bypass valves. HNP included these valves in the LRA within the scope of license renewal per the criteria of 10 CFR 54.4(a) (2), because it's believed that these valves are nonsafety-related components. However, the staff initially believed that these valves perform main feedwater isolation function under certain postulated transients and fulfill a safety-related function; therefore, these valves should be included within the scope under 10 CFR 54.4(a)(1). The applicant pointed out a past NRC precedence which allowed that credit be taken for the backup nonsafety-related components to mitigate the consequences of a main steamline break inside containment. The staff agreed with the applicant's assessment and determined that the feedwater regulating and bypass valves are properly categorized as nonsafety-related components, and that the requirements of 10 CFR 54.4(a)(2) apply to these valves. The applicant also discussed electrical manholes issue. The applicant described the cable design, manhole inspection program, historical operating experience, and corrective actions. The staff described its review of the applicant's scoping, screening, aging management programs, time-limited aging analyses, and the resolution of the open item, described above. The staff concluded that the requirement of 10 CFR 54.29(a) has been met.

The Committee issued a report to the NRC Chairman on this matter, dated October 16, 2008. The Committee concluded that the programs established and committed to by the applicant to manage age-related degradation provide reasonable assurance that the HNP, Unit 1 can be operated in accordance with the current licensing basis for the period of extended operation without undue risk to the health and safety of the public. The Committee recommended that the CP&L application for renewal of the operating license for HNP, Unit 1 be approved. In addition, the Committee recommended that prior to entering the period of extended operation, the staff inspect the applicant's programs for managing water intrusion into underground cable vaults and cable insulation testing.

III. Status of Resolution of Generic Safety Issue - 191

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

The Committee met with representatives of the NRC staff to discuss the status of resolution of Generic Safety Issue (GSI)-191, “Assessment of Debris Accumulation on PWR Sump Performance.”

The Committee noted that significant progress has been made towards resolving GSI-191. All PWR licensees have installed significantly larger sump screens and some have undertaken further actions, such as changing fibrous insulation and chemical buffer. The staff described progress made toward resolving GSI 191, including the response of licensees to Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors.” The presentation focused on sump strainer head loss testing, chemical effects, and in-vessel sump strainer downstream effects. The staff indicated that testing of the new strainer design has been performed by nearly all licensees and that the initial review of the industry submittals by the staff is nearing completion. The staff provided an overview of the testing that has been performed and the results obtained.

The Committee issued a report to the NRC Chairman on this matter, dated October 22, 2008, stating that significant progress has been made towards resolving GSI-191. The Committee recommended, to ensure the prototypicality of tests for extrapolation to plant conditions, further guidance be developed for the test cases in which a significant portion of the debris is allowed to settle out upstream of the screens. Also, the staff should develop guidance with regard to the conditions and protocols for the PWR Owners Group tests to facilitate closure of issues related to in-vessel downstream effects.

IV. Selected Chapters of the SER Associated with the ESBWR Design Certification Application

The Committee met with representatives of the NRC staff and General Electric – Hitachi Nuclear Energy (GEH) to discuss Chapters 19 and 22 of the NRC Staff’s SER with Open Items associated with the Economic Simplified Boiling Water Reactor (ESBWR) Design Certification Application.

GEH discussed the objectives of the Design Certification Probabilistic Risk Assessment (PRA), and stated that 386 of 450 requests for additional information on this topic have been resolved. GEH intends to issue Revision 4 of the PRA in December 2008. GEH described the nature and approach of the risk analysis calculations, discussed the results and their implications for the ESBWR risk profile, and noted that the PRA was a major influence on the ESBWR design. They also discussed the criteria used to classify structures, systems, and components (SSCs) as part of the “regulatory treatment of non-safety systems” (RTNSS) program. A high relative risk importance is one of several criteria which can add SSCs to this list.

NRC staff stated that its review of these Chapters covered overall PRA quality as well as the seismic margins analysis, high winds analysis, non-power operational modes, severe accident management, and severe accident mitigation. The staff discussed a number of open items which need to be resolved and also summarized its review of RTNSS. The objectives of this review were to confirm that all non-safety SSCs requiring treatment are identified, to confirm that the reliability and availability missions for active systems are consistent with the risk assessment, and to confirm that the level of treatment is based on the ability to meet reliability/availability missions.

The Committee issued a letter to the Executive Director for Operations on this matter, dated October 29, 2008. The Committee recommended that a more detailed explanation of the applicant's analysis of the low failure probabilities of the passive systems and a more systematic evaluation of the relevant uncertainties be provided. In addition, the technical basis for the failure probability of the digital instrumentation and control systems should be provided and specific issues need to be clarified to ensure the functionality of the Basemat-Internal Melt Arrest and Coolability (BiMAC) device. The Committee also stated that it is awaiting the completion of the staff's review of Revision 3 of the ESBWR PRA, and that it would review the resolution of open items in SER Chapters 19 and 22 in future meetings.

V. Quality Assessment of Selected Research Projects

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

The Committee completed its report on the quality assessment of the research projects on: "Assessment of Predictive Bias and the Influence of Manufacturing, Model, and Power Uncertainties in NRC Fuel Performance Code Predictions," and NUREG/CR - 6943, "A Study of Remote Visual Methods to Detect Cracking in Reactor Components." The Committee issued a letter to the Director of the Office of Nuclear Regulatory Research, dated October 22, 2008, transmitting its report on the quality assessment of the research projects noted above.

VI. Historical Perspectives and Insights on Reactor Consequence Analyses

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

The Committee discussed the draft White Paper prepared by Dr. Nourbakhsh, ACRS Senior Technical Advisor, regarding historical perspectives and insights on reactor consequence analyses. The Committee plans to issue a letter transmitting the White Paper to the Executive Director for Operations during its November 2008 meeting.

VII. Subcommittee Reports

Materials, Metallurgy, and Reactor Fuels Subcommittee Report

The Chairman of the Materials, Metallurgy, and Reactor Fuels Subcommittee provided a report to the Committee summarizing the results of the October 1, 2008, meeting with the NRC staff to review the proposed amendment to 10 CFR 50.61: "Fracture Toughness Requirements for

Protection against Pressurized Thermal Shock Events." He stated that the NRC staff provided a detailed discussion on the technical basis for the proposed supplemental rule, including treatment of uncertainties, PRA event sequence analysis, thermal hydraulic analysis, and probabilistic fracture mechanics analysis. The Subcommittee requested additional information regarding the independent review of the FAVOR code, the studies conducted to show the general applicability of the rule to the entire fleet of PWRs, and the justification for not treating thermal hydraulic parameter uncertainties. The staff is expected to provide additional information to the Subcommittee on these topics. The Subcommittee will review the draft final revision to 10 CFR 50.61 after reconciliation of public comments.

Reliability and PRA Subcommittee Report

The Chairman of the Reliability and Probabilistic Risk Assessment Subcommittee provided a report to the Committee summarizing the results of the September 28, 2008, meeting with the NRC staff and EPRI to discuss draft NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," and the latest version of the companion EPRI report, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments."

The draft NUREG-1855 is intended to provide the needed guidance recommended by the ACRS. EPRI, in parallel with the NRC, has been developing guidance documents on the treatment of uncertainties. This work is meant to complement the guidance included in NUREG-1855. Where possible the NUREG-1855 refers to the EPRI work for acceptable treatment of uncertainties. The staff plans to complete the NUREG report by the end of the calendar year and then hold a public workshop.

VI. Executive Session

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

A. Reconciliation of ACRS Comments and Recommendations/EDO Commitments

- The Committee will review the draft final revision to 10 CFR 50.61, after reconciliation of public comments. Additional information was requested on the independent review of the FAVOR code, the studies conducted to show the general applicability of the rule to the entire fleet of PWRs, and the justification for not treating thermal hydraulic parameter uncertainties.

- The Committee would like to have an opportunity to review the final versions of Interim Staff Guidances COL/ESP-ISG-004 and DC/COL-ISG-07, after reconciliation of public comments.

- The Committee plans to review draft NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," during a future meeting.

- The Committee plans to review the resolution of open items in SER Chapters 19 and 22 associated with the ESBWR design certification application.

B. Report of the Planning and Procedures Subcommittee Meeting

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

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

Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through February 2009 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

Meeting With the Commission

The ACRS is scheduled to meet with the Commission between 2:00 and 3:30 p.m. on Friday, November 7, 2008. Topics approved by the Commission are as follows:

1. Overview (WJS/SD)
    - Accomplishments
    - New Plant Activities
    - Ongoing/Future ACRS Activities, including challenges in the coming year
2. PWR Sump Performance Issues (SB/DB)
3. Committee Views on Power Uprates for BWRs (MVB/ZA)
4. TRACE Computer Code Development (SAK/HPN)

Operating Plan, Self-Assessment, and Summary Matrix

The ACRS Operating Plan, Self-Assessment, and Summary Matrix of the ACRS reports and letters issued in FY2008 were due to the Commission on October 31, 2008.

Proposed Revisions to the ACRS/EDO Memorandum of Understanding (MOU)

This MOU establishes a process for ensuring that (1) the NRC staff solicits ACRS views early in the development of NRC rules and safety- and risk-significant guidance; in licensing decisions; and in resolution of technical issues, (2) the NRC staff keeps the ACRS informed of emerging issues, and (3) the ACRS responds to staff requests for review and comment in a timely manner. The significant changes are in the following areas:

- Scope of Responsibility - Since the merger of the ACNW&M and ACRS, the scope of responsibility has been expanded to include items from the MOU between the EDO and ACNW. All references to ACNW&M have also been removed.

- Submittal of Documents - The current MOU states that documents needed for discussions at a Subcommittee meeting will be provided no later than two weeks before the Subcommittee meeting. The ACRS staff proposes to change the deadline to four weeks before the Subcommittee meeting. The deadline for providing documents for full committee meetings will remain at four weeks prior to the meeting. The MOU also states that exceptions will be made only with the agreement of the appropriate Office Director and the ACRS Executive Director.

- Resolving ACRS Comments – The current MOU states that the EDO will ensure consideration of ACRS comments by the NRC staff and will respond to ACRS comments in a timely manner. The proposed change by the OEDO staff would state, "The EDO will ensure consideration of ACRS comments by the NRC staff. If no response is required, the ACRS will indicate such on correspondence to the Commission or EDO. Otherwise, the EDO, or designee, will respond to ACRS comments in a timely manner. The EDO, or designee, may respond by email or letter addressed to the ACRS Chairman with a copy to the ACRS Executive Director."

<u>Draft Final Regulatory Guides</u>

The staff plans to issue the following Draft Final Regulatory Guides and would like to know whether the Committee wants to review these Guides prior to issuance.

- <u>Draft Final Revision 3 to Regulatory Guide 1.114 "Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Plant"</u>

Revision 3 to Regulatory Guide 1.114 was issued for public comment as DG-1194. The public comment period closed on June 6, 2008. This Guide describes a method acceptable to the NRC staff for complying with the Commission regulations that require the presence of an operator at the controls of a nuclear power plant unit and a senior operator in the control room. This Guide was updated to correct references and improve language to enhance understanding.

- <u>Draft Final Revision 3 to Regulatory Guide 3.11, "Design, Construction, and Inspection of Embankment Retention Systems at Uranium Recovery Facilities"</u>

Revision 3 to Regulatory Guide 3.11 was issued for public comment as DG-3032. The public comment period closed on May 16, 2008. This Guide has been updated to describe the latest engineering practices and methods generally considered by the NRC to be satisfactory for the design, construction, and inspection of the embankment retention systems used for retaining liquid and solid waste from uranium recovery operations.

Withdrawal of Regulatory Guide 6.8, "Identification Plaque for Irretrievable Well-Logging Sources"

The NRC is withdrawing Regulatory Guide 6.8, issued for public comments in October 1978, because it is no longer required. This Guide has never been issued as final. Regulatory Guide 6.8 references 10 CFR Parts 30 and 70, which no longer contain guidance for design or mounting of identification plaques. The current regulation regarding the identification plaques is found in 10 CFR 39.15, "Agreement with Well Owner or Operator." 10 CFR 39.15(a)(5) provides a specific description for the design and mounting of identification plaques for irretrievable well-logging sources. The instruction in 10 CFR 39.15(a)(5) is sufficient without further guidance.

The staff proposes to withdraw the Regulatory Guide 6.8 and seeks Committee's endorsement.

Proposed Interim Staff Guidances

The staff plans to issue the following proposed Interim Staff Guidances (ISGs) for public comment and would like to know whether the Committee wants to review these ISGs prior to issuance.

- Proposed Supplement to the Interim Staff Guidance COL/ESP-ISG-004, "Definition of Construction Interim Staff Guidance on Limited Work Authorizations"

This supplemental guidance would be added, along with the resolution of other issues identified in comments received from the public, to COL/ESP-ISG-004 to provide clarifications and examples related to the definition of construction. This additional guidance is intended to clarify the delineation of preconstruction activities and those activities that require prior NRC approval (i.e., construction activities). Upon receiving public comments, the NRC staff will evaluate and disposition the comments, as appropriate. Once the NRC staff completes the COL/ESP-ISG, including this supplemental information, the staff will issue it for use. The NRC staff will also incorporate the approved COL/ESP-ISG-004 into the next revisions of the Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants," and related guidance documents.

- Proposed Interim Staff Guidance DC/COL-ISG-07, "Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures"

The purpose of this ISG is to clarify the NRC position on identifying winter precipitation events as site characteristics and site parameters for determining normal and extreme winter precipitation loads on the roofs of Seismic Category I structures. This ISG revises the previously issued staff guidance in March 2007 in NUREG-0080, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants. " The NRC staff issues DC/COL-ISGs to facilitate timely implementation of the current staff guidance and to facilitate activities associated with review of applications for design certifications and combined licenses by the Office of New Reactors. The NRC staff will also incorporate the approved DC/COL-ISG-007 into the next revision of the Standard Review Plan and related guidance documents.

ACRS Retreat

During the September meeting, the Committee decided to hold a retreat on January 27-29, 2009. Because of the new conference room construction, arrangements are being made to hold the retreat at the Residence Inn, Bethesda. A proposed list of topics for discussion during the retreat is being prepared by the ACRS Executive Director.

Proposed Changes to Design Certification Rulemaking Process

As part of the EDO's Lean Six Sigma effort for streamlining the design certification (DC) rulemaking, the staff reviewed the role of the ACRS during the rulemaking process. A SECY paper is about to start the concurrence process. The staff believes that the design to be certified constitutes the "proposed reactor safety standard" that the ACRS must review under its statutory role, as opposed to the words of the design certification rule itself. As such, the ACRS fulfills its statutory obligation during its review of the staff's safety evaluation report (SER) and DC application. Therefore, the ACRS does not need to review separately the Federal Register Notice of either the proposed or final design certification rulemaking. Due to the standardized rule language being created for the DC rulemaking, and due to the ACRS' previous review of the technical basis (DC application and associated SER) for the rulemaking, the staff proposes to streamline the ACRS review by focusing its review on the technical comments made on the proposed rule. As a result, all technical comments and the staff's resolution of these comments will be sent to the ACRS for review. In addition, since ACRS briefings will not be expected, they will not be scheduled, but will be provided upon request.

The staff is proposing to address the ACRS in the upcoming months (brown bag lunch, possibly) so that the details about the standardized rule and the overall streamlined process can be better explained. The staff seeks Committee's views on the proposed approach.

Travel Request

Dr. Armijo requests Committee approval and support to attend the ANS 2008 Winter Meeting on November 9-13, 2008, in Reno, Nevada.

notice of proposed action and an opportunity for hearing or a notice of hearing is not warranted. Notice is hereby given of the right of interested persons to request a hearing on whether the action should be rescinded or modified.

Further Information:

For further details with respect to this action, see the application dated September 26, 2007, and Amendment No. 1, which are available electronically, at NRC's Electronic Reading Room, at: *http://www.nrc.gov/reading-rm/adams.html*. From this site, you can access NRC's Agencywide Document Access and Management System (ADAMS), which provides text and image files of NRC's public documents. The ADAMS accession number for the application is ML072820139 and the ADAMS accession number for Amendment No. 1 is ML082560545. If you do not have access to ADAMS, or if there are problems in accessing the documents located in ADAMS, contact NRC's Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov*.

These documents may also be viewed electronically on the public computers located at NRC's PDR, O1–F21, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852. The PDR reproduction contractor will copy documents, for a fee.

Dated at Rockville, Maryland, this 12th day of September, 2008.

For the Nuclear Regulatory Commission.

Kevin M. Witt,

Project Manager, Licensing Branch, Division of Spent Fuel Storage and Transportation, Office of Nuclear Material Safety and Safeguards.

[FR Doc. E8–22048 Filed 9–19–08; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on the Medical Uses of Isotopes: Meeting Notice

AGENCY: U.S. Nuclear Regulatory Commission.

ACTION: Notice of meeting.

SUMMARY: NRC will convene a meeting of the Advisory Committee on the Medical Uses of Isotopes (ACMUI) October 27–28, 2008. A sample of agenda items to be discussed during the public session includes: (1) ACMUI subcommittee reports on cesium chloride (CsCl), permanent implant brachytherapy rulemaking, and fingerprinting; (2) Y–90 microsphere brachytherapy licensing guidance; (3) potential changes to 10 CFR Parts 20 and 35; (4) patient needs, concerns, and rights in radiation medicine; (5) infiltration of fluorine-18 (F–18) and therapeutic radiopharmaceuticals as medical events; (6) status of recommendations for modifying training and experience attestation requirements; (7) status of technical basis for the Petition for Rulemaking (PRM) 35–20 (Ritenour) and follow-up; (8) Potential rulemaking and associated Regulatory Issue Summary (RIS) regarding multiple RSOs on a medical-use license; (9) status of current and future 10 CFR Part 35 rulemaking; and (10) medical isotope shortages. A copy of the agenda will be available at *http://www.nrc.gov/reading-rm/doc-collections/acmui/agenda* or by e-mailing Ms. Ashley Tull at the contact information below.

Purpose: Discuss issues related to 10 CFR Part 35 Medical Use of Byproduct Material.

Date and Time for Closed Sessions: October 27, 2008 from 8 a.m. to 9 a.m. This session will be closed so that ACMUI can discuss internal Committee business and receive annual ethics training.

Date and Time for Open Sessions: October 27, 2008, from 9 a.m. to 5 p.m. and October 28, 2008, from 8 a.m. to 4 p.m.

Address for Public Meeting: U.S. Nuclear Regulatory Commission, Two White Flint North Building, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland 20852.

Public Participation: Any member of the public who wishes to participate in the meeting should contact Ms. Tull using the information below.

FOR FURTHER INFORMATION CONTACT: Ashley M. Tull, e-mail: *ashley.tull@nrc.gov*, telephone: (240) 888–7129.

Conduct of the Meeting

Leon S. Malmud, M.D., will chair the meeting. Dr. Malmud will conduct the meeting in a manner that will facilitate the orderly conduct of business. The following procedures apply to public participation in the meeting:

1. Persons who wish to provide a written statement should submit an electronic copy to Ms. Tull at the contact information listed above. All submittals must be received by October 20, 2008, and must pertain to the topic on the agenda for the meeting.

2. Questions and comments from members of the public will be permitted during the meeting, at the discretion of the Chairman.

3. The transcript will be available on ACMUI's Web site (*http://www.nrc.gov/reading-rm/doc-collections/acmui/tr/*) on or about January 27, 2009. A meeting summary will be available on or about December 11, 2008.

4. Persons who require special services, such as those for the hearing impaired, should notify Ms. Tull of their planned attendance.

This meeting will be held in accordance with the Atomic Energy Act of 1954, as amended (primarily Section 161a); the Federal Advisory Committee Act (5 U.S.C. App); and the Commission's regulations in Title 10, *U.S. Code of Federal Regulations,* Part 7.

Dated: September 16, 2008.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E8–22066 Filed 9–19–08; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards

In accordance with the purposes of Sections 29 and 182b of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on October 2–4, 2008, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Monday, October 22, 2007 (72 FR 59574).

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

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

8:35 a.m.–10 a.m.: License Renewal Application and Final Safety Evaluation Report (SER) for the Shearon Harris Nuclear Power Plant, Unit 1 (Open)—The Committee will hear a briefing by and hold discussions with representatives of the NRC staff and Carolina Power & Light Company regarding the license renewal application for the Shearon Harris Nuclear Power Plant, Unit 1, and the associated NRC staff's final SER.

10:15 a.m.–12:15 p.m.: Status of Resolution of Generic Safety Issue (GSI)–191, ''Assessment of Debris Accumulation on Pressurized-Water Reactor (PWR) Sump Performance'' (Open)—The Committee will hear a briefing by and hold discussions with representatives of the NRC staff and PWR Owners Group regarding the staff

and industry activities associated with the resolution of GSI–191.

1:15 p.m.–3:15 p.m.: Selected Chapters of the SER Associated with the Economic Simplified Boiling Water Reactor (ESBWR) Design Certification Application (Open/Closed)—The Committee will hear a briefing by and hold discussions with representatives of the NRC staff and General Electric-Hitachi Nuclear Energy (GEH) regarding selected Chapters of the NRC staff's SER With Open Items associated with the ESBWR design certification application.

3:30 p.m.–4 p.m.: Quality Assessment of Selected Research Projects (Open)—The Committee will discuss the draft final report on the quality assessment of the NRC research projects on: FRAPCON/FRAPTRAN Code work at the Pacific Northwest National Laboratory (PNNL), and NUREG/CR–6943, ''A Study of Remote Visual Methods to Detect Cracking in Reactor Components.''

4 p.m.–5:15 p.m.: Historical Perspectives and Insights on Reactor Consequence Analyses (Open)—The Committee will discuss the draft White Paper prepared by the ACRS Senior Technical Advisor on historical perspectives and insights on reactor consequence analyses.

5:30 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters discussed during this meeting.

Friday, October 3, 2008, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

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

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

9:45 a.m.–10 a.m.: Subcommittee Reports (Open)—Report by and discussions with the Chairman of the ACRS Subcommittee on Materials, Metallurgy, and Reactor Fuels regarding Proposed Supplemental Pressurized Thermal Shock Rule (10 CFR 50.61) that was discussed during the Subcommittee meeting on October 1, 2008. Report by and discussions with the Chairman of the ACRS Subcommittee on Reliability and PRA regarding the draft final NUREG–1855, ''Guidance on the Treatment of Uncertainties in Risk-Informed Decisionmaking,'' that was discussed during the Subcommittee meeting on September 30, 2008.

10:15 a.m.–11:30 a.m.: Preparation for Meeting with the Commission on November 7, 2008 (Open)—Discussion of proposed topics for meeting with the Commission on November 7, 2008.

12:30 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Saturday, October 4, 2008, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

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

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

In accordance with Subsection 10(d) Public Law 92–463, I have determined that it may be necessary to close portions of this meeting noted above to discuss organizational and personnel matters that relate solely to the internal personnel rules and practices of the ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy pursuant to 5 U.S.C. 552b(c)(2) and (6). In addition, it may be necessary to close a portion of the meeting to protect information designated as proprietary by General Electric-Hitachi or its contractors pursuant to 5 U.S.C. 552b c(4).

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

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

Dated: September 16, 2008.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E8–22069 Filed 9–19–08; 8:45 am]

BILLING CODE 7590–01–P

Appendix II

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

September 11, 2008

AGENDA
556th ACRS MEETING
OCTOBER 2-4, 2008

THURSDAY, OCTOBER 2, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - 10:00 A.M. License Renewal Application and Final Safety Evaluation Report (SER) for the Shearon Harris Nuclear Power Plant, Unit 1 (Open) (JS/PW)
- 2.1) Remarks by the Subcommittee Chairman
- 2.2) Briefing by and discussions with representatives of the NRC staff and Carolina Power & Light Company regarding the license renewal application for the Shearon Harris Nuclear Power Plant, Unit 1, and the associated NRC staff's final Safety Evaluation Report (SER).

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

10:00 - 10:15 A.M. * BREAK *****

3) 10:15 -12:15 P.M. Status of Resolution of Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor (PWR) Sump Performance" (Open) (SB/DAW/DB)
- 3.1) Remarks by the Subcommittee Chairman
- 3.2) Briefing by and discussions with representatives of the NRC staff and PWR Owners Group regarding the staff and industry activities associated with the resolution of GSI-191.

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

12:15 - 1:15 P.M. * LUNCH *****

4) 1:15 - 3:15 P.M. Selected Chapters of the SER Associated with the Economic Simplified Boiling Water Reactor (ESBWR) Design Certification Application (Open/Closed) (MLC/HJV)

4.1) Remarks by the Subcommittee Chairman

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

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

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

3:15 - 3:30 P.M. * BREAK *****

5) 3:30 - 4:00 P.M. Quality Assessment of Selected Research Projects (Open) (DAP/HPN)

5.1) Remarks by the Subcommittee Chairman

5.2) Discussion of the draft final report on the quality assessment of the NRC research projects on: FRAPCON / FRAPTRAN Code work at the Pacific Northwest National Laboratory (PNNL), and NUREG/CR - 6943, "A Study of Remote Visual Methods to Detect Cracking in Reactor Components."

6) 4:00 - 5:15 P.M. Historical Perspectives and Insights on Reactor Consequence Analyses (Open) (WJS/HPN)

Discussion of the draft White Paper prepared by Dr. Nourbakhsh, ACRS Senior Technical Advisor, regarding historical perspectives and insights on reactor consequence analyses.

5:15 - 5:30 P.M. * BREAK *****

7) 5:30 - 7:00 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

7.1) License Renewal Application for the Shearon Harris Nuclear Power Plant, Unit 1 (JS/PW)

7.2) Status of Resolution of Generic Safety Issue - 191 (SB/DAW/DB)

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

FRIDAY, OCTOBER 3, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

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

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

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

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

10) 9:30 - 9:45 A.M. Reconciliation of ACRS Comments and Recommendations (Open) (WJS/CS/AFD)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

11) 9:45 – 10:00 A.M. Subcommittee Reports (Open)

11.1) Report by and discussions with the Chairman of the ACRS Subcommittee on Materials, Metallurgy, and Reactor Fuels regarding Proposed Supplemental Pressurized Thermal Shock Rule (10 CFR 50.61) that was discussed during the Subcommittee meeting on October 1, 2008. (WJS/CLB)

11.2) Report by and discussions with the Chairman of the ACRS Subcommittee on Reliability and PRA regarding the draft final NUREG-1855, "Guidance on the Treatment of Uncertainties in Risk-Informed Decisionmaking," that was discussed during the Subcommittee meeting on September 30, 2008. (GEA/HJV)

10:00 - 10:15 A.M. * BREAK *****

12) 10:15 – 11:30 A.M. Preparation for Meeting with the Commission on November 7, 2008 (Open) (WJS, et al. /EMH, et al.)
Discussion of proposed topics for meeting with the Commission on November 7, 2008.

11:30 - 12:30 P.M. ***** LUNCH *****

13) 12:30 - 7:00 P.M. Preparation of ACRS Reports (Open)
Continued discussion of proposed ACRS reports on:
13.1) License Renewal Application for the Shearon Harris Nuclear Power Plant, Unit 1 (JS/PW)
13.2) Status of Resolution of Generic Safety Issue - 191 (SB/DAW/DB)
13.3) Selected Chapters of the SER Associated with the ESBWR Design Certification Application (MLC/HJV)

SATURDAY, OCTOBER 4, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 8:30 - 1:00 P.M. Preparation of ACRS Reports (Open)
(10:30-10:45 A.M. BREAK) Continue discussion of the proposed ACRS reports listed under Item 13.

15) 1:00 - 1:30 P.M. Miscellaneous (Open) (WJS/EMH)
Discussion of matters related to the conduct of Committee activities and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTES:

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

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

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

Appendix III

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

October 17, 2008

AGENDA
557th ACRS MEETING
NOVEMBER 6-8, 2008

THURSDAY, NOVEMBER 6, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 – 10:00 A.M. Chapter 14 of the SER Associated with the Economic Simplified Boiling Water Reactor (ESBWR) Design Certification Application (Open/Closed) (MLC/HJV)
 2.1) Remarks by the Subcommittee Chairman
 2.2) Briefing by and discussions with representatives of the NRC staff and General Electric - Hitachi Nuclear Energy (GEH) regarding Chapter 14, "Verification Programs," of the NRC staff's SER With Open Items associated with the ESBWR design certification application.

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

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

10:00 – 10:15 A.M. * BREAK *****

3) 10:15 – 12:00 P.M. Position Paper on Incorporating the International Commission on Radiological Protection (ICRP) Recommendations into 10 CFR Parts 20 and 50 (Open) (MTR/NMC)
 3.1) Remarks by the Subcommittee Chairman
 3.2) Briefing by and discussions with representatives of the NRC staff regarding their plans to develop options to revise NRC regulations and guidance in light of the new recommendations of the ICRP.

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

12:00 – 1:00 P.M. ***** LUNCH *****

4) 1:00 – 2:30 P.M. Status of License Renewal Activities (Open) (MVB/PW)
- 4.1) Remarks by Subcommittee Chairman
- 4.2) Briefing by and discussions with representatives of the NRC staff regarding the status of the license renewal activities, Interim Staff Guidance, and implementation of the recommendations from the self-assessment.

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

2:30 – 2:45 P.M. ***** BREAK *****

5) 2:45 – 3:15 P.M. Subcommittee Reports (Open)
- 5.1) Report by and discussions with the Chairman of the US-Advanced Pressurized Water Reactor (US-APWR) Subcommittee regarding Topical Reports associated with the US-APWR design that was discussed during the Subcommittee meetings on October 23-24, and November 4-5, 2008. (OLM/NMC)
- 5.2) Report by and discussions with the Chairman of the Plant License Renewal Subcommittee regarding the license renewal application for the Vogtle Plant that was discussed on November 5, 2008. (JDS/CLB)

6) 3:15 – 7:00 P.M. Preparation of ACRS Report (Open)
Discussion of proposed ACRS report on:
- 6.1) Position Paper on incorporating ICRP Recommendations into10 CFR Parts 20 and 50 (MTR/NMC)

FRIDAY, NOVEMBER 7, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

8) 8:35 – 10:00 A.M. Current Issues Associated with Fire Protection and Related Matters (Open) (JDS/PW)
- 8.1) Remarks by the Subcommittee Chairman
- 8.2) Briefing by and discussions with representatives of the NRC staff regarding current fire protection issues, such as, the fire protection issues closure plan, Commission direction to the staff on fire protection issues, GAO recommendations and planned staff actions, and draft Regulatory Guides for implementing National Fire Protection Association (NFPA) – 805 Standard, and related matters.

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

10:00 – 10:15 A.M. ***** BREAK *****

9) 10:15 – 11:15 A.M. Proposed Changes to the Review Process for Subsequent Combined License Applications (SCOLAs) (Open) (MLC/YKS)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussion with representatives of the NRC staff regarding the proposed changes to the SCOLA review process and related matters.

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

10) 11:15 – 12:00 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed) (WJS/EMH)

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

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

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

12:00 – 1:00 P.M. ***** LUNCH *****

11) 1:00 – 1:45 P.M. Preparation for Meeting with the Commission (Open) (WJS, et al. /EMH, et al.)

Discussion of the following topics for meeting with the Commission:

- Overview (WJS/SD)
- PWR Sump Performance Issues (SB/DB)
- Committee Views on Power Uprates for BWRs (MVB/ZA)
- Development of the TRACE Thermal-Hydraulic System Analysis Code (SAK/HPN)

1:45 – 2:00 P.M. ***** BREAK *****

12)	2:00 – 3:30 P.M.	<u>Meeting with the Commission</u> (Open) (WJS, et al. /EMH, et al.) Meeting with the Commission, Commissioners' Conference Room, One White Flint North, to discuss topics listed under Item 11.
	3:30 – 4:00 P.M.	***** BREAK *****
13)	4:00 – 4:15 P.M.	<u>Reconciliation of ACRS Comments and Recommendations</u> (Open) (WJS/CS/AFD) Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.
14)	4:15 – 7:00 P.M. **(6:00-6:15 P.M. BREAK)**	<u>Preparation of ACRS Report</u> (Open) Continue discussion of the proposed ACRS report listed under Item 6.

<u>SATURDAY, NOVEMBER 8, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND</u>

15)	8:30 – 12:30 P.M. **(10:30-10:45 A.M. BREAK)**	<u>Preparation of ACRS Report</u> (Open) Continue discussion of the proposed ACRS report listed under Item 6.
16)	12:30 – 1:00 P.M.	<u>Miscellaneous</u> (Open) (WJS/EMH) Discussion of matters related to the conduct of Committee activities and specific issues that were not completed during previous meetings, as time and availability of information permit.

<u>NOTES:</u>

- During the days of the meeting, phone number 301-415-7360 should be used in order to access anyone in the ACRS Office.
- Presentation time should not exceed 50 percent of the total time allocated for a given item. The remaining 50 percent of the time is reserved for discussion.
- Thirty five (35) hard copies and one (1) electronic copy of the presentation materials should be provided to the ACRS in advance of the briefing.

Appendix IV

LIST OF DOCUMENTS FROM THE
556th ACRS MEETING OCTOBER 2-4, 2008

Agenda Item 2:
<u>License Renewal Application and Final Safety Evaluation Report (SER) for the Shearon Harris Nuclear Power Plant, Unit 1</u>

1. Proposed Schedule
2. Status Report
3. SER Open Item Resolution

Agenda Item 3:
<u>Status of Resolution of Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor (PWR) Sump Performance"</u>

4. Table of Contents
5. Proposed Schedule
6. Status Report
7. Attachments

Agenda Item 4:
<u>Selected Chapters of the SER Associated with the Economic Simplified Boiling Water Reactor (ESBWR) Design Certification Application</u>

8. Proposed Schedule
9. Status Report
10. Attachments

Harris Nuclear Plant

ACRS License Renewal Presentation
October 2, 2008

Shearon Harris Nuclear Plant License Renewal Representatives

- Mike Heath – License Renewal Supervisor
- Dave Corlett – Licensing/Regulatory Programs Supervisor
- Matt Denny – Equipment Performance Supervisor
- Chris Mallner – License Renewal Mechanical Lead

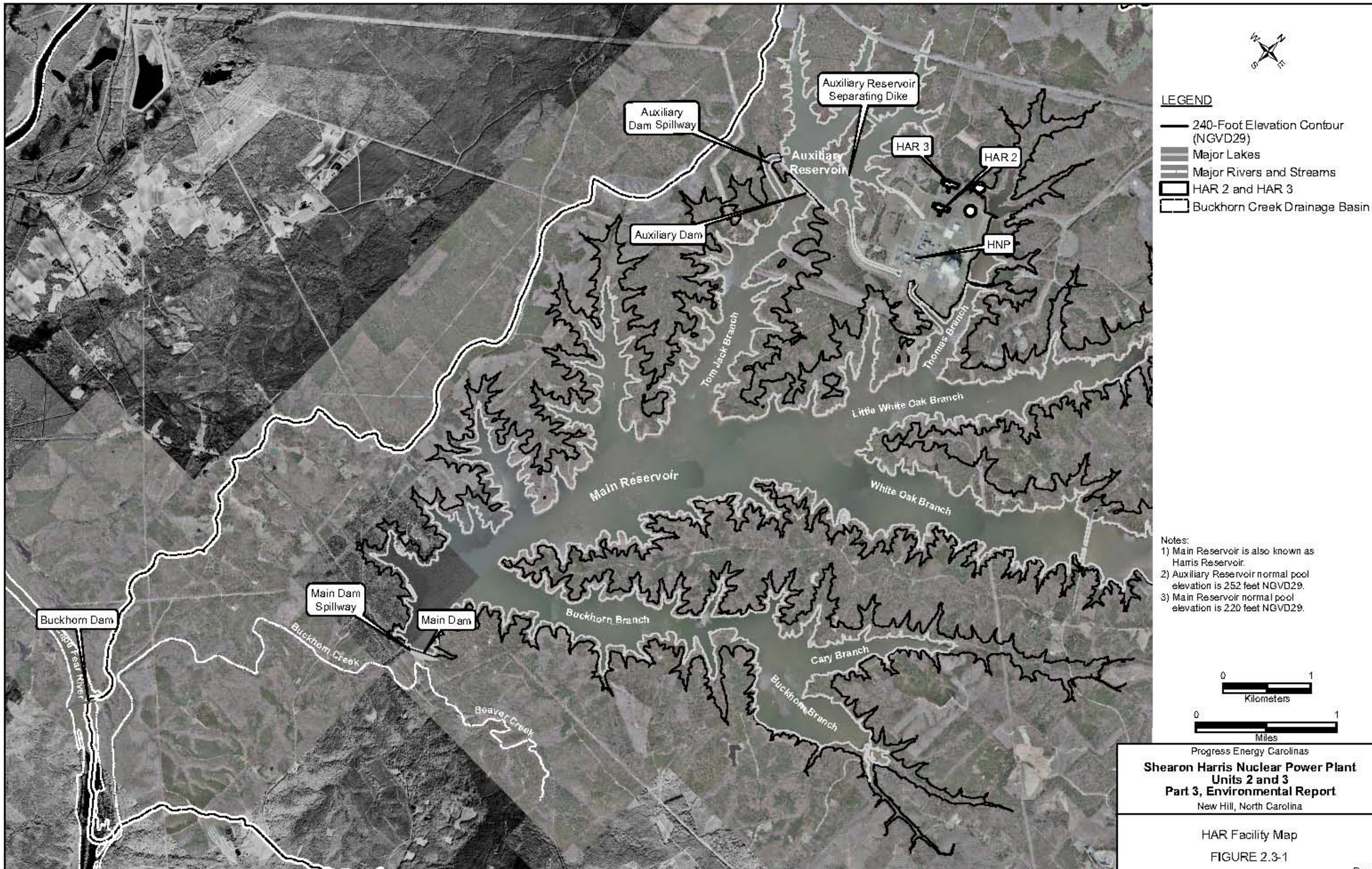
Agenda

- Introductions – Mike Heath
- Harris Plant Information – Dave Corlett
- HNP Water Sources – Dave Corlett
- Feedwater Regulating Valves Open Item – Dave Corlett
- Status of Electrical Manholes – Mike Heath
- Containment Valve Chamber External/Internal Corrosion – Matt Denny

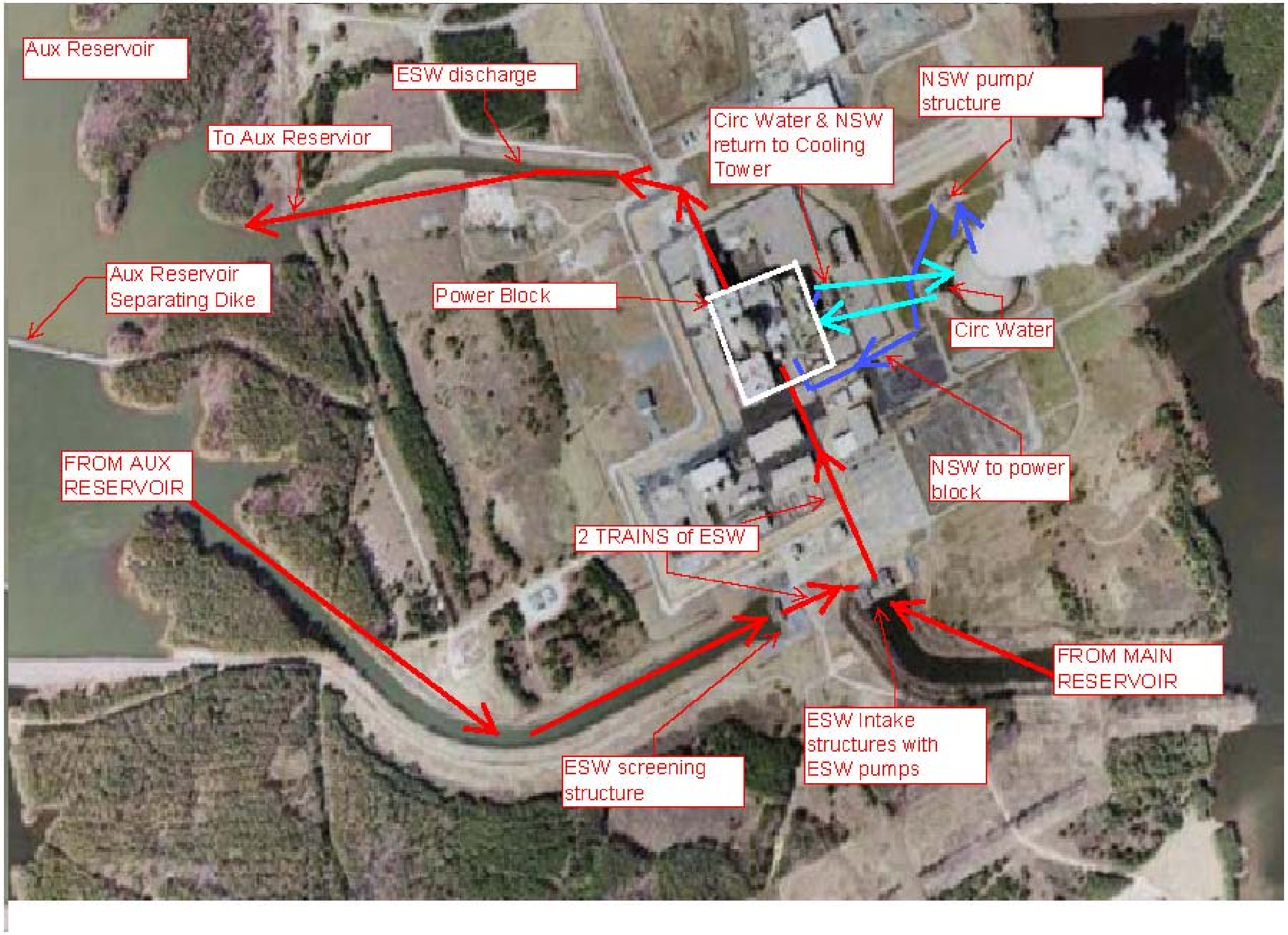
Shearon Harris Plant

- Located South of Raleigh, NC on Harris Lake
- Facility License Issued October 24, 1986
- Westinghouse 3 Loop PWR
  - 2900 MWt; 900 MWe(net)
  - Steel lined, reinforced concrete containment
  - UHS - Cooling via lake with Cooling Tower

HNP Water Sources

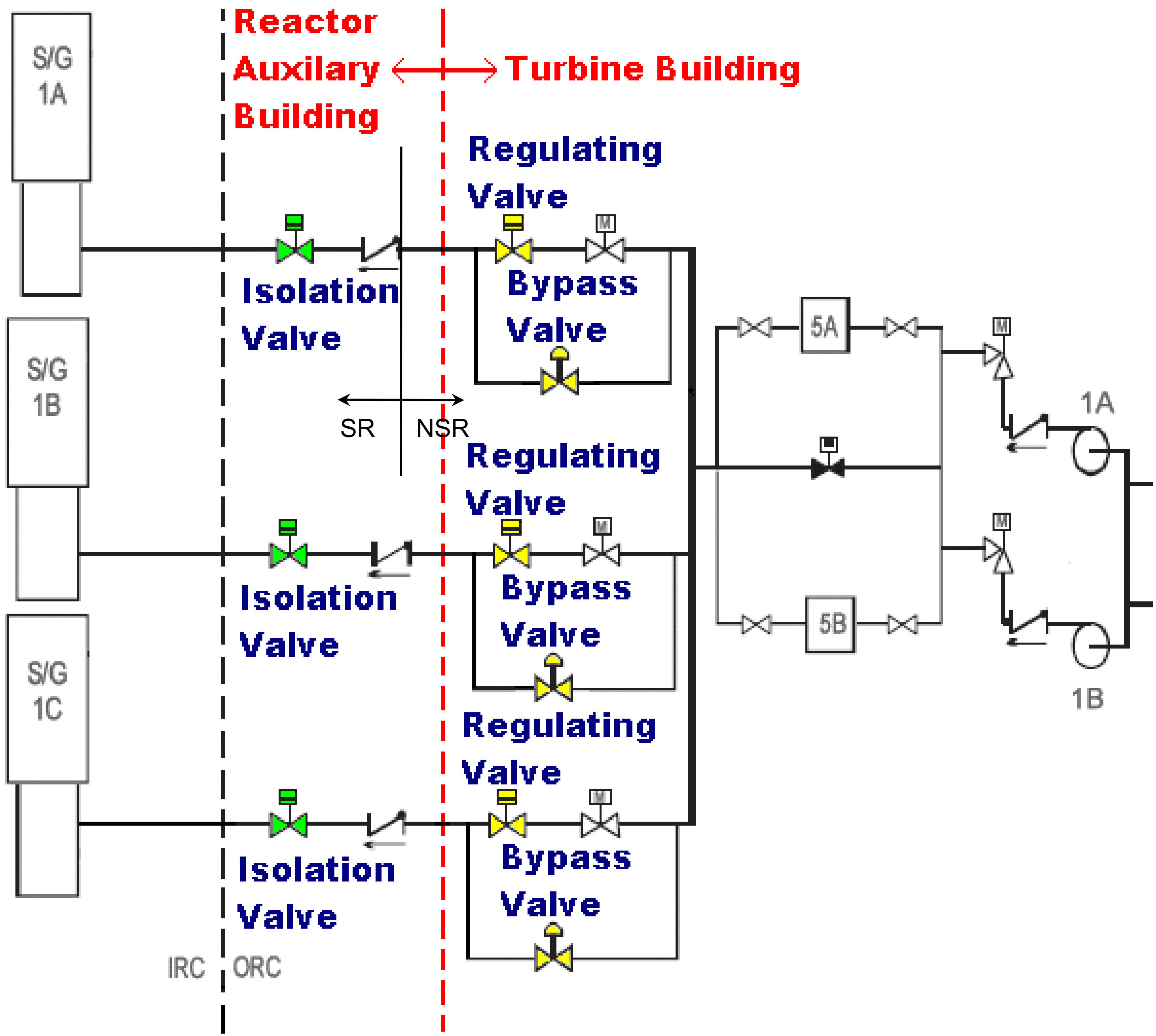


HNP Water Sources & Flow Diagram



Feedwater Regulating Valve Open Item Discussion

- Scoping
  - The Feedwater Regulating Valves Scoped Per 10 CFR 54.4(a)(2) versus (a)(1)



S/G 1A
S/G 1B
S/G 1C
Reactor Auxilary Building
Turbine Building
Regulating Valve
Isolation Valve
Bypass Valve
SR
NSR
Regulating Valve
Isolation Valve
Bypass Valve
Regulating Valve
Isolation Valve
Bypass Valve
5A
5B
1A
1B
IRC
ORC

Feedwater Regulating Valve Open Item Discussion

- Feedwater Regulating Valves and Bypass Valves are nonsafety-related
  - Not Protected From Hazards per CLB
- Safety Function Accomplished by Feedwater Isolation Valves
- Consistent with NUREG-0138, Issue 1, “Treatment of Non-Safety Grade Equipment in Evaluation of Postulated Steam Line Break Accidents.”

Feedwater Regulating Valve Open Item Discussion

- Feedwater Regulating Valves and Bypass Valves Safety Factors
  - Valves close on
    - Main Feedwater Isolation Signal
    - Loss of Instrument Air System
    - Loss of power from Engineered Safety Features Actuation System
    - Loss of DC electric power to solenoids
  - Designed to ASME Section III, Class 3 and Seismic Category 1

Electrical Manholes

- HNP has had two 6.9 kV cable failures:
  - Cable 11525A – MCC 1-4A101 Feeder failed on December 11, 2002 after approximately 15 years in service.
  - Cable 11882A – 1&2X CTMU Pump failed on January 12, 2006 after approximately 19 years in service.

Electrical Manholes

- Base line inspections of all manholes were completed in 2003
- Manholes are pumped down every 90 days
  - SR manhole M505B-SB is pumped down every 45 days
- Water levels trended
  - Some water levels over cables

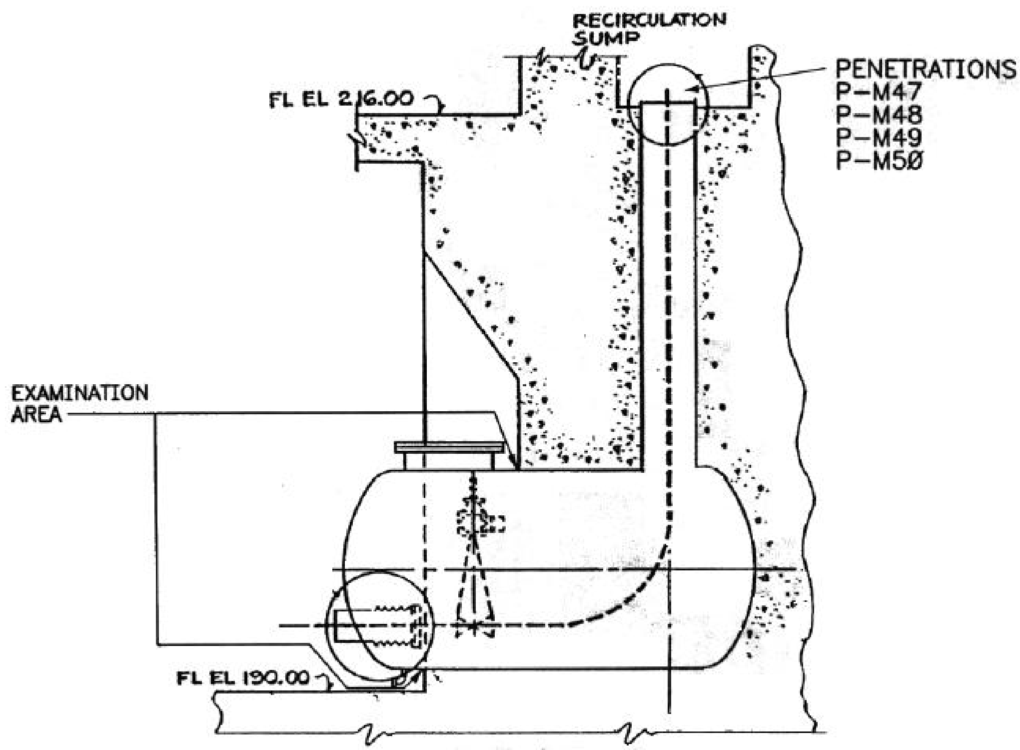
SR Manhole M523D-SB

Electrical Manholes

- Medium voltage wetted cables are tested every 6 years
  - Use High Voltage - Very Low Frequency Tan Delta Testing
  - Total of 17 cables
  - Normal Service Water Pump ‘B’, Emergency Service Water Pump ‘A’, and Circulating Water Pump ‘C’ cables tested satisfactorily
  - Maintenance shop feeder cable tested unsatisfactorily

Containment Valve Chamber Corrosion

Containment Valve Chamber Corrosion



ELEVATION VIEW

CT AND RHR VALVE CHAMBERS

Containment Valve Chamber External Corrosion

- Ground Water Intrusion EL 190’ & 216’ RAB
- Detected as early as the1980’s
- 1984 - Pressure grouting
- Later other techniques used
  - e.g. sealant injection (floors & exterior walls)

Containment Valve Chamber External Corrosion

- Water In-leakage Action Plan (1996)
- 15 general areas in several structures
- Corrective actions include:
  - Channeling water in-leakage to floor drains
  - Design changes to core bore drain holes
  - Sump Pumps installed
- Continuing to monitor in-leakage locations

Containment Valve Chamber External Corrosion

- Structures Monitoring Program
  - Engineering personnel inspect SSCs for in-leakage impacts
    - RAB every 6 years
    - FHB and WPB every 7 years
- QC personnel inspect per IWE every ISI period
- HNP Maintenance maintains water control measures
- External surfaces recoated to prevent corrosion

Containment Valve Chamber Internal Corrosion

- RFO10 (2000)
  - Some small blisters on floors of chambers – found acceptable
  - Apparent cause was condensation
- RFO12 (2004)
  - Corrosion under blisters on floor of chambers
  - UT showed wall thickness were above nominal thickness
  - Cause was degraded coatings

Containment Valve Chamber Internal Corrosion

- RFO13 (2006)
  - Coatings were repaired with improved material
- RFO14 (2007)
  - No indications
- QC inspects per IWE every ISI period

Containment Valve Chamber Corrosion

- Conclusion
  - Valve chamber integrity maintained by routine inspections and maintenance

Questions

Advisory Committee on Reactor Safeguards (ACRS) License Renewal Full Committee

Shearon Harris Nuclear Power Plant Unit 1

Safety Evaluation Report

October 2, 2008

Maurice Heath, Project Manager
Office of Nuclear Reactor Regulation

Introduction

- Overview
- Resolution of Open Item 2.2
- Resolution Confirmatory Item 3.4-1
- Resolution Confirmatory Item 4.3

Overview

- License Renewal Application submitted by letter dated November 14, 2006
- Single Unit, Westinghouse 3-Loop - PWR
- 2900 megawatt thermal, 900 megawatt electric
- Operating license NPF-63 expires October 24, 2026
- Location is approximately 20 miles SW of Raleigh, NC

Overview

- Safety Evaluation Report with Open Item was issued March 18, 2008
  - One (1) open item
  - Two (2) confirmatory items
- 346 Audit Questions
- 75 RAIs Issued
- 35 Commitments

Overview

- SER issued August 21, 2008
- Resolution of Open Item (OI) 2.2
- Resolution of Confirmatory Items (CI) 3.4-1 and CI 4.3
- 2 additional commitments added, which were added to resolve the two confirmatory items

Section 2.2: Plant Level Scoping

<u>OI - 2.2</u>

– HNP FSAR credits feedwater regulating and bypass valves for redundant isolation function following a main steam line break. Feedwater isolation is not listed as a function of the feedwater system in the LRA

– The LRA states that the feedwater regulating and bypass valves are non-safety related (NSR), per the CLB and are in scope per 10 CFR 54.4(a)(2)

Section 2.2: Plant Level Scoping

<u>OI - 2.2</u>

- In addressing this OI the staff identified the following:
  - 54.4(a)(1) specifies that safety-related SSCs should be included in scope if they meet 54.4(a)(1)(i),(ii), or (iii)
  - The criteria in 54.4(a)(1)(i-iii) agrees with the definition of safety-related specified in 10 CFR 50.2

Section 2.2: Plant Level Scoping

OI - 2.2

- If the applicants definition of safety-related (SR) differs from 54.4(a), then NEI 95-10 states that applicants should use the criteria of 54.4(a)(1)(i-iii) to determine what SSCs to include in scope.
- If an applicant has CLB documentation indicating the NRC has approved specific SSCs that to be classified as NSR, which would otherwise meet the applicants definition of SR or the 54.4(a)(1) criteria, these SSCs are not required to be within scope in accordance with 54.4(a)(1)

Section 2.2: Plant Level Scoping

OI - 2.2

– If SSCs, classified NSR in accordance with CLB, have the potential to affect the functions described in 54.4(a)(1) they should be included within scope in accordance with 54.4(a)(2) – nonsafety-related affecting safety-related.

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Section 2.2: Plant Level Scoping

OI - 2.2

- Resolution
  - LRA Amendment 8, dated May 30, 2008, revised Section 2.3.4.6 to add feedwater isolation as an intended function in the Feedwater System
  - HNP has CLB documentation indicating the NRC has approved classifying these valves as NSR
  - LRA Amendment 8, HNP took exception to scoping methodology in NEI 95-10 and used the CLB and scoping definition in 54.4 to determine the valves are in scope per 54.4(a)(2)
  - The staff agrees with the this position as it is consistent with the CLB and scoping definition in 10 CFR 54.4

Section 3: Aging Management Review Results

- <u>Confirmatory Item 3.4-1</u>
  - Applicant credits managing changes in materials and cracking of elastomeric and other plastic components with External Surfaces Monitoring Program
  - GALL AMP XI.M36 recommends visual inspection for carbon steel components but does not address elastomeric and other plastic components
- Resolution
  - Applicant will use the preventative maintenance program, which will periodically replace these components based on site and industry operating experience, equipment history, and vendor recommendations

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Protecting People and the Environment

Section 4: Time-Limited Aging Analysis

- <u>Confirmatory Item 4.3</u>
  - Applicant used WESTEMS™ special purpose computer code in calculating stresses from thermal transients
  - The code is bench marked for pressure, external moments, and thermal transients
  - 60-year fatigue reanalyses were completed for all NUREG/CR 6260 components with two (2) components having 60-year CUFen>1.0
  - CI 4.3 was issued to ensure consistency between reanalysis and original design specification

Section 4: Time-Limited Aging Analysis

CI - 4.3

- Resolution
  - HNP committed to update the design specification to reflect the revised design basis operating transients (Commitment 37)
  - The FSAR supplement was updated to reflect HNP's crediting of the fatigue monitoring program to manage aging for reactor coolant pressure boundary components according to 10 CFR 54.21(c)(1)(iii)

Conclusion

On the basis of its review, the staff determines that the requirements of 10 CFR 54.29(a) have been met.

QUESTIONS

Presentation to the ACRS Full Committee

ESBWR Design Certification Review
Chapter 19 & 19A

Presented by
NRO/DNRL/NGE1 and NRO/SPLB

October 2, 2008

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 19

Purpose:

- Brief the Committee on the status of the staff's review of the ESBWR DCD application, Chapter 19 and 19A (RTNSS)

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 19

Review Team for Chapter 19:

- Lead Technical Reviewer
  – Mark Caruso, Sr. Risk & Reliability Engineer
- Technical Reviewers
  – Edward Fuller, Sr. Risk & Reliability Engineer
  – Marie Pohida, Sr. Risk & Reliability Engineer
  – Glenn Kelly, Sr. Risk & Reliability Engineer
  – John Lai, Risk & Reliability Engineer
  – Jim Xu, Sr. Structural Engineer

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 19

Outline of Presentation:

- Objectives of Staff's review
- Summary of Staff's review
- Open Items

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 19

Commission's Objectives:

- Use the PRA to identify and address potential design features and plant operational vulnerabilities.
- Use the PRA to reduce or eliminate the significant risk contributors
- Use the PRA to select among alternative features and design options.
- Identify risk-informed safety insights
- Determine how the risk associated with the design compares against the Commission's goals of less than 1x10-4/yr for CDF and less than 1x10-6/yr for LRF and containment performance goals
- Assess the balance between severe accident prevention and mitigation.
- Determine whether the plant design represents a reduction in risk compared to existing operating plants
- Demonstrate compliance with 10 CFR 50.34(f)(1)(i) (i.e., perform a PRA)
- Use PRA in support of programs and processes (e.g., RTNSS, RAP)

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 19

Areas of Review with Open Items

- PRA Quality
- Seismic Margins Analysis
- High Winds Analysis
- PRA for Non-power Operational Modes
- Severe Accident Mitigation
- Severe Accident Management

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 19

Open Items
PRA Quality

- Applicant's basis for stating PRA quality is adequate for design certification not provided in DCD
  - GEH response to RAI 19.1-155 acceptable
  - Staff will confirm quality, including completeness, of PRA Rev. 3 in site audit
- Concerns with success criteria for passive systems resolved

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 19

Open Items
Seismic Margins Analysis

- GEH used a spectrum shape different from the Certified Seismic Design Response Spectra (CSDRS) for HCLPF* estimates in Seismic Margins Analysis (SMA)

- Majority of SSCs treated in SMA assume a HCLPF equal to the limit of 1.67xSSE; however, the SSE has not been defined as CSDRS in the DCD.

- Staff requested that GEH include an ITACC for verification of the assumed seismic capacity for differential building displacements of 1.67***CSDRS**. Staff is awaiting response to RAI from GEH.

*High Confidence of Low Probability of Failure defined as: Earthquake level at which, with high confidence (95 percent), it is unlikely (probability less than 5x10-2) that failure of the SSC will occur.

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 19

Open Items
High Winds Analysis

- Assumed conditional probability that Category 4 or 5 hurricanes will damage structures not justified
  - Awaiting GEH response to RAI

- Not clear whether credit was taken for equipment in Seismic Category II structures hit by tornado missiles
  - Awaiting GEH Response to RAI

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 19

Open Items
PRA for Other Operational Modes

- Staff requests GE to add DPS operability to TS for Modes 5 and 6 or assess risk of RWCU/SDC breaks outside of containment (RAI 19.1.-178)
- Staff requests GE to document sizes of piping penetrations and associated alarm/position indication upstream of RWCU/SDC isolation valves or assess operator induced leaks (RAI 19.1.0-4 Supplement 2)
- Staff questions ability of Isolation Condenser to function effectively for some operational conditions in Mode 5 (RAI 19.1-144 Supplement 2)

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 19

Open Items
PRA for Other Operational Modes

- GEH must determine range of conditions (temperature and level) for which the RWCU/SDC can adequately remove decay heat in Modes 4, 5, and 6 (RAI 5.4-59 Supplement 1)
  - Staff concerned about inadequate vessel circulation between inside and outside shroud
  - Staff concerned that RWCU/SDC injection may bypass the core due to inadequate mixing in downcomer.

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 19

Open Items

Severe Accident Mitigation

- BiMAC performance test report
  - Response to RAIs19.2-23 S02 and 19.2-25 S02 included a topical report documenting the results of the BiMAC tests.
  - Topical report NEDE-33392 has been reviewed and 27 RAIs prepared.

- Sent a new RAI to GEH asking for transient analyses of BiMAC behavior during severe accidents for both high and low RCS pressure scenarios.

ACRS Subcommittee Presentation
ESBWR Design Certification Review
Chapter 19

Open Items

Accident Management

- Description of the process for developing Severe Accident Guidelines
  - The staff requested additional information on the process that will be used by GEH to develop the Severe Accident Guidelines (SAGs) in RAI 19.2.4-1 and its supplements.
  - A new supplemental RAI has been issued, asking for the technical basis for ESBWR severe accident management.

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 19A (SER Chap. 22)

Review Team for Chapter 19A (SER Chap. 22):

- Lead Technical Reviewer
  - Mark Caruso, Sr. Risk & Reliability Engineer
- Technical Reviewers
  - Eugene Eagle, Instrumentation and Controls Engineer
  - Craig Harbuck, Sr. Operations Engineer
  - Thomas Scarbrough, Sr. Mechanical Engineer
  - Mohamed Shams, Structural Engineer
  - David Shum, Sr. Reactor Systems Engineer
  - George Thomas, Sr. Reactor Systems Engineer
  - Hanry Wagage, Sr. Reactor Engineer

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 19A (SER Chap. 22)

Outline of Presentation:

- Objectives of Staff’s review
- Summary of Staff’s review
- Open Items

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 19A (SER Chap. 22)

Regulatory Treatment of Non-Safety Systems (RTNSS)

Objectives of Staff's Review

- Confirm all non-safety SSCs requiring treatment are identified
- Confirm reliability and availability (R/A) missions for active systems are consistent with risk assessment
- Confirm level of treatment is based on ability to meet R/A missions (i.e., TS, Availability Controls Manual, Maintenance Rule program)

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 19A (SER Chap. 22)

Areas of Review with Open Items

- Augmented Design Standards for Post-72 hour equipment
- Regulatory Treatment of Active Systems
- Availability Controls

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 19A (SER Chap. 22)

Open Items

Augmented Design Standards for Post-72 Hours Equipment

- Staff is satisfied that RTNSS systems can be adequately protected from flood-related effects associated with both natural phenomena and system and component failures (design meets standards).
- Staff wants GEH to propose an ITAAC to ensure as-built plant implements the design properly.

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 19A (SER Chap. 22)

Open Items

Regulatory Treatment

- Risk significance criteria for determining treatment level of active systems applied inconsistently
  - Awaiting GEH response to RAI 22.5-26

- Treatment of electric fire pump dedicated to low pressure injection needs to be clarified.
  - Awaiting GEH response to RAI 22.5-27

ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 19A (SER Chap. 22)

Open Items

Availability Controls (AC)

- ACs did not state the associated instrumentation functions and the number of required divisions in the AC LCOs for some functions
  - Awaiting GEH response to RAI 22.5-22

- AC bases do not explicitly state the minimum level of system degradation that corresponds to a function being unavailable, or the number of divisions used to determine the test interval for each required division (or component) for AC surveillance requirements
  - Awaiting GEH response to RAI 22.5-22

- No AC Surveillance Requirements provided for FAPCS pumps
  - Awaiting GEH response to RAI 22.5-23

- AC LCOs for FAPCS and EDGs inconsistent with PRA assumptions
  - Awaiting GEH response to RAI 22.5-24

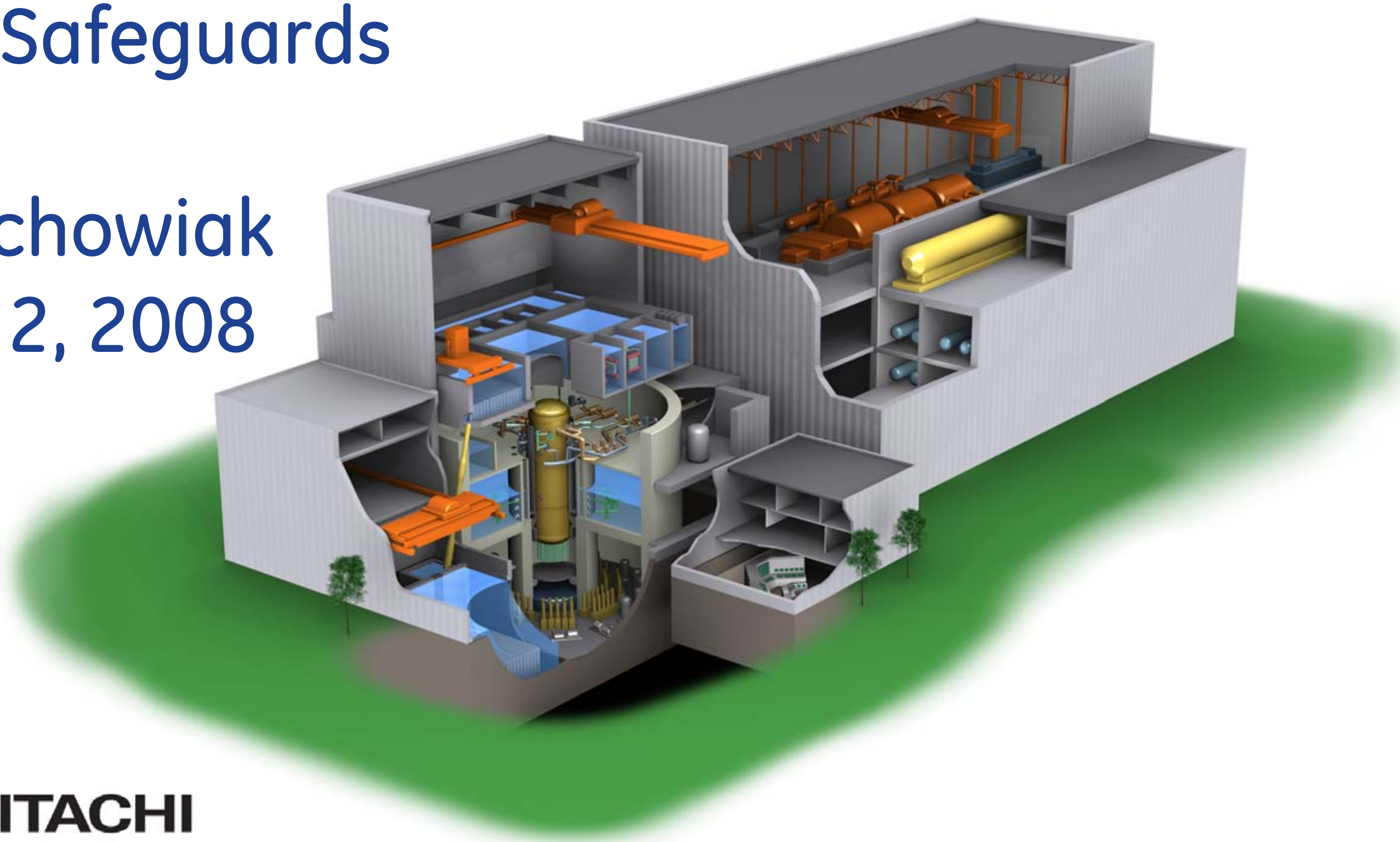
ACRS Full Committee Presentation
ESBWR Design Certification Review
Chapter 19A (SER Chap. 22)

Discussion / Questions

ESBWR PRA and Severe Accidents

Presented to the
Advisory Committee on
Reactor Safeguards

Rick Wachowiak
October 2, 2008



GE | HITACHI

Design Certification PRA Objectives

10 CFR 50.34(f)(1)(i) requires a Design Certification PRA to address known design issues with respect to core and containment heat removal systems

Identify vulnerabilities

Demonstrate that the plant meets the Commission's safety goals

Reduce/eliminate risk contributors in existing plants

Select among SAM design features

Identify risk-informed safety insights

Show a balance of severe accident prevention and mitigation

Show a reduction in risk in comparison to existing plants

Support design programs such as RTNSS and D-RAP

HITACHI

Interaction With NRC Staff On ESBWR PRA

Nearly 450 RAIs (almost 8% of total for certification)

- 386 resolved

Three on-site audits

Several meetings and teleconferences

Audit of revision 4 PRA expected in the first week of December

Focused on the design certification PRA objectives

HITACHI

Design Certification Not the Last ESBWR PRA

Revised PRA required by 10 CFR 50.71(h)(1)

- Level 1 and Level 2
- Prior to initial fuel load
- Must meet all endorsed standards

No intention that the DC PRA must satisfy this requirement

Maintained by the licensee for NRC inspection

Need for submittal to NRC based on each specific risk informed application requirements

HITACHI

Ongoing PRA Upgrade Requirements

10 CFR 50.71(h)(2) requires PRA maintenance or upgrade as new standards are endorsed

- 4 year periodicity
- PRA maintenance and PRA upgrade consistent with definition in ASME "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications"

ESBWR Design Certification PRA

Meets the scope and quality for certification

Meets the scope and quality for COL given no significant departures from the certified design

Provides a starting point for operating plant PRA

Organization of ESBWR PRA Reports

DCD Chapter 19 describes the PRA and lists key insights

NEDO 33201 ESBWR Certification Probabilistic Risk Assessment, R3 May 2008

NEDO 33289 ESBWR Reliability Assurance Program, R2 September 2008

NEDO 33306 ESBWR Severe Accident Mitigation Design Alternatives, R1 August 2007

NEDO/NEDE 33386 ESBWR Plant Flood Zone Definition Drawings and Other PRA Supporting Information, R0 September 2007

NEDO/NEDE 33392(P) The MAC Experiments: Fine Tuning of the BiMAC Design, R0 March 2008

NEDO 33411 Risk Significance of Structures, Systems, and Components for the Design Phase of the ESBWR, R0 March 2008

HITACHI

Key Features of ESBWR Design Risk Management

Passive safety systems

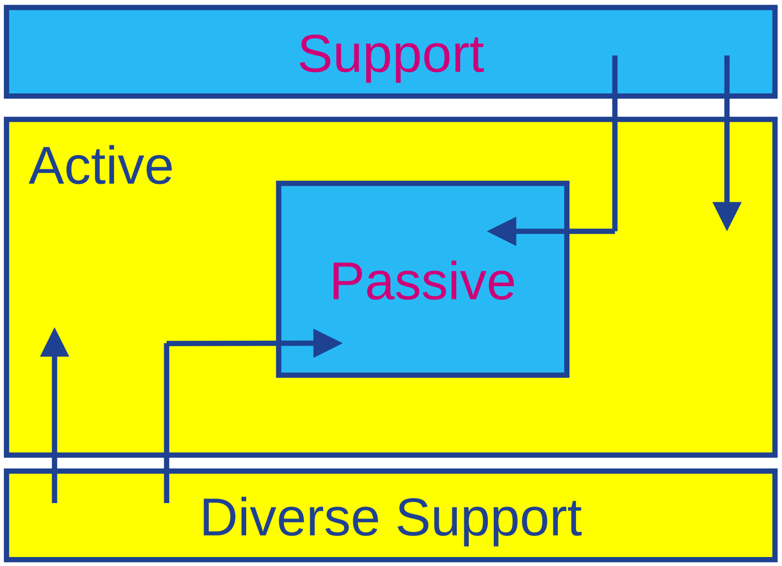
Active asset protection systems

Support system diversity

Minimize reliance on human actions

Use applicable historical data

Target configuration for core damage prevention functions



Features of ESBWR PRA

- Detailed Fault Tree / Event Tree Models
- Level 1, 2, and 3
- Internal & External Events
- All Modes
- Seismic Margins
- Generic Data
- Historical Initiating Event Frequencies
- Parametric Uncertainty
- Systematic Search for Key Modeling Uncertainties
- Internal review for compliance with ASME-RA-Sb-2005

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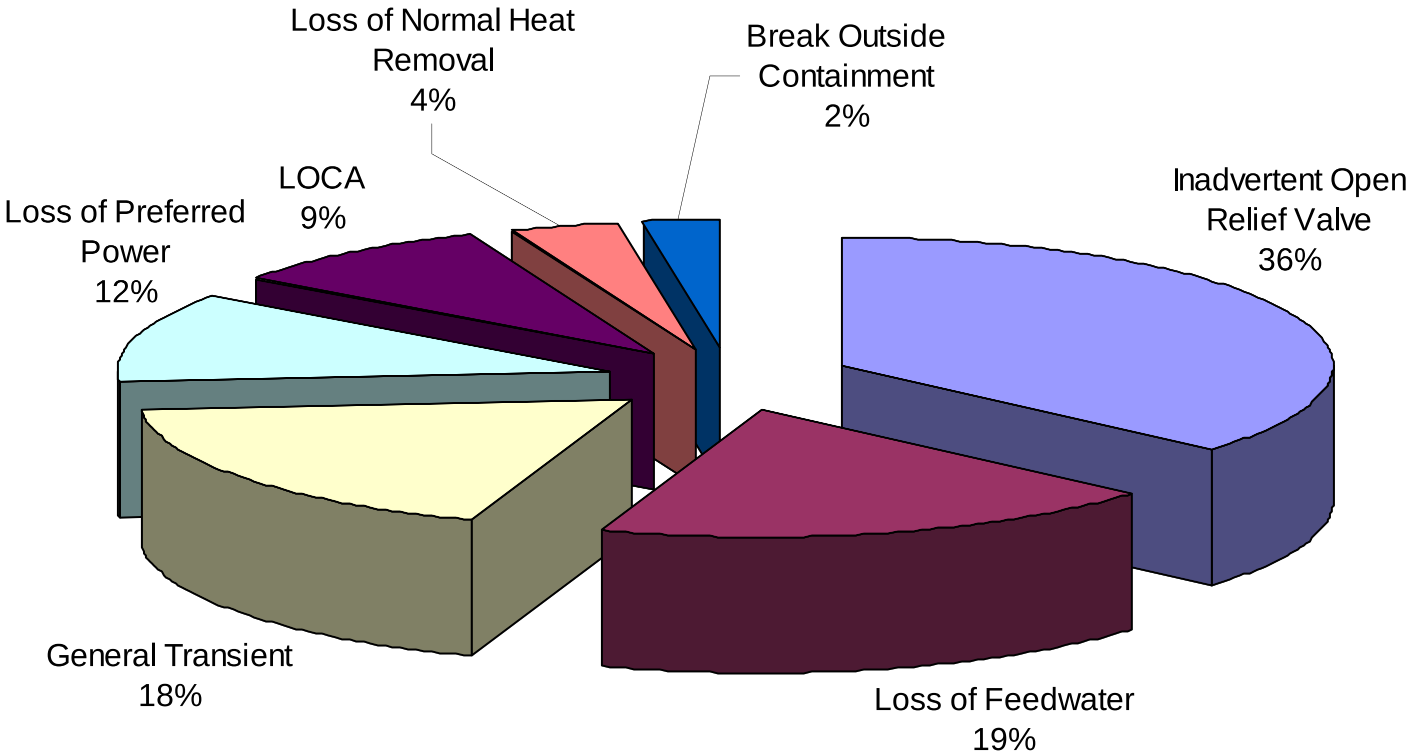
ESBWR Core Damage Risk Profile



$$CDF_{pe} = 1.2x10^{-8}\ /yr$$

At power internal events

HITACHI

Overall Results

	Internal Events	Fire	Flood	High Winds
At-Power CDF	1.2×10^{-8}	8.1×10^{-9}	1.6×10^{-9}	1.3×10^{-9}
Shutdown CDF	9.4×10^{-9}	2.7×10^{-8}	5.2×10^{-9}	1.2×10^{-9}
At-Power LRF	1.0×10^{-9}	5×10^{-10}	2×10^{-10}	3×10^{-11}
Shutdown LRF	9.4×10^{-9}	2.7×10^{-8}	5.2×10^{-9}	1.2×10^{-9}

Point Estimate Values
Units are per calendar year

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Scope of Severe Accident Analyses

Discussion of severe accident prevention

- Examples: ATWS, SBO, Fire Protection & ISLOCA

Discussion of severe accident mitigation

- Examples: Hydrogen control, debris coolability, high-pressure melt eject, containment performance, containment vent, equipment survivability

Severe accident mitigation design alternatives

Contained in DCD Ch 19, NEDO-33201 Ch 21, and NEDO-33306

PRA Was a Major Influence on Design

Examples

- Design of digital / mechanical interface to eliminate spurious actuations from fire
- Selection of diverse components
- Addition of redundancy to RWCU isolation features
- Addition of BiMAC to preclude containment failure
- Main control room design
- Addition of severe accident water injection pump
- More enhancements identified to resolve during procedure development

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NRC Staff Review Helped Enhance PRA

Examples

- Extend Level 3 to external events
- Enhanced documentation of assumptions
- Upgrade from FIVE to Fire PRA
- Systematic evaluation of the PRA with respect to endorsed standards

Limited Open Items Remain

PRA quality assessment

- GEH responded and it is under staff review
- Audit of ESBWR PRA scheduled for December

Seismic margins analysis

- Selection of response spectrum
- GEH response is in development

High winds analysis

- Assumptions for building capabilities in extreme wind events
- GEH response is in development

Shutdown event details

- GEH responded to 2 issues / in development for 2 issues

Severe accident resolution

- Questions from BiMAC test report
- GEH responses are in development

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NRC RTNSS Criteria

A SSC functions relied upon to meet beyond design basis deterministic NRC performance requirements such as 10CFR50.62 for anticipated transient without scram (ATWS) mitigation and 10CFR50.63 for station blackout

B SSC functions relied upon to resolve long-term safety (beyond 72 hours) and to address seismic events

C SSC functions relied upon under power-operating and shutdown conditions to meet the Commission's safety goal guidelines of a core damage frequency of less than 1.0E-4 each reactor year and large release frequency of less than 1.0E-6 each reactor year

D SSC functions needed to meet the containment performance goal (SECY-93-087, Issue I.J), including containment bypass (SECY-93-087, Issue II.G), during severe accidents

E SSC functions relied upon to prevent significant adverse systems interactions

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RTNSS Design Treatment

Redundant active components

Fire and flood protected

Hurricane category 5 missile protection

Designed for accident environment

Quality suppliers (not Appendix B)

Seismic category II for post-72 hr functions

Technical Specifications for SSCs Needed to Meet CDF and LRF Goals

Availability Controls Manual for Frontline Systems

HITACHI

RTNSS Open Items

Availability Controls

- ACs did not state the associated instrumentation functions and the number of required divisions in the AC LCOs for some functions
- AC bases do not explicitly state the minimum level of system degradation that corresponds to a function being unavailable, or the number of divisions used to determine the test interval for each required division (or component) for AC surveillance requirements
- No AC Surveillance Requirements provided for FAPCS pumps
- AC LCOs for FAPCS and EDGs inconsistent with PRA assumptions

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RTNSS Open Items

Design standards for post-72 hour functions

- Resolved

Augmented design standards for flood protection

- Existing RAIs resolved

RTNSS status of some active systems

- Responses in development

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Conclusions

ESBWR PRA and Severe Accident chapters meet the requirements for certification

Limited open items to be resolved

NRC review confirms that the required objectives will be satisfied in the DCD

HITACHI

Historical Perspectives and Insights on Reactor Accident Consequences Analyses

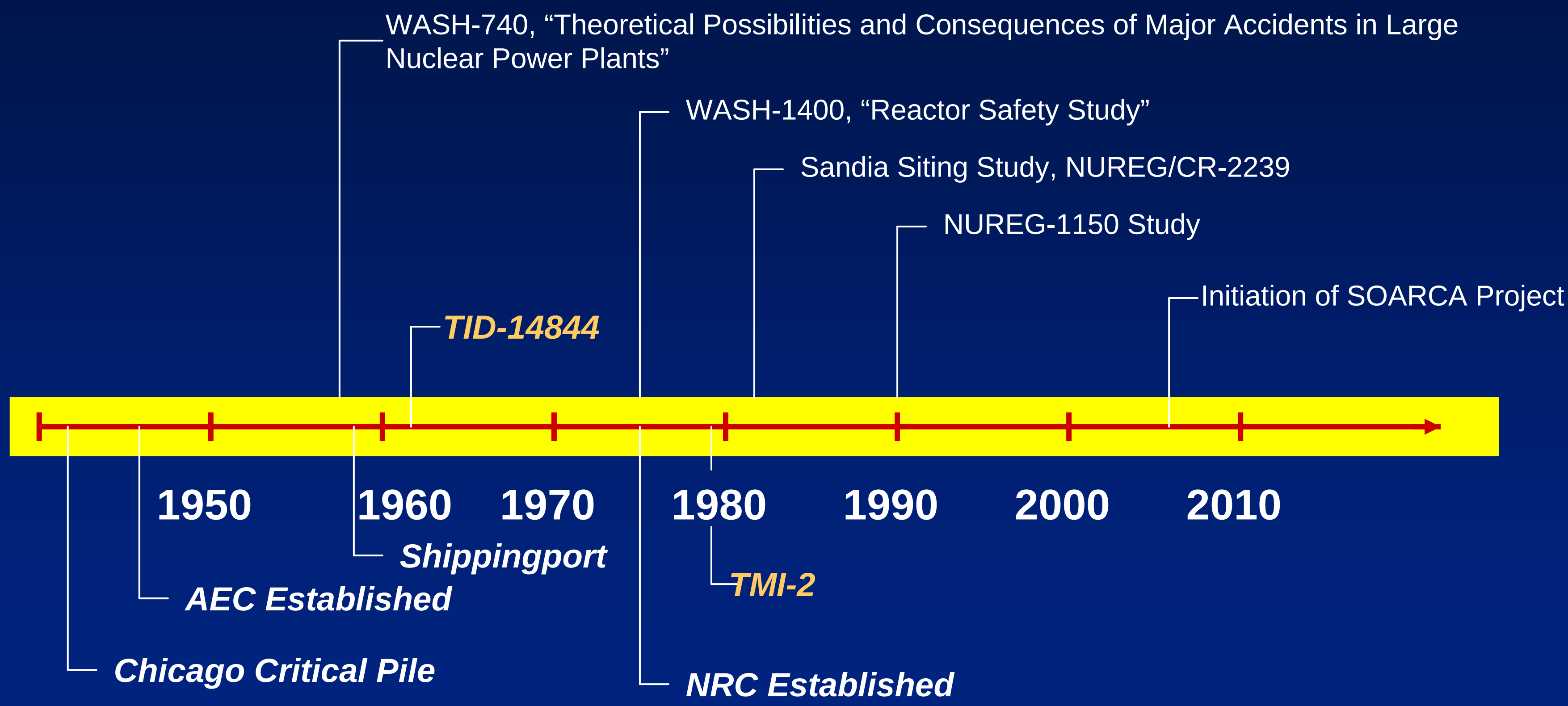
Hossein Nourbakhsh
Senior Technical Advisor
Advisory Committee on Reactor Safeguards (ACRS)

Presented at
556th Meeting of ACRS
October 2, 2008

Objectives

- To provide historical perspectives and insights on previous state-of-the-art analyses of the consequences of severe reactor accidents
- To discuss the feasibility of using a simplified, yet systematic and defensible, approach to benchmark many aspects of SOARCA

Timeline of Major Studies of Reactor Accident Consequences



WASH-740

- The first estimates of consequences of severe accidents were published in the 1957 U.S. Atomic Energy Commission report (WASH-740), “Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants”
- An attempt to provide upper bounds of the potential public hazards resulting from certain severe hypothetical accidents
- Conservative values were used for many factors influencing the magnitude of the estimated accident consequences
- At the time, the technology and the state-of-knowledge of severe accidents had not progressed to the point where it was possible to use quantitative techniques to estimate the probabilities of such accidents. However, there was a general agreement that the probability of occurrence of severe accidents in nuclear power reactors was exceedingly low.

Reactor Safety Study (WASH-1400)

- The first systematic attempt to provide realistic estimates of public risk from potential accidents in commercial nuclear power plants
- Included analytical methods for determining both the probabilities and consequences of various accident scenarios
- Two specific reactor designs were analyzed in WASH-1400, Surry and Peach Bottom
- Calculations were performed for a number of accident sequences and the results for these calculations were used to define a series of release categories (nine for PWR and five for BWR) into which all of the identified accident sequences could be placed.

Post TMI-2 Review of Source Term Technical Basis

- Following the publication of WASH-1400 and the accident at TMI-2, work initiated to review the predictive methods for calculating fission product release and transport
- Review resulted in several conclusions that represented significant departure from WASH-1400 assumptions including the suggestion that cesium iodide (CsI) will be the expected predominant iodine chemical form under most postulated LWR accident conditions
- These studies formed the basis for development of a generic set of radiological releases, characterized as Siting Source Terms (denoted SST1-5), used in Sandia Siting Study (NUREG/CR-2239)

Brief Descriptions of the Characteristics of the Accident Groups

(NUREG-0771, p. 8)

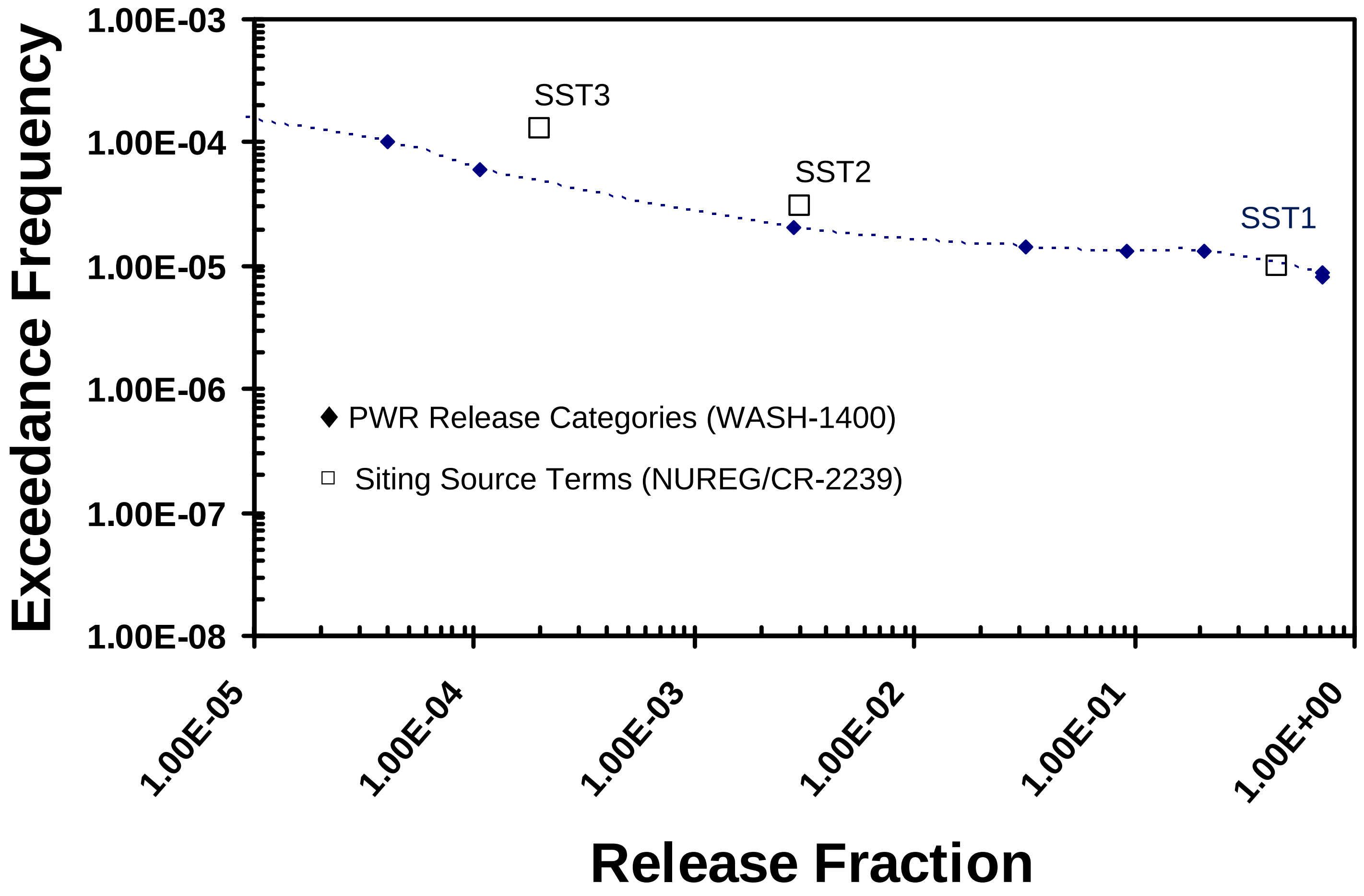
Group 5 -	Limited core damage. No failures of engineered safety features beyond those postulated by the various design basis accidents are assumed. The most severe accident in this group includes substantial core melt, but containment functions as designed (siting DBA equivalent).
Group 4 -	Limited to modest core damage. Containment systems operate but in somewhat degraded mode (TMI-2 equivalent)
Group 3 -	Severe core damage. Containment fails by basemat melt-through. All other release mitigation systems have functioned as designed (analogous to Reactor Safety Study Pressurized Water Reactor, PWR, Categories 6 and 7)
Group 2 -	Severe core damage. Containment fails to isolate. Fission product release mitigating systems (e.g., sprays, suppression pool, fan coolers) operate to reduce release (analagous to Reactor Safety Study PWR Categories 4 and 5)
Group 1 -	Severe core damage. Essentially involves loss of all installed safety features. Severe direct breach of containment (analogous to Reactor Safety Study PWR Categories 1 and 3)

Sandia Siting Study (NURG/CR-2239)

- Used Siting Source Terms (SSTs) at 91 existing or proposed reactor sites to perform accident consequence analyses
- Detailed PRAs were not performed for all reactors. Based on available PRAs at the time, NRC suggested the following representative probabilities for the SSTs
  - SST1 1×10^{-5}
  - SST2 2×10^{-5}
  - SST3 1×10^{-4}

Frequecy of Release for Iodine

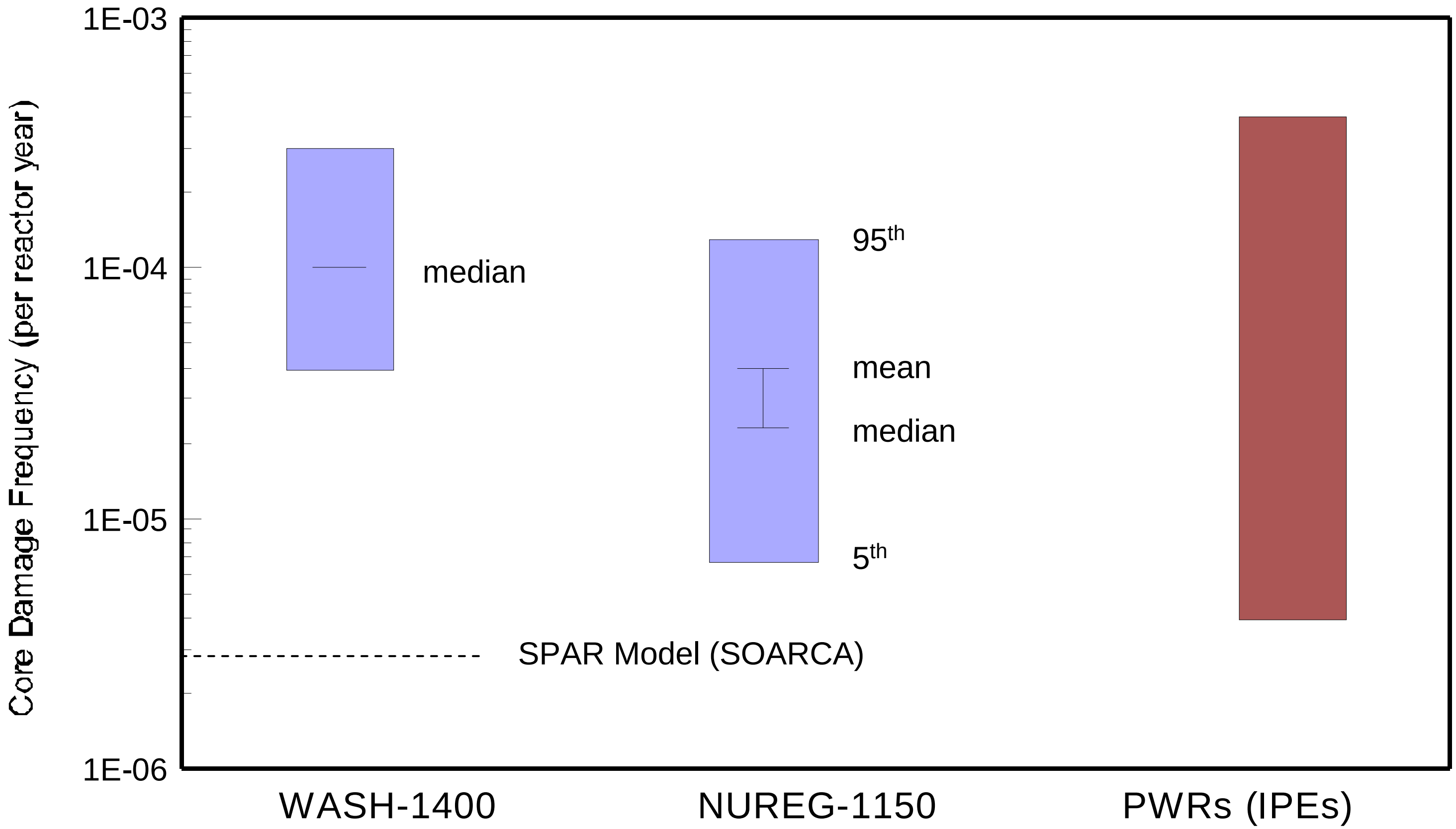
(Comparison of WASH-1400 PWR Release Categories and SSTs)



NUREG-1150 Study

- The NUREG-1150 study was a major effort to put into a risk perspective the insights into system behavior and phenomenological aspects of severe accidents
- An important characteristic of this study was the inclusion of the uncertainties in the calculations of core damage frequency and risk that exist because of incomplete understanding of reactor systems and severe accident phenomena
- The elicitation of expert judgment was used to develop probability distributions for many accident progression, containment loading, structural response, and source term issues
- Five specific commercial nuclear power plants were analyzed :
  - Surry, a 3-loop Westinghouse PWR with a subatmospheric containment
  - Zion, a 4-Loop Westinghouse PWR with large dry containment
  - Sequoyah, a 4-loop Westinghouse PWR with ice-condenser containment
  - Peach Bottom, a BWR-4 reactor with a Mark I containment
  - Grand Gulf, a BWR-6 reactor with a Mark III containment

Internal Core Damage Frequency for Surry



Conditional Probability of Accident Progression Bins at Surry

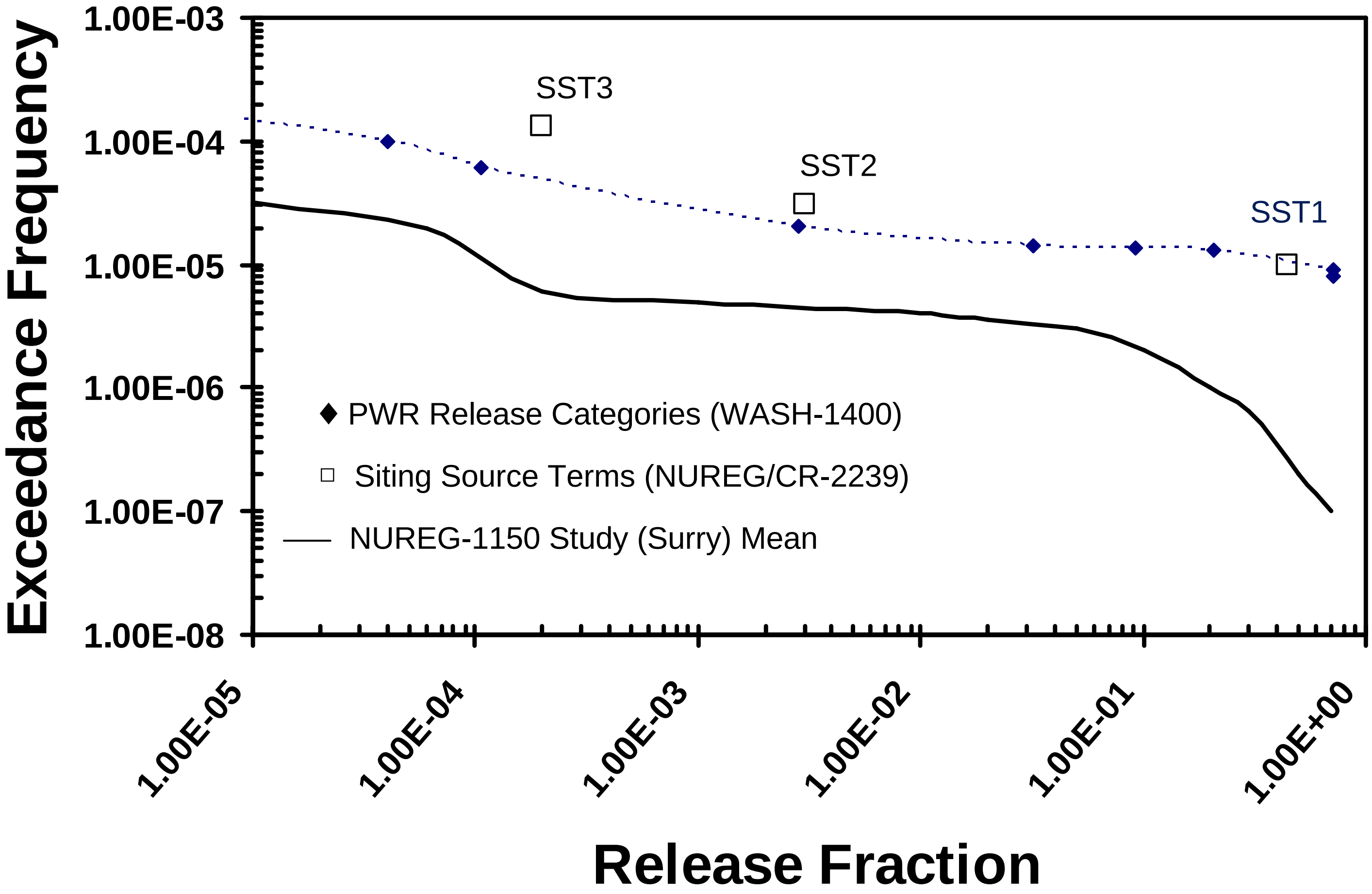
(NUREG-1150, p. 3-12)

SUMMARY ACCIDENT PROGRESSION BIN GROUP	SUMMARY PDS GROUP (Mean Core Damage Frequency)							
	Internal Initiators						Fire	Seismic
	LOSP (2.8E-05)	ATWS (1.4E-06)	Transients (1.8E-06)	LOCAs (6.1E-06)	Bypass (3.4E-06)	All (4.1E-05)	(1.1E-05)	LLNL (1.9E-04)
VB, alpha, early CF	0.003	0.003		0.005		0.003	0.005	0.006
VB > 200 psi, early CF	0.005		0.001	0.001		0.004	0.013	0.008
VB, < 200 psi, early CF								0.082
VB, BMT or late CL	0.079	0.046	0.013	0.055		0.059	0.292	0.280
Bypass	0.003	0.078	0.007		1.000	0.122		0.001
VB, No CF	0.310	0.523	0.217	0.586		0.346	0.690	0.435
No VB	0.599	0.350	0.762	0.352		0.466		0.189

Key: BMT = Basemat Melt-Through
CF = Containment Failure
CL = Containment Leak
VB = Vessel Breach

Frequecy of Release for Iodine Group

(Comparison of WASH-1400 PWR Release Categories, SSTs, and NUREG-1150)



Reassessment of Selected Factors Affecting Siting of Nuclear Power Plants (NUREG/CR-6295)

- A series of probabilistic consequence assessment calculations were performed in support of an effort to re-assess reactor siting
- Insights from NUREG-1150 and the LaSalle independent risk assessment studies were used to develop representative source terms
  - A small set of source terms (4 to 7 for each plant) based on dominant plant damage states, accident progression groups and the associated release characteristics were developed for each reactor design to represent the full spectrum of severe accidents
- Examined consequences in a risk based format consistent with the quantitative health objectives (QHOs) of the NRC’s Safety Goal Policy

Characteristics of Surry Release Categories, Internal Events

(NUREG/CR-6295, pp 3-19)

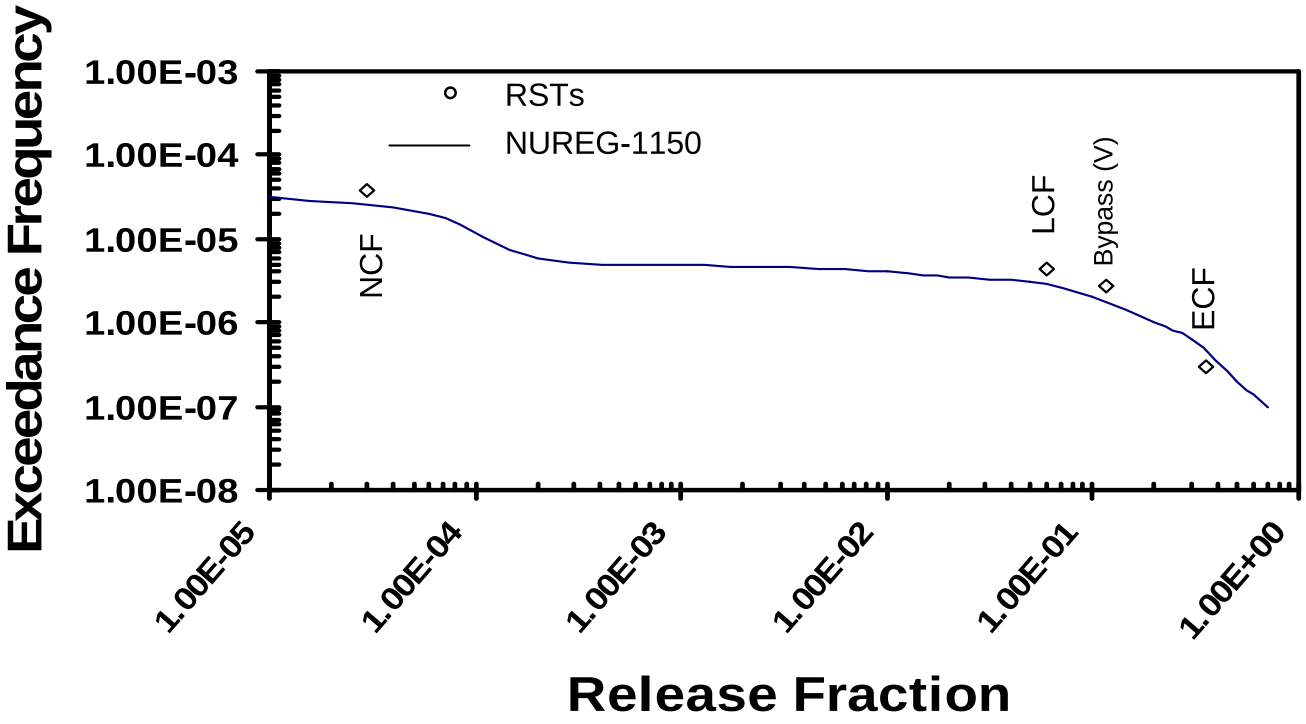
Release Category	Plant Damage State	Accident Progression Characteristics						
		Containment Failure Time	Containment Failure Mode	CCI	Amt CCI	RCS Pres.	VB Mode	Sprays
RSUR1	LOSP	CF at VB	Rupture	Prm Dry	Medium	Low	Alpha	No
RSUR2	LOSP	Late CF	Leak	Prm Dry	Large	Low	Pour	No
RSUR3	LOSP	No CF	No CF	Prm Dry	Large	Low	Pour	L+VL
RSUR4	Bypass (V)	No CF	Bypass	Prm Dry	Large	Low	Pour	No

Radionuclide Release Characteristics into Environment for Surry, Internal Events

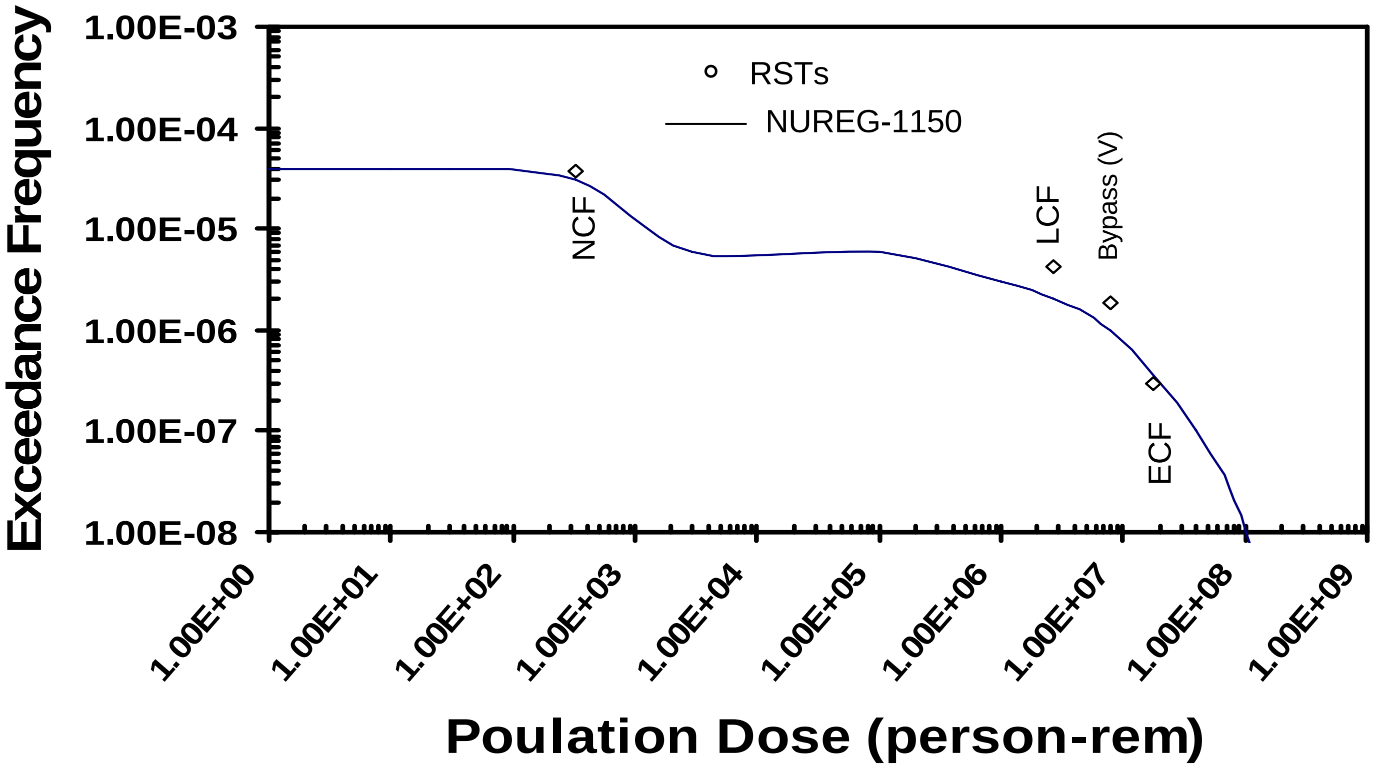
(NUREG/CR-6295, pp3-19)

Release Category	Frequency	Elevation (m)	Energy (W)	Time of Core Uncovery	Time of Release (hrs)	Release Duration	Fractional Releases								
							Ng	I	Cs	Te	Sr	Ba	Ru	La	Ce
RSUR1	2.9E-7	10	2.8E+7	5.0 hrs	6.0	200 sec	1.0E+0	2.5E-1	1.8E-1	8.0E-2	2.0E-2	2.0E-2	5.0E-3	1.0E-3	5.0E-3
		10	1.6E+6		6.06	2 hrs	0.0E+0	1.0E-1	1.3E-1	1.0E-1	4.0E-2	4.0E-2	1.0E-3	5.0E-3	5.0E-3
RSUR2	2.4E-6	10	5.2E+5	5.0 hrs	12.0	3 hrs	1.0E+0	6.0E-2	3.0E-2	9.0E-2	3.0E-3	3.0E-3	1.0E-3	4.0E-4	4.0E-4
RSUR3	3.3E-5	0	0.0E+0	5.0 hrs	6.0	10 hrs	2.5E-3	1.5E-5	1.2E-8	7.5E-9	2.5E-9	2.5E-10	2.0E-10	3.0E-10	4.0E-10
		0	0.0E+0		16.0	10 hrs	2.5E-3	1.5E-5	1.2E-8	7.5E-9	2.5E-9	2.5E-10	2.0E-10	3.0E-10	4.0E-10
RSUR4	1.6E-6	0	1.9E+6	20 min	1.0	30 min	1.0E+0	7.5E-2	6.0E-2	2.0E-2	5.0E-3	5.0E-3	1.0E-3	3.0E-4	1.0E-3
		0	1.7E+5		1.5	2 hrs	0.0E+0	4.0E-2	6.0E-2	6.0E-2	2.0E-2	2.0E-2	6.0E-4	3.0E-3	3.0E-3

Frequency of Release for Iodine Based on Representative Source Terms (RSTs) for Surry Internal Events



Frequency of Population Dose to Entire Region at Surry



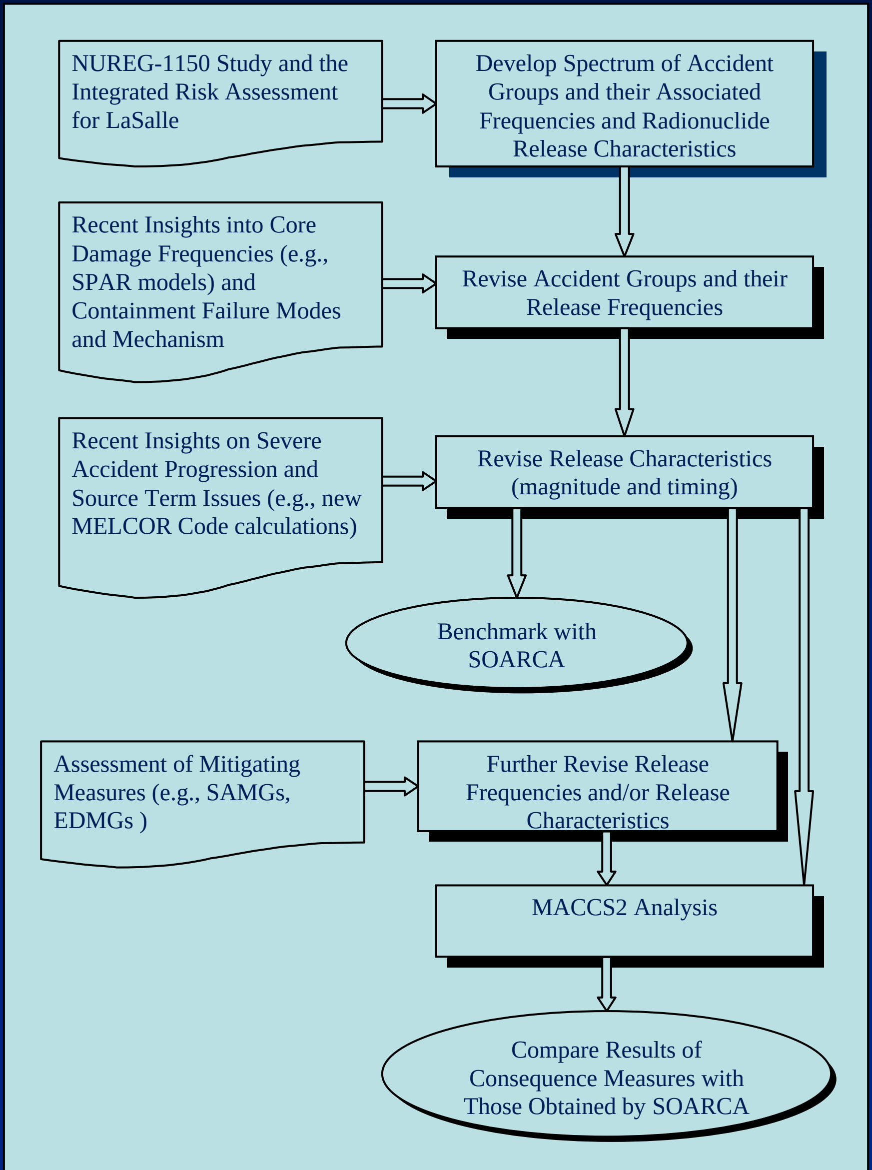
Recent Advances in Understanding of Severe Accident Phenomenology and Containment Failure Mechanisms

- Since the completion of NUREG-1150 Study, more analytical and experimental studies have been performed to address many severe accident issues including:
  – Direct Containment Heating (DCH) Issue
  – “Mark I Liner Attack” Issue
  – In-vessel steam explosion (alpha mode failure)

A SIMPLIFIED APPROACH TO SOARCA BENCHMARKING

- Although performing Level-3 PRAs for the pilot plants is the best way to benchmark the SOARCA methodology, results and insights from the NUREG-1150 Study and Integrated Risk Assessment for LaSalle, together with more recent advances in understanding of the severe accident issues and containment failure mechanisms, could be used for developing a simplified, yet systematic and defensible, approach to benchmark many aspects of SOARCA.

Elements of the Proposed Approach to Benchmark SOARCA



Impact of current knowledge and understanding of early containment failure on NUREG-1150 results for the conditional probability of accident progression bins at Surry

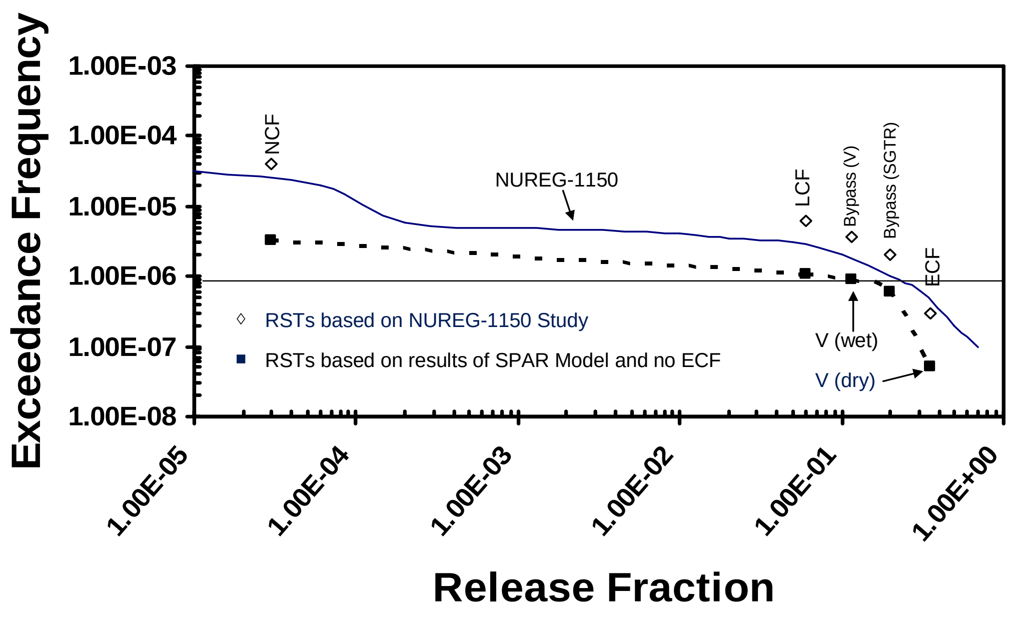
Summary Accident progression Bin Group	Summary PDS Group (Mean Core Damage Frequency)							
	Internal initiators (4.1E-05)						Fire (1.1E-05)	Seismic LLNL (1.9E-04)
	LOSP (2.8E-05)	ATWS (1.4E-06)	Transients (1.8E-06)	LOCAs (6.1E-06)	ISLOCA (1.6E-06)	SGTR (1.8E-06)		
Early CF	-- (0.008)[a]	-- (0.003)	-- (0.001)	-- (0.006)			-- (0.018)	0.082 (0.096)
Late CF	0.084 (0.079)	0.046 (0.046)	0.014 (0.013)	0.056 (0.055)			0.305 (0.292)	0.288 (0.280)
Bypass	(0.003)	(0.078)	(0.007)		(1.0)	(1.0)		(0.001)
No CF	0.913 (0.909)	0.876 (0.873)	0.979 (0.979)	0.944 (0.939)			0.695 (0.690)	0.630 (0.624)

(a) Numbers in parentheses are the results of the NUREG-1150 Study.

Frequencies and Magnitudes of Iodine Releases for Representative Source Terms for Surry (Internal Initiators)

Release Category	Summary PDS Group	Containment Failure Time	Containment Failure Mode	Frequency		Fractional Release for Iodine Group
				Based on NUREG-1150 Study	Revised Based on Results of SPAR Model and no Early Failure of Cont.	
RSUR1	LOSP	CF at VB (ECF)	Rupture	2.9E-07	----	0.35
RSUR2	LOSP	Late CF (LCF)	Leak	2.4E-06	1.5E-07	0.06
RSUR3	LOSP	No CF (NCF)	No CF	3.3E-05	1.95E-06	3.E-05
RSUR4	Bypass (V)	NCF	Bypass	1.6E-06 Wet (~85%) Dry (~15%)	3.5E-07 Wet (~3.0E-07) Dry (~5.0E-08)	0.115 0.115 (Wet) 0.37 (Dry)
RSUR5	Bypass (SGTRs)	NCF	Bypass	1.8E-06	5.5E-07	0.2

Comparison of frequency distribution (CCDF) of iodine release predicted by NUREG-1150 Study for Surry with that obtained from the results of SPAR model and the recent insights on early containment failure mechanisms (Internal Events)



Summary and Conclusion

- An overview of major contributions to consequence assessment was presented to provide historical perspectives and insights on previous state-of-the-art analyses of the consequences of severe reactor accidents
- It is feasible to use the results and insights from the NUREG-1150 Study and Integrated Risk Assessment for LaSalle, together with more recent advances in understanding of the severe accident issues and containment failure mechanisms, and develop a simplified, yet systematic and defensible, approach to benchmark many aspects of SOARCA