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

March 5, 2008

MEMORANDUM TO: Carol A. Brown, Technical Secretary
Advisory Committee on Reactor Safeguards

FROM: William J. Shack ***/RA/***
ACRS Chairman

SUBJECT: MINUTES OF THE 548th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), December 6-8, 2007

I certify that based on my review of the minutes from the 548th 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.

NA
Comments

ADAMS Accession: ML080640924

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NAME	JFlack		
DATE	03/05/08		

CERTIFIED

Date Issued: 03/05/08
Date Certified: 03/05/08

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December 6 - 8, 2007

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- Draft Final NUREG-1829, “Estimating Loss-Of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process,” and Draft NUREG-XXXX, “Seismic Considerations for the Transition Break Size,” dated December 20, 2007.

- Susquehanna Steam Electric Station Units 1 and 2 Extended Power Uprate Application, dated December 20, 2007.

LETTER

Letter to Luis A. Reyes, Executive Director for Operations, NRC, from William J. Shack, Chairman, ACRS:

- AREVA Detect and Suppress Stability Solution and Methodology, dated December 27, 2007.

MINUTES OF THE 548thMEETING OF THE
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
December 6 - 8, 2007
ROCKVILLE, MARYLAND

The **548th** meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on **December 6 - 8, 2007**. Notice of this meeting was published in the *Federal Register* on **November 20, 2007** (72 FR 65358) (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. Dennis Bley,Dr. Said Abdel-Khalik (Member-at-Large), Dr. George E. Apostolakis, Dr. Sam Armijo, Dr. Michael Corradini, Mr. Otto L. Maynard, Dr. Dana A. Powers, Mr. Jack Sieber, and Mr. John Stetkar. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

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

Dr. William J. Shack, Committee Chairman, convened the meeting at 8:30 A.M. He announced in his opening remarks that the meeting was being conducted in accordance with the provisions of the Federal Advisory Committee Act. In addition, he reviewed the agenda for the meeting 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. He discussed the items of current interest and administrative details for consideration by the full Committee.

II. Draft Final NUREG-1829, “Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process,” and Draft NUREG-XXXX, “Seismic Considerations for the Transition Break Size”

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

The Committee met with the representative of NRC staff to discuss draft final NUREG-1829 on estimating LOCA frequencies through the elicitation process, and draft NUREG-XXXX on seismic considerations for the transition break size (TBS).

The Commission has directed the staff to develop a risk-informed alternative to 10 CFR 50.46. An essential element of this effort is the selection of break size that has a mean frequency of occurrence of about 10^{-5} per reactor year.

These reports provide the basis for a conservative selection of this TBS.

Draft final NUREG-1829 presents the results of a formal expert elicitation process that was used to estimate generic boiling water reactor (BWR) and pressurized water reactor (PWR) passive-system LOCA frequencies associated with material degradation.

Draft NUREG-XXXX provides additional insights by investigating seismically induced failures in unflawed piping, flawed piping, and indirect piping failures caused by the failure of other components and supports. The results of the study indicate that, for PWRs, the likelihood of seismically induced failures in unflawed piping of size greater than the TBS is very low. Even for pipes with long surface flaws, the depths of these flaws must be very large for a high likelihood of failure during earthquakes. Inspection programs, leak detection systems, and other measures taken to eliminate failure mechanisms such as stress corrosion cracking would make the likelihood of such flaws very low.

Committee Action

The Committee issued a report to the NRC Chairman on this matter dated December 20, 2007, recommending that both NUREG reports be published. The Committee also recommended that regulatory decisions be based on the totality of the results from the sensitivity studies rather than the results from individual methods of expert judgment aggregation and that a set of consistent guidelines be established throughout the agency for the elicitation and aggregation of expert judgments including the performance of sensitivity studies.

III. AREVA Enhanced Option III Long Term Stability Solution

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

The Committee met with representatives of the NRC staff and AREVA to discuss the staff's draft Safety Evaluations (SEs) for Topical Reports BAW-10255P, Revision 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," and ANP-10262P, Revision 0, "Enhanced Option III Long Term Stability Solution." Representatives of AREVA presented an overview of the detect and suppress methodology described in these topical reports. Topical Report BAW-10255P describes a plant-specific Option III stability methodology using AREVA analytical methods and codes. The proposed plant-specific stability methodology resolves the technical deficiencies associated with the application of the generic Option III DIVOM methodology to certain core thermal-hydraulic conditions and power densities. The proposed plant-specific DIVOM calculation methodology relies on the AREVA RAMONA5-FA 3D code. In ANP-10262P AREVA proposed extension of the plant-specific DIVOM methodology to operation at an expanded operating domain in which the power densities and power-to-flow ratios increase. Operation at the expanded operating domain is expected to decrease the stability of the reactor. Therefore, the Enhanced Option III method introduces additional design features to ensure that General Design Criteria – 12 (GDC-12) requirements are met. GDC-12 requires that the core be designed such that instability is not possible or the instability is detected and suppressed. The staff summarized the results of its evaluation of these topical reports.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter dated December 27, 2007, concluding that the AREVA detect and suppress methodology is acceptable, subject to certain limitations and conditions. The Committee recommended that additional conditions be imposed to address issues regarding the extent and depth of the staff's review of the RAMONA5-FA code, the need for further documentation of the technical bases for the margins added to some of the key instability detect and suppress parameters, and the need for additional assessment of the validation of the RAMONA5-FA calculation based on the steady state dryout correlation.

IV. State-of-the-Art Reactor Consequence Analysis (SOARCA)

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

The Committee met with representatives of the NRC staff and the Union of Concerned Scientists (UCS) to discuss the status of staff's efforts associated with the State-of-the-Art Reactor Consequence Analysis (SOARCA) Project. The staff is initially focusing on two sites, Peach Bottom in Pennsylvania, and Surry in Virginia. The staff presented its initial findings of the accident sequence selection, preliminary MELCOR insights, containment performance, and emergency preparedness for these two plants. The staff also presented the various options that it is evaluating for assessment of dose thresholds for latent cancer fatalities. A representative from UCS stated that the UCS is supportive of an authoritative and independent study that improves the technical credibility and accuracy of analyses of the consequences of severe accidents but is concerned that the SOARCA Project does not appear to be on track to fulfill such a role.

Committee Action

The Committee plans to consider a report on SOARCA during its February 2008 meeting.

V. Draft ACRS Report on the NRC Safety Research Program

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

The ACRS provides the Commission a biennial report, presenting the Committee's observations and recommendations concerning the overall NRC Safety Research Program. During the December 2007 meeting, the Committee discussed its draft 2008 report to the Commission on the NRC Safety Research Program. The committee also discussed the scope of long-term research the agency needs to consider.

Committee Action

The Committee plans to continue its discussion of the draft ACRS report on the NRC Safety Research Program during its February 2008 meeting.

VI. Extended Power Uprate Application for the Susquehanna Power Plant

[Note: Ms. Zena Abdullahi was the Designated Federal Official for this portion of the meeting.] The Committee met with representatives of the NRC staff, Pennsylvania Power and Light (PPL or "the licensee"), and its consultant (AREVA) to discuss the extended power uprate (EPU) application for the Susquehanna Steam Electric Station (SSES) and the associated NRC staff's Safety Evaluation. The PPL application requested that operation of SSES Units 1 and 2 be increased to 3952 MWt, which corresponds to a 20 percent increase from the originally licensed thermal power.

The discussions focused on Member concerns regarding the applicability of AREVA analytical methods and codes.

A series of codes based on different void fraction correlations were used to determine the operating limits. Members expressed concern that the measured uncertainties and biases in these correlations were not propagated through the codes to determine their impact on the operating limits. The Members also noted the lack of measured data at higher void fraction. To address these concerns, the licensee and AREVA described the propagation of void fraction uncertainty by replacing the void fraction correlation used in the neutronics method with another correlation. The Members found that this replacement of correlations did not account for the appropriate measurement uncertainty and the uncertainty associated with each code was not propagated.

The Members also expressed concern regarding the potential for pellet-cladding-interaction (PCI) failures since SSES uses conventional nonbarrier fuel. The revised SE did not address PCI failures during slow transients considering the flatter EPU core designs and the associated changes in the KW/ft. The staff noted that PCI failures are not considered as part of the regulatory process.

Members raised issues associated with the adequacy and applicability of the database benchmarking the power distribution uncertainties applied to the safety limit calculations. The revised staff SE increased the power distribution uncertainties to account for the limited validation data and the applicability of the available data.

Members were also concerned with the impact of bypass voiding on the neutron monitoring readings during transient events such as a recirculation pump trip that would result in reduced core flow and higher in-channel and bypass voiding.

Committee Action

The Committee issued a report to the NRC Chairman on this matter dated December 20, 2007, recommending that the SSES Units 1 and 2 EPU application be approved subject to the condition that an appropriate margin be added to the operating limit minimum critical power ratio as an interim measure to account for uncertainties in the void fraction correlation and the lack of data for its validation at void fractions above 90 percent. The Committee also recommended that the staff perform a thorough review and assessment of the risk of PCI fuel failures with conventional fuel cladding during anticipated operational occurrences and that Review Standard 001 be improved to include cross referencing of related sections between the power uprate safety analysis report and the staff's SEs.

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VII. Subcommittee Report on ESBWR

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

The Chairman of the ESBWR Subcommittee provided a report to the Committee summarizing the results of the November 15, 2007, meeting with the NRC staff and GE-Hitachi to review selected chapters of the staff's Safety Evaluation Report (SER) with Open Items associated with the ESBWR design certification. This meeting focused on Chapter 9 (Auxiliary Systems), Chapter 10 (Steam and Power conversion Systems), Chapter 13 (Conduct of Operations), and Chapter 16 (Technical Specifications).

VIII. Election of ACRS Officers for CY 2008

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

The Committee elected William J. Shack as ACRS Chairman, Mario V. Bonaca as ACRS Vice Chairman, and Said Abdel-Khalik as Member-at-Large for the Planning and Procedures Subcommittee for CY 2008.

IX. Executive Session

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

A. RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's response of November 1, 2007, to comments and recommendations included in the September 26, 2007, ACRS report on the development of a technology-neutral regulatory framework. The Committee plans to continue discussions with the staff on this matter during future ACRS meetings.

- The Committee considered the EDO's response of November 23, 2007, to comments and recommendations included in the October 19, 2007, ACRS letter on the NRC staff's safety assessment of the industry study related to dissimilar metal weld issues in pressurizer nozzles. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of November 21, 2007, to comments and recommendations included in the October 16, 2007, ACRS report on the NRC staff's Digital Instrumentation and Control (I&C) Systems Project Plan and Interim Staff Guidance. The Committee decided that it was satisfied with the EDO's response.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from November 4, 2007, through December 5, 2007, the following Subcommittee meetings were held:

- Thermal-Hydraulic Phenomena - November 14, 2007

The Subcommittee reviewed the staff's draft safety evaluations associated with topical reports BAW-10255P, Revision 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," and ANP-10262P, Revision 0, "Enhanced Option III Long Term Stability Solution."

- Power Uprates - November 14, 2007

The Subcommittee reviewed the application by Pennsylvania Power and Light (PPL) for an extended power uprate for SSES Units 1 and 2 and the associated staff's safety evaluation.

- ESBWR — November 15, 2007

The Subcommittee discussed several SER Chapters with open items associated with the ESBWR design certification application.

- Regulatory Policies and Practices — November 16, 2007

The Subcommittee discussed the status of staff's efforts associated with the State-of-the-Art Reactor Consequence Analysis (SOARCA) Project.

- Reliability & Probabilistic Risk Assessment – November 27, 2007

The Subcommittee discussed the Draft Final NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," and Draft NUREG-XXXX, "Seismic Considerations for the Transition Break Size."

- Planning and Procedures — December 5, 2007

The Subcommittee discussed proposed ACRS activities, practices, and procedures for conducting Committee business and organizational and personnel matters relating to ACRS and its staff.

- ABWR — December 5, 2007

The Subcommittee discussed the ABWR design and the South Texas Project Combined License Application.

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The Committee plans to continue its discussion on SOARCA during its February 2008 meeting.

- The Committee plans to continue its discussion of its draft 2008 report to the

Commission on the NRC Safety Research Program during its February 2008 meeting.

- The Committee would like the opportunity to review the applicability of the AREVA methodology for operation at the newly proposed General Electric Hitachi expanded domain (MELLLA+), before application to plant-specific submittal.

- The Committee would like the opportunity to review the staff's assessment of the RAMONA5-FA code.

- The Committee plans to continue its review of the staff's SER with Open Items associated with ESBWR design certification during a future meeting.

- The Committee plans to continue its review of the South Texas Project Combined License Application during a future meeting.

PROPOSED SCHEDULE FOR THE 549th ACRS MEETING

The Committee agreed to consider the following topics during the 549th ACRS meeting, to be held on February 7-9, 2008:

- Final Review of the License Renewal Application for the Vermont Yankee Nuclear Power Station

- Draft Final Revision 1 to Regulatory Guide 1.45 (DG-1173), "Guidance on Monitoring and Responding to Reactor Coolant System Leakage"

- Proposed Licensing Strategy for the Next Generation Nuclear Plant (NGNP)

- Cable Response to Live Fire (CAROLFIRE) Testing and Fire Model Improvement Program

- Proposed BWR Owners Group (BWROG) Topical Report on Methodology for Calculating Available Net Positive Suction Head (NPSH) for ECCS Pumps

- Draft ACRS Report on the NRC Safety Research Program

- State-of-the-Art Reactor Consequence Analysis (SOARCA) Program

B. Report on the Meeting of the Planning and Procedures Subcommittee Held on December 5, 2007

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

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

Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through March 2008 is attached. The objectives are to:

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

Staff Requirements Memorandum – Evaluation of the Overall Effectiveness of the Rulemaking process improvement Implementation Plan

In a Staff Requirements Memorandum dated October 25, 2007, the Commission states proposed rule packages should be provided to the ACRS for comment, and that the ACRS will be briefed on proposed rules only as a result of an ACRS request. For the draft final rule, the ACRS should continue its practice of reviewing the rule package prior to its submittal for Commission review and approval.

This is somewhat similar to the existing process. The members are reminded that if they want a briefing on a proposed rule, they should ask the staff to do so.

Revised Subcommittee Structure

A proposed revision to the Subcommittee Structure was sent to the members and the ACRS staff on November 8, 2007 for review and comment. This new structure was discussed.

This revision involves:

- Assignments to Dennis Bley and John Stetkar. **[Note: John Stetkar will Chair the Reliability and PRA Subcommittee when reviewing the EPR PRA, and Dennis Bley will Chair the Subcommittee when reviewing the US-APWR PRA.]**
- Minor changes to some members assignments.
- Abolishment of completed tasks and addition of new tasks.
- Changes to the staff assignments.

The revised Subcommittee Structure will become effective on December 10, 2007.

Election of Officers for CY 2008

During its December meeting, the Committee will elect Chairman and Vice Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee. Section 8.4 of the ACRS Bylaws state "A member may withdraw his name from consideration by written notice to the Executive Director, no later than two weeks before the scheduled election." Accordingly, during the November ACRS meeting, we requested that those members who do not wish to be considered for all or any of the

Offices should notify the ACRS Executive Director in writing by November 23, 2007. So far, two members have withdrawn their names.

Christmas Party

The Christmas party, sponsored by the members, is scheduled to be held between 12:00-1:30 p.m. on Friday, December 7, 2007.

Japanese Earthquake Effects

In an e-mail dated November 28, 2007, Dr. Powers states that the advanced LWR designs being certified by the NRC claim very low core damage frequencies (CDFs). In fact, the CDF for plants of such modern designs will be limited by the vulnerability to earthquakes rather than the vulnerability to operational events. The vulnerable locations identified in the IPEEE analysis seem not to have been so vulnerable in the June 2007 Japanese earthquake. Seismically induced fires did occur. Such fires are not addressed in current PRAs or seismic margins analyses of plants. It is not clear whether such seismically induced fires will be addressed in the ANS standard for external events PRA. The ACRS should follow-up on how the staff and certification applicants are reacting to all this as further understanding of the earthquake develops.

ANS is planning a panel session on the inspection findings of the impact of the Japanese earthquake in June 2008 and paper sessions in November 2008.

Interview of a Candidate for ACRS Membership

The ACRS Member Candidate Screening Panel and the ACRS members interviewed a candidate with I&C experience on December 5 and December 6, 2007, respectively. The Committee should provide feed back to the Chairman of the Panel.

Impact of Continuing Resolution on ACRS Activities

The agency expects to operate under a long-term continuing resolution (CR). Therefore, it is imperative that we know regarding your anticipated travel plans that are not related to ACRS Subcommittee or Full Committee meetings, as well as any other anticipated expenses.

Quadripartite Working Group Meeting

France’s Groupe Permanent Réacteurs (GPR) will host the second Quadripartite Working Group (WG) meeting in France on October 9-10, 2008 on the general topic of “EPR”. The proposed agenda is not ready yet.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 549^{h} ACRS Meeting, February 7 - 9, 2008.

The 548th ACRS Meeting was adjourned at 1:00 PM, December 8, 2007.

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The meeting will be held in Meeting Rooms 1 and 2 of the Postal Square Building Conference Center. The schedule and agenda for the meeting are as follows:

9 a.m.—Opening session
9:15 a.m.—Agency updates and discussion of statistical priorities
11 p.m.—Measures of Intangible Capital: Labor Composition
1 p.m.—Health Care Statistics
2:45 p.m.—Nonresponse bias
4:45 p.m.—Conclude (approximate time)

The meeting is open to the public. Any questions concerning the meeting should be directed to Margaret Johnson, Federal Economic Statistics Advisory Committee, on Area Code (202) 691–5600. Individuals with disabilities, who need special accommodations, should contact Ms. Johnson at least two days prior to the meeting date.

Signed at Washington, DC. the 9th day of November 2007.

Philip L. Rones,

Deputy Commissioner, Bureau of Labor Statistics.

[FR Doc. E7–22585 Filed 11–19–07; 8:45 am]

BILLING CODE 4510–24–P

NATIONAL ARCHIVES AND RECORDS ADMINISTRATION

Agency Information Collection Activities: Submission for OMB Review; Comment Request

AGENCY: National Archives and Records Administration (NARA).

ACTION: Notice.

SUMMARY: NARA is giving public notice that the agency has submitted to OMB for approval the information collections described in this notice. The public is invited to comment on the proposed information collections pursuant to the Paperwork Reduction Act of 1995.

DATES: Written comments must be submitted to OMB at the address below on or before December 20, 2007 to be assured of consideration.

ADDRESSES: Send comments to Desk Officer for NARA, Office of Management and Budget, New Executive Office Building, Washington, DC 20503; fax: 202–395–5167.

FOR FURTHER INFORMATION CONTACT: Requests for additional information or copies of the proposed information collections and supporting statements should be directed to Tamee Fechhelm at telephone number 301–837–1694 or fax number 301–713–7409.

SUPPLEMENTARY INFORMATION: Pursuant to the Paperwork Reduction Act of 1995 (Public Law 104–13), NARA invites the general public and other Federal agencies to comment on proposed information collections. NARA published a notice of proposed collection for this information collection on August 30, 2007 (72 FR 50128 and 50129). One comment was received. NARA has submitted the described information collection to OMB for approval.

In response to this notice, comments and suggestions should address one or more of the following points: (a) Whether the proposed information collection is necessary for the proper performance of the functions of NARA; (b) the accuracy of NARA's estimate of the burden of the proposed information collection; (c) ways to enhance the quality, utility, and clarity of the information to be collected; and (d) ways to minimize the burden of the collection of information on respondents, including the use of information technology; and (e) whether small businesses are affected by this collection. In this notice, NARA is soliciting comments concerning the following information collection:

Title: Order Forms for Genealogical Research in the National Archives.

OMB number: 3095–0027.

Agency form numbers: NATF Forms 81, 82, 83, 84, 85, and 86.

Type of review: Regular.

Affected public: Individuals or households.

Estimated number of respondents: 42,515.

Estimated time per response: 10 minutes.

Frequency of response: On occasion.

Estimated total annual burden hours: 7,086.

Abstract: Submission of requests on a form is necessary to handle in a timely fashion the volume of requests received for these records (2,479 per year for the NATF 81; 280 per year for the NATF 82; 526 per year for the NATF 83; 3,669 per year for the NATF 84; 17,716 per year for the NATF 85; and 17,845 per year for the NATF 86) and the need to obtain specific information from the researcher to search for the records sought. As a convenience, the form will allow researchers to provide credit card information to authorize billing and expedited mailing of the copies. You can also order online at *https://eservices.archives.gov/orderonline.* These forms will also be posted as .pdf files within NARA's online ordering system.

Dated: November 14, 2007.

Martha Morphy,

Assistant Archivist for Information Services.

[FR Doc. E7–22714 Filed 11–19–07; 8:45 am]

BILLING CODE 7515–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting Notice

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on December 6–8, 2007, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Wednesday, November 15, 2006 (71 FR 66561).

Thursday, December 6, 2007, Conference Room T–2b3, Two White Flint North, Rockville, Maryland

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

8:35 a.m.–10:30 a.m.: Draft Final NUREG–1829, ''Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process,'' and Draft NUREG–XXXX, ''Seismic Considerations for the Transition Break Size'' (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding draft NUREG reports on estimating LOCA frequencies through the expert elicitation process and on seismic considerations for the transition break size.

10:45 a.m.–12:15 p.m.: AREVA Enhanced Option III Long Term Stability Solution (Topical Report ANP–10262) (Open/Closed)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and AREVA regarding AREVA Topical Report ANP–10262 on Enhanced Option III Long Term Stability Solution.

Note: A portion of this session may be closed to discuss and protect information that is proprietary to AREVA and their contractors pursuant to 5 U.S.C. 552b(c)(4).

1:15 p.m.–3:15 p.m.: State-of-the-Art Reactor Consequence Analysis (SOARCA) (Open/Closed)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding State-of-the-Art Reactor Consequence Analysis.

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

3:30 p.m.–5:30 p.m.: Draft ACRS report on the NRC Safety Research Program (Open)—The Committee will discuss the draft ACRS report on the NRC Safety Research Program.

5:45 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Friday, December 7, 2007, 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.–11:15 a.m.: Extended Power Uprate Application for the Susquehanna Nuclear Power Plant (Open/Closed)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and the Pennsylvania Power & Light Company regarding the Extended Power Uprate Application for the Susquehanna Nuclear Power Plant and the associated NRC staff's Safety Evaluation.

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

11:30 a.m.–12 p.m.: Subcommittee Report (Open)—The Committee will hear a report by and hold discussions with the Chairman of the ACRS Subcommittee on ESBWR regarding items discussed during the meeting on November 15, 2007.

1:30 p.m.–2:30 p.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

2:30 p.m.–2:45 p.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

2:45 p.m.–3:15 p.m.: Election of ACRS Officers for CY 2008 (Open)—The Committee will elect the Chairman and Vice-Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee for CY 2008.

3:30 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Saturday, December 8, 2007, 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, as well as the draft ACRS report on the NRC Safety Research Program.

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 and protect information classified as proprietary to General Electric, AREVA, and their contractors pursuant to 5 U.S.C 552b (c) (4), and National Security information as well as Safeguards information pursuant to 5 U.S.C. 552b (c) (1) and (3).

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

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.

The ACRS meeting previously scheduled for April 3–5, 2008, and published in the **Federal Register** on October 22, 2007 (72 FR 59573), is rescheduled for April 10–12, 2008.

Dated: November 14, 2007.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E7–22641 Filed 11–19–07; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Federal Register Notice

AGENCY HOLDING THE MEETINGS: Nuclear Regulatory Commission.

DATES: Weeks of November 19, 26; December 3, 10, 17, 24, 2007.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

November 14, 2007

SCHEDULE AND OUTLINE FOR DISCUSSION
548th ACRS MEETING
DECEMBER 6 - 8, 2007

THURSDAY, DECEMBER 6, 2007, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - ~~10:30~~ A.M. 10:05 Draft Final NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," and Draft NUREG-XXXX, "Seismic Considerations for the Transition Break Size" (Open) (GEA/GSS)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff regarding draft NUREG reports on estimating LOCA frequencies through the expert elicitation process and on seismic considerations for the Transition Break Size.

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

~~10:30~~ 10:05 - 10:45 A.M. ***BREAK***

3) 10:45 - ~~12:15~~ 12:20 P.M. AREVA Enhanced Option III Long Term Stability Solution (Topical Report ANP-10262) (Open/Closed) (SAK/ZA)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff and AREVA regarding AREVA Topical Report ANP-10262 on Enhanced Option III Long Term Stability Solution

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

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

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

4) ~~1:15~~ - 3:15 P.M. 1:20 — State-of-the-Art Reactor Consequence Analysis (SOARCA) (Open/Closed) (WJS/HPN)

4.1) Remarks by the Subcommittee Chairman
4.2) Briefing by and discussions with representatives of the NRC staff regarding State-of-the-Art Reactor Consequence Analysis.

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

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

~~3:15 - 3:30~~ P.M. 3:00 – 3:15 — *****BREAK*****

5) 3:30 - 5:30 P.M. — Draft ACRS Report on the NRC Safety Research Program (Open) (DAP/HPN)

5.1) Remarks by the Subcommittee Chairman
5.2) Discussion of the draft ACRS report on the NRC Safety Research Program.

5:30 - 5:45 P.M. — *****BREAK*****

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

Discussion of proposed ACRS reports on:

6.1) Draft Final NUREG-1829 on LOCA Frequencies and Draft NUREG-XXXX on Seismic Considerations for the Transition Break Size (GEA/GSS)
6.2) AREVA Topical Report ANP-10262 on Enhanced Option III Long Term Stability Solution (SAK/ZA)
6.3) State-of-the-Art Reactor Consequence Analysis (WJS/HPN)

FRIDAY, DECEMBER 7, 2007, 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/CS/SD)

8) 8:35 - 11:15 A.M. **(10:00-10:15 A.M. BREAK)** — Extended Power Uprate Application for the Susquehanna Nuclear Power Plant (Open/Closed) (SB/ZA)

8.1) Remarks by the Subcommittee Chairman
8.2) Briefing by and discussions with representatives of the NRC staff and the Pennsylvania Power & Light Company regarding the Extended Power Uprate Application for the Susquehanna Nuclear Power Plant and the associated NRC staff's Safety Evaluation.

[Note: A portion of this session may be closed to discuss and protect information that is proprietary to General Electric and their contractors pursuant to 5 U.S.C. 552b (c) (4).]

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

11:15 - 11:30 A.M. *****BREAK*****

9) 11:30 - 12:00 P.M. <u>Subcommittee Report</u> (Open) (MLC/CGH)
Report by and discussion with the Chairman of the ACRS Subcommittee on ESBWR regarding items discussed during the meeting on November 15, 2007.

12:00 - 1:30 P.M. *****LUNCH*****

10) 1:30 - 2:30 P.M. <u>Future ACRS Activities/Report of the Planning and Procedures Subcommittee</u> (Open) (WJS/FPG/SD)
- 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.

11) 2:30 - 2:45 P.M. <u>Reconciliation of ACRS Comments and Recommendations</u> (Open) (WJS, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

12) 2:45 - 3:15 P.M. <u>Election of ACRS Officers for CY 2008</u> (Open) (FPG/SD)
Election of the Chairman and Vice-Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee for CY 2008.

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

13) 3:30 - 7:00 P.M. <u>Preparation of ACRS Reports</u> (Open)
Discussion of proposed ACRS reports on:
- 13.1) Draft Final NUREG-1829 on LOCA Frequencies and Draft NUREG-XXXX on Seismic Considerations for the Transition Break Size (GEA/GSS)
- 13.2) AREVA Topical Report ANP-10262 on Enhanced Option III Long Term Stability Solution (SAK/ZA)
- 13.3) State-of-the-Art Reactor Consequence Analysis (SOARCA) (WJS/HPN)

13.4) Extended Power Uprate Application for the Susquehanna Nuclear Power Plant (SB/ZA)

SATURDAY, DECEMBER 8, 2007, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 8:30 - 1:00 P.M. **(10:30-10:45 A.M. BREAK)**

Preparation of ACRS Reports (Open)
Continue discussion of proposed ACRS reports listed under Item 13, as well as the draft ACRS report on the NRC Safety Research Program.

15) 1:00 - 1:30 P.M.

Miscellaneous (Open) (WJS/FPG)
Discussion of 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.

NOTE:

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

One (1) electronic copy and thirty-five (35) hard copies of the presentation materials should be provided to the ACRS.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
548th FULL COMMITTEE MEETING

December 6 - 8, 2007

PLEASE PRINT CLEARLY

NRC Attendees

TODAY'S DATE: December 6, 2007

	NAME	NRC ORGANIZATION
1	Harold Vanermolen	RES/DRA
2	Nilesh Chokshi	NRO/DSER
3	Michael Cullingford	NRR/OD
4	Rob Tregoning	RES/DE
5	Stephen Dinsmore	NRR/DRA
6	Dale Rasmuson	RES/PRA
7	Richard Dudley	NRR/DPR
8	Syed K. Shankat	RES/DE
9	Lee Abramson	RES/DE
10	Steven Laur	NRR/DRA
11	Tim Collins	NRR/DSS
12	Yeon-Ki Chung	NRR/DLR
13	Doug Coe	OCM/PBL
14	Tai Huang	NRR/DSS
15	Greg Cranston	NRR/DSS
16	Jocelyn Mitchell	RES
17	Don Dube	NRO/DSRA
18	Marity Stutzke	RES/DRA
19	Jason Schaperow	RES/DSA
20	Robert Prato	RES/DSA
21	ATA Istar	RES/DE
22	Mike Cheok	RES/PRA
23	Jimi Yerokun	RES
24	Jim Vail	NRR/DRA
25	Jim Beall	NRR/DSS
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
548th FULL COMMITTEE MEETING

December 6 - 8, 2007

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NRC Attendees

TODAY'S DATE: December 7, 2007

	NAME	NRC ORGANIZATION
1.	Peter Yarsky	NRR/DSS
2	John Lubinki	NRR/DORL
3	Anthony J. Mendiola	NRR/DSS
4	Diane Jackson	NRR/DSS
5	Steve Smith	NRR/DSS
6	Charles Harris	NRR/DSS
7	Mark Kowal	NRR/DORL
8	Tai Huang	NRR/DSS
9	John Voglewede	RES/DSA
10	Tony Attard	NRR/DSS
11	Paul Clifford	NRR/DSS
12	Kamal Manoly	NRR/DE
13	Ganesh Cheruvenki	NRR/DCI
14	Greg Makar	NRO/DE
15	Muhammad Razzaque	NRR/DSS
16	Evelyn Gettys	NRR/RLRA

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
548th FULL COMMITTEE MEETING

December 6 - 8, 2007

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NRC Attendees - TODAY'S DATE: December 8, 2007

	NAME	NRC ORGANIZATION
1	Mark Orr	RES
2	Rich Guzman	NRR/DORL
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
548th FULL COMMITTEE MEETING

December 6 - 8, 2007

PLEASE PRINT CLEARLY

Visitors

TODAY'S DATE: December 6, 2007

	NAME	ORGANIZATION
1	John Butler	NEI
2	Chet Lehmann	PPL
3	John Geosits	PPl
4	Yousef Farawila	AREVA
5	Ralph Grummer	AREVA
6	Jerry Holm	AREVA
7	Doug Pruitt	AREVA
8	Don Vanover	Exelon
9	Robert Sonders	ORNL
10	Edwin Lyman	UCS
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
548th FULL COMMITTEE MEETING

December 6 - 8, 2007

PLEASE PRINT CLEARLY

Visitors

TODAY'S DATE: December 7, 2007

	NAME	ORGANIZATION
1	Rick Pagodin	PPl/Susquehanna
2	Mike Crowthers	PPL/Susquehanna
3	Jerry Holm	AREVA
4	John A. Bartos	PPL/Susquehanna
5	John Geosits	PPL
6	John Krais	PPL
7	Mike Gorski	PPL
8	Doug Pruitt	AREVA
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
548th FULL COMMITTEE MEETING

December 6 - 8, 2007

PLEASE PRINT CLEARLY

Visitors

TODAY'S DATE: December 8, 2007

	NAME	ORGANIZATION
1	Ralph Grummer	
2	James K. Williams	PPL/Susquehanna
3	John A. Bartos	PPL/Susquehanna
4	Douglas Pruitt	AREVA
5	James M. Smith	PPL/Susquehanna
6	John Krais	PPL/SSES
7	Rick Pagodin	PPL/SSES
8	Jerry Holm	AREVA
9	John Geosits	PPL
10	Rocco R Sgarro	PPL/Susquehanna
11	Michael Garrett	AREVA
12	Yousef Farawil	AREVA
13	Rick Heath	AREVA
14	Chris Hoffman	PPL
15	Michael Crowthers	PPL
16	Mike Gorski	PPl
17	Bruce Swoy	PPL
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SCHEDULE AND OUTLINE FOR DISCUSSION
549th ACRS MEETING
FEBRUARY 7-9, 2008

THURSDAY, FEBRUARY 7, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - 10:30 A.M. Final Review of the License Renewal Application for the Vermont Yankee Nuclear Power Station (Open) (MVB/CGH)
- 2.1) Remarks by the Subcommittee Chairman
- 2.2) Briefing by and discussions with representatives of the NRC staff and Entergy Nuclear Operations regarding the License Renewal Application for the Vermont Yankee Nuclear Power Station and the associated NRC staff's Final Safety Evaluation Report.

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

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

3) 10:45 - 12:00 P.M. Draft Final Revision 1 to Regulatory Guide 1.45 (DG-1173), "Guidance on Monitoring and Responding to Reactor Coolant System Leakage" (Open) (JSA/DEB)
- 3.1) Remarks by the Subcommittee Chairman
- 3.2) Briefing by and discussions with representatives of the NRC staff regarding draft final revision 1 to Regulatory Guide 1.45 (DG-1173) and the staff's resolution of public comments.

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 - 3:00 P.M. Proposed Licensing Strategy for the Next Generation Nuclear Plant (NGNP) (Open/Closed) (MLC/MB)
- 4.1) Remarks by the Subcommittee Chairman
- 4.2) Briefing by and discussions with representatives of the NRC staff and Department of Energy regarding the proposed licensing strategy for the Next Generation Nuclear Plant.

[Note: A portion of this session may be closed to prevent disclosure of information the premature disclosure of which would be likely to significantly frustrate implementation of a proposed agency action pursuant to 5 USC 552b (c)(9) (B).]

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

3:00 - 3:15 P.M. ***BREAK***

5) 3:15 - 5:00 P.M. Cable Response to Live Fire (CAROLFIRE) Testing and Fire Model Improvement Program (Open) (SB/GSS/HJV)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff and its contractors regarding the results of the CAROLFIRE Testing and Fire Model Improvement Program, including staff's resolution of public comments.

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

5:00 - 5:15 P.M. ***BREAK***

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

Discussion of proposed ACRS reports on:

6.1) License Renewal Application for the Vermont Yankee Nuclear Power Station (MVB/CGH)

6.2) Draft Final Revision 1 to Regulatory Guide 1.45 (DG-1173), "Guidance on Monitoring and Responding to Reactor Coolant System Leakage" (JSA/DEB)

6.3) Proposed Licensing Strategy for the Next Generation Nuclear Plant (NGNP) (MLC/MB)

6.4) Cable Response to Live Fire Testing and Fire Model Improvement Program (SB/GSS/HJV)

6.5) State-of-the-Art Reactor Consequence Analysis (SOARCA) Program (WJS/HPN)

FRIDAY, FEBRUARY 8, 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/CS/SD)

8) 8:35 - 10:30 A.M. Proposed BWR Owners Group (BWROG) Topical Report on Methodology for Calculating Available Net Positive Suction Head (NPSH) for ECCS Pumps (Open/Closed) (SAK/ZA)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff and the BWR Owners Group regarding the proposed topical report on Methodology for Calculating the Available NPSH for ECCS Pumps, including NRC staff's position on this topical report.

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

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

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

9) 10:45 -11:30 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (WJS/FPG/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.

10) 11:30 -11:45 A.M. Reconciliation of ACRS Comments and Recommendations (Open) (WJS, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

11) 11:45 – 12:00 P. M. Subcommittee Report (Open) (GEA/HPN) Report by the Chairman of the ACRS Subcommittee on Reliability and PRA regarding Draft NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," that was discussed during the meeting on December 19, 2007.

12:00 - 1:00 P.M. *****LUNCH*****

12) 1:00 - 3:00 P.M. Draft ACRS Report on the NRC Safety Research Program (Open) (DAP/HPN)

12.1) Remarks by the Subcommittee Chairman

12.2) Discussion of the draft ACRS report on the NRC Safety Research Program.

3:00 - 3:15 P.M. *****BREAK*****

13) 3:15 - 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

13.1) License Renewal Application for the Vermont Yankee Nuclear Power Station (MVB/CGH)

13.2) Draft Final Revision 1 to Regulatory Guide 1.45 (DG-1173), "Guidance on Monitoring and Responding to Reactor Coolant System Leakage" (JSA/DEB)

13.3) Proposed Licensing Strategy for the Next Generation Nuclear Plant (NGNP) (MLC/MB)

13.4) Cable Response to Live Fire Testing and Fire Model Improvement Program (SB/GSS/HJV)
13.5) State-of-the-Art Reactor Consequence Analysis (SOARCA) Program (WJS/HPN)

SATURDAY, FEBRUARY 9, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 7:30 - 9:30 A.M. Draft ACRS Report on the NRC Safety Research Program (Open) (DAP/HPN)
Continue discussion of the draft ACRS report on the NRC Safety Research Program.

9:30-9:45 A.M. BREAK

15) 9:45 - 1:00 P.M. Preparation of ACRS Reports (Open)
Continue discussion of proposed ACRS reports listed under Item 13.

16) 1:00 - 1:30 P.M. Miscellaneous (Open) (WJS/FPG)
Discussion of 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.

NOTE:

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

One (1) electronic copy and thirty-five (35) hard copies of the presentation materials should be provided to the ACRS.

13.4) Cable Response to Live Fire Testing and Fire Model Improvement Program (SB/GSS/HJV)
13.5) State-of-the-Art Reactor Consequence Analysis (SOARCA) Program (WJS/HPN)

SATURDAY, FEBRUARY 9, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 7:30 - 9:30 A.M. Draft ACRS Report on the NRC Safety Research Program (Open) (DAP/HPN)
Continue discussion of the draft ACRS report on the NRC Safety Research Program.

9:30-9:45 A.M. BREAK

15) 9:45 - 1:00 P.M. Preparation of ACRS Reports (Open)
Continue discussion of proposed ACRS reports listed under Item 13.

16) 1:00 - 1:30 P.M. Miscellaneous (Open) (WJS/FPG)
Discussion of 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.

NOTE:

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One (1) electronic copy and thirty-five (35) hard copies of the presentation materials should be provided to the ACRS.

ACRS	ACRS	ACRS	Filed: CM-180
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APPENDIX V

LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
548th ACRS MEETING
December 6 - 8, 2007

MEETING HANDOUTS

AGENDA ITEM #	DOCUMENTS/HANDOUTS LISTED IN ORDER
1.	Opening Remarks by the ACRS Chairman
2.	Draft Final NUREG-1829, “Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process,” and Draft NUREG-XXXX, “Seismic Considerations for the Transition Break Size” 1. Seismic Considerations for TBS (Slides from NRC/NRO & RES, Chokshi/Shaukat and Wilkowski) 2. Passive System LOCA Frequencies for Risk-Informed Revision of 10 CFR 50.46 (Slides from NRC/RES, Tregoning and Abramson)
	Break 3. Interview Schedule
3.	AREVA Enchanced Option III Long Term Stability Solution (Topical Report ANP-10262) 4. AREVA Stability Methodologies: DIVOM & Enhanced Option III (Slides from NRC/NRR, Huang) 5. Enhanced Option III Long Term Stability Solution and DIVOM Methodology using RAMONAA5-FA Code (Slides from AREVA)
4.	State-of-the-Art Reactor Consequence Analysis (SOARCA) 6. State-of-the-Art Reactor Consequence Analyses (Slides from NRC) 7. NRC’s SOARCA Program: UCS Concerns (Slides from Dr.Edwin Lyman, Union of Concerned Scientists)
6.	Preperation of ACRS Report 8. NUREG-1635, Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program (copy of the NUREG)
8.	Extended Power Uprate Application for the Susquehanna Nuclear Power Plant 9. Susquehanna Steam Electric Station (Slides from PPL) 10. Thermal Mechanical Methods (Slides from AREVA, Garrett) 11. Susquehanna Power Uprate Fuel System Design Review (Slides from NRC/NRR, Clifford)
9.	Subcommittee Report 12. MLC Summary of ESBWR Subcommittee Meeting (11-15-2007)
11.	Reconciliation of ACRS Comments and Recommendations 13. Reconciliation Handout

**Copies of most of the handouts can be found posted on the ACRS portion of the NRC Public Website.

[Note: Some documents listed herein may have been provided or prepared for the Committee use only. These documents must be reviewed prior to release to the public.]

Seismic Considerations for TBS

Presented to

The Advisory Committee on Reactor Safeguards

Presented by:
Nilesh Chokshi, Dep. Dir. DSER/NRO, ncc1@nrc.gov
S. Khalid Shaukat, DE/RES, sks1@nrc.gov
Gery Wilkowski, Emc2, gwilkowski@emc-sq.com

Project Team:
C. Carpenter, J. Fair, C. Greene, G. Hammer, A. Hiser, M. Kirk, A. Wilson, NRC
G. DeGrassi, BNL, J. Johnson, JJJ and Associate, R. Olson, Battelle

December 6, 2007

Outline of the Presentation

- Basic Objective
- Approach
- Key Assumptions
- Results
- Draft Rule and Questions
- Public Comments and Response to Questions
- Current Status and Future Activities

Objectives and Approach

- **Objectives**
  - **To examine likelihood and conditions that would result in seismically-induced breaks incompatible with the proposed TBS.**
  - **Provide key considerations to facilitate the public review and comments**
- **Approach**
  - **Use of hybrid deterministic and probabilistic approaches**
  - **Six supporting activities**
    - **Unflawed piping**
    - **Flawed piping**
    - **Indirect failures**
    - **Review of past earthquake experience**
    - **Review of past PRAs**
    - **Review of a LLNL study conducted in connection with revision to GDC4**

Approach – Key Assumptions and Scope (Unflawed and Flawed Piping Analysis)

- Used available design information (e.g., normal operating stresses, seismic stresses, and material properties)
  - Such results only available for PWRs from LBB application database; therefore, evaluations are limited to PWRs
- Used LLNL hazard curves – then latest publicly available– for plants east of Rocky Mountains
- Include piping systems with diameter larger than the TBS diameter (e.g., hot leg, cold leg, and cross-over leg)
- Determined seismic stresses at 10^{-5} (or 10^{-6}) seismic event (elastic stresses) by scaling plant specific SSE stresses
- Apply a correction to 10^{-5} seismic stresses to account for conservatisms in the design process and the extrapolation to higher levels

Key Findings –Unflawed Piping

- **Our results show frequency of seismically-induced breaks much lower than 1E-5 / year for the piping systems evaluated**

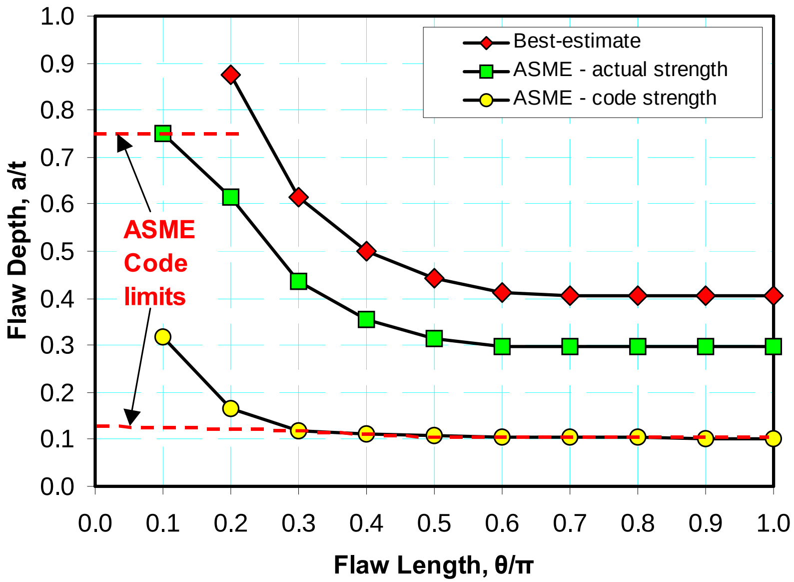
- **Unflawed piping case can be eliminated from further analyses as flawed piping will have to be evaluated.**

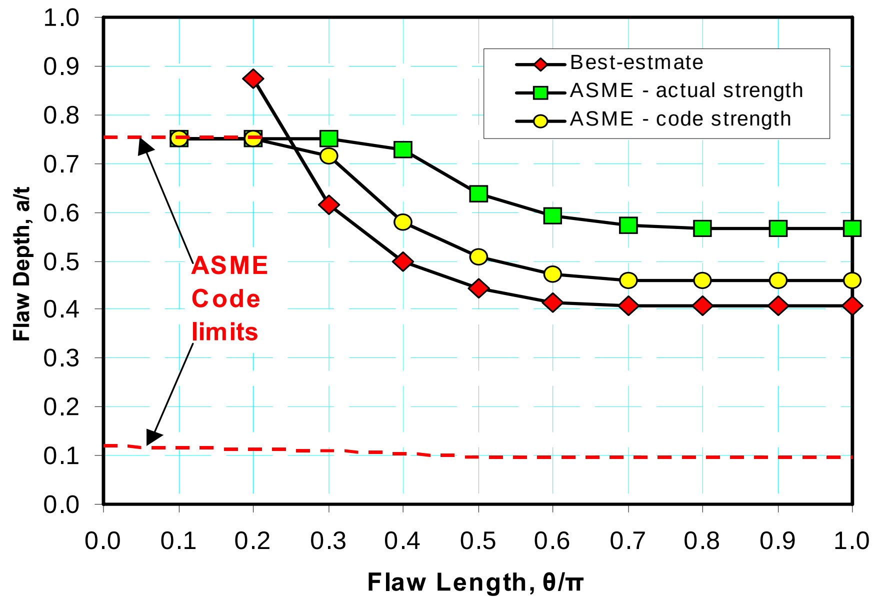
Approach – Flawed Piping Two Key Questions

- Rather than conducting a full probabilistic analysis for flaw development and critical flaw sizes for the entire seismic hazard curve, the approach examined maximum allowable flaw sizes at the N+SSE seismic condition (with all the normally imposed safety factors) relative to critical flaws for 10^{-5} (or 10^{-6}) seismic events (with more realistic criteria). If the N+SSE flaw sizes are smaller than the critical flaw sizes corresponding to the 10^{-5} or 10^{-6} seismic events, then there is inherent protection for the 10^{-5} or 10^{-6} seismic flaws from the N+SSE allowable flaw sizes.

- Two flaw evaluation procedures for N+SSE loading included:
  1. ASME inspection/evaluation criteria for circumferential ***surface flaws***
  2. NRC LBB procedures for circumferential ***through-wall flaws***

Example of Results: Code Surface Flaw Evaluations at N+SSE (with all SFs) Relative to Critical Flaw Size at 10^{-5} Seismic Event



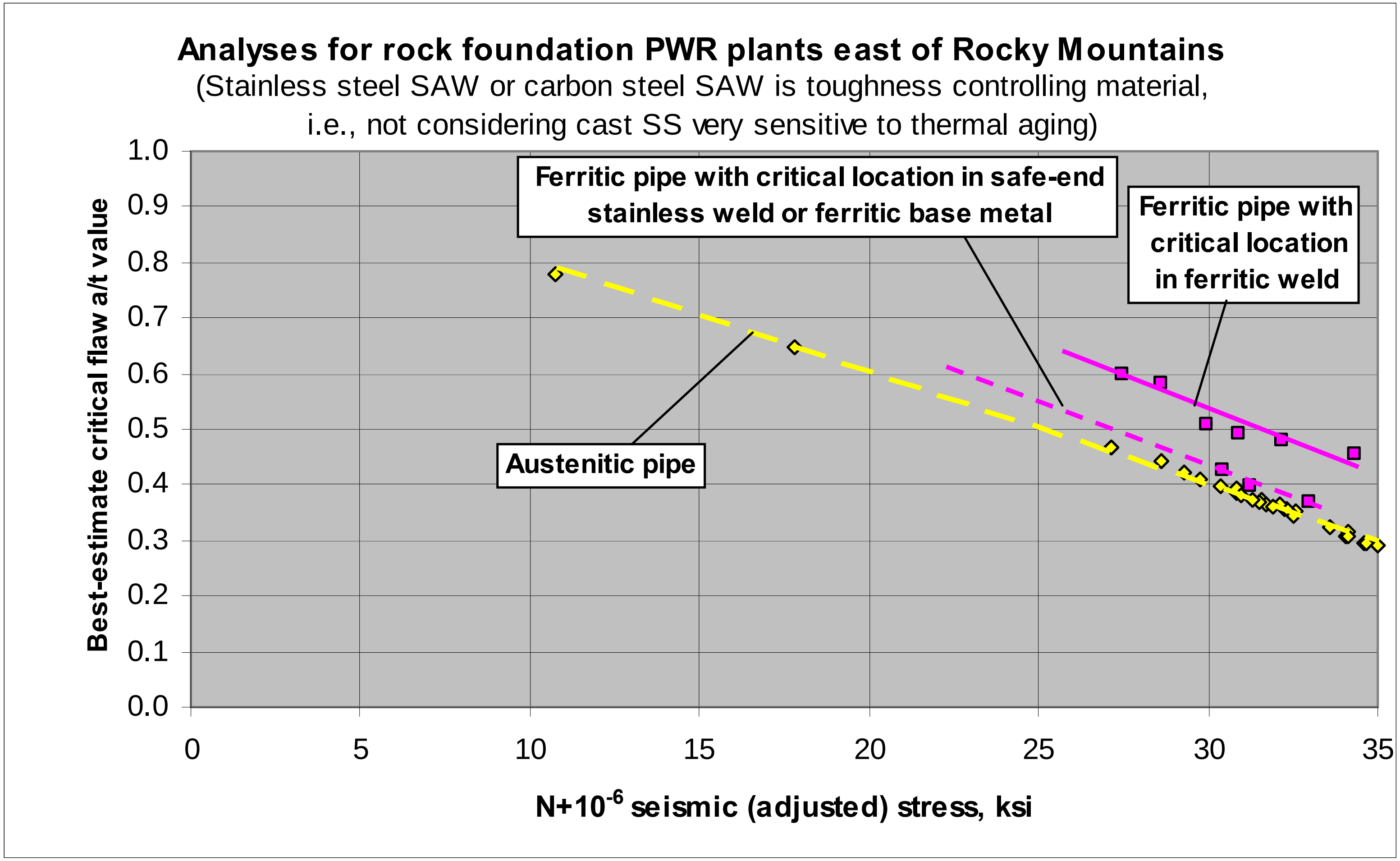


a) ASME flaw sizes *smaller* than critical flaw at 10^{-5} seismic

(b) ASME flaw sizes *greater* than critical flaw at 10^{-5} seismic

Results From All Analyses For Surface Flawed Piping

a/t values for long flaws at 10^{-6} seismic event



Results – Surface Flawed Piping Evaluation of Maximum Allowable Code Flaws

- **Results from analysis of 52 large-diameter pipe systems**

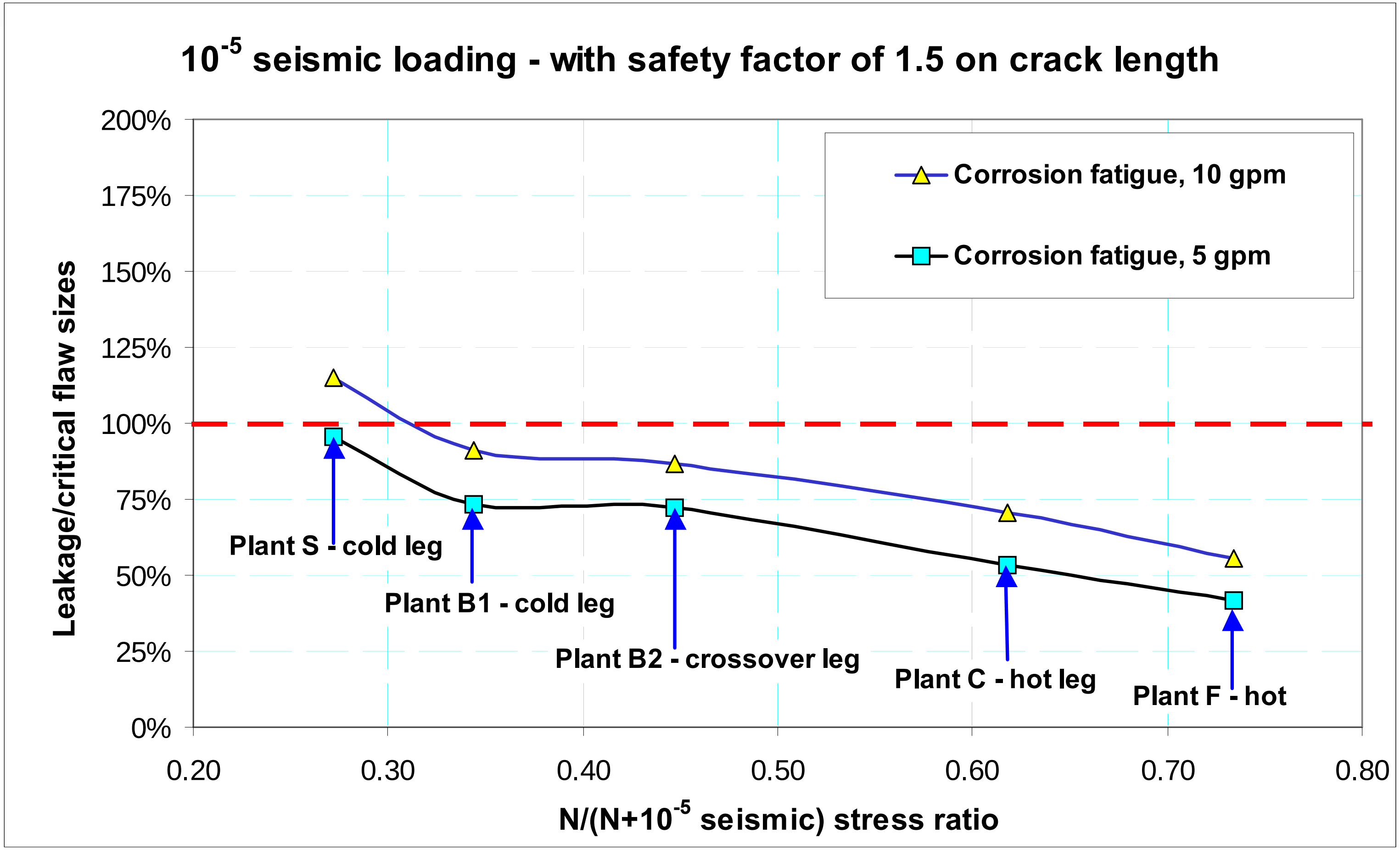
	N +10^{-5} seismic loading	**N +10^{-6} seismic loading**
ASME Code N+SSE allowable flaw *smaller* than critical flaw size *(Desirable result)*	48 cases	20 cases
Critical flaw size *bracketed by* two different *ASME Code flaw evaluation procedures*	1 case	20 cases
ASME Code N+SSE allowable flaw *larger* than critical flaw size *(Undesirable result, but still large flaw sizes)*	3 cases (Limiting surface flaw depth = 40% of thickness)	12 cases (Limiting surface flaw depth = 30% of thickness)

Through-Wall Flaw (LBB) Evaluation Approach

- **For standard LBB analysis at N+SSE stresses with applicable safety factors (SF) on leak rate (SF = 10) and leakage flaw size (SF = 2) and code parameters for critical flaw size analysis**

- **For N+10^{-5} and 10^{-6} seismic loading considered alternate cases with different SFs, but with more realistic accounting for fracture toughness properties**

N + 10^{-5} Seismic Stresses with Safety Factor of 1.5 on Crack Length



Key Findings – Flawed Piping

- In most cases, the ASME maximum allowable surface-flaw size at N+SSE loading is smaller than the critical flaw at 10^{-5} or 10^{-6} seismic event loading. For cases that don't meet this condition, flaw sizes are still quite large.
  - Critical crack depths are larger than 40% of thickness for 10^{-5} seismic stresses
  - Critical crack depths are larger than 30% of thickness for 10^{-6} seismic stresses

- The LBB flaw sizes associated with the SSE loading are smaller than the critical mean through-wall flaws at 10^{-5} and 10^{-6} seismic events for most cases with the SFs of 1.5 and 1.0. respectively.
  - The few cases that don't pass with these SFs, could pass with a smaller normal operating leak-rate detection capabilities.

Approach - Indirect Failure

- Failure of support of large components which may lead to failure of piping – supports are of most interest
- Use LLNL results and update them to reflect new hazard and ground motion information
- Convolve a support fragility with mean LLNL hazard to obtain mean failure probability
- Assumption – large component support failures lead to piping failure

Approach - Indirect Failure Sample LLNL Results

- **Our mean result for Calvert Cliffs – 1.7E-06/year compared to LLNL 90% confidence value of 6.1E-6**

Group A Plants **(Combustion Engineering)**	Confidence Limit (1)		
	10%	50%	90%
Calvert Cliffs	2.3×10^{-8}	6.1×10^{-7}	6.1×10^{-6}
Millstone 2	9.0×10^{-10}	6.6×10^{-8}	1.2×10^{-6}
Palisades	5.0×10^{-7}	6.4×10^{-6}	5.2×10^{-5}
St. Lucie 1	1.2×10^{-8}	3.8×10^{-7}	4.1×10^{-6}
St. Lucie 2	6.6×10^{-8}	1.4×10^{-6}	1.1×10^{-5}
Westinghouse Lowest Capacity Plant	2.3×10^{-7}	3.3×10^{-6}	2.3×10^{-5}

(1) A confidence limit of 90% implies that there is a 90% subjective probability (confidence) that the probability of indirect DEGB is less than the value indicated.

(1) Generic seismic hazard curves used in evaluation.

Summary of Key Findings

- Frequency of seismically-induced breaks much lower than 1E-5/year for the unflawed piping systems evaluated

- Critical surface flaw and through-wall flaw evaluations
  - ASME Code maximum allowable surface flaws generally smaller than critical flaws at N+1E-5 or 1E-6 seismic event. In all cases, -critical crack sizes are very large.
  - The LBB flaw sizes for N+SSE loading (with SFs on flaw length) generally smaller than critical through-wall flaws at seismic events of 1E-5 and 1E-6/year with reduced safety factors.

- For two cases analyzed, indirectly induced piping failure (attributable to major component support failure) has a mean failure probability on the order of 1E-6/year.

Draft Rule and Specific Questions

- **Draft rule issued with the discussion of the seismic issue including whether a plant-specific assessments were needed or not.**

- **To facilitate feedback, comments were solicited on the following points:**
  - **Results of the evaluations contained in the report**
  - **Effects of pipe degradation on seismically-induced LOCA frequencies and the potential affecting the selection of the TBS**
  - **Potential approaches and options to address this issue**

Public Comments

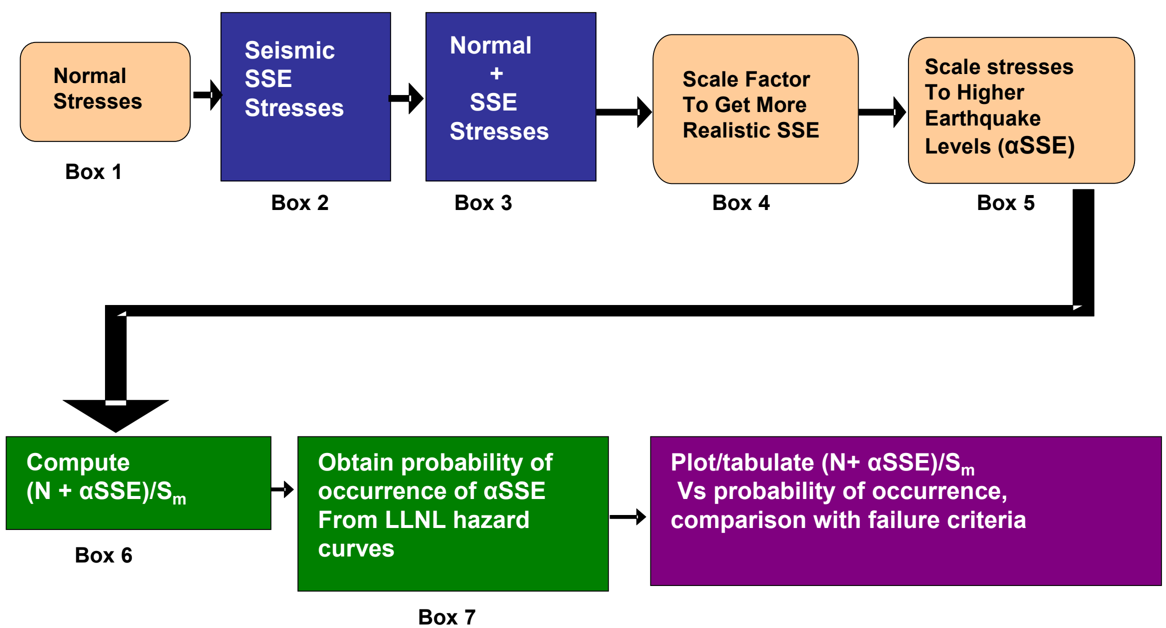
- **Industry responses and comments:**
  - **TBS is not adversely affected by seismic considerations**
  - **Delta risk due to seismic is considered low**
  - **EPRI evaluated sample cases of indirect failure using updated seismic hazard with failure frequency less than 1E-5/yr**
  - **Plant-specific assessments should not be required**

Current Status and Future Activities

- **The staff will evaluate the need for plant-specific assessment considering the following factors:**
  - **Response to the questions issued with the draft rule**
  - **How the rule is revised to address the Commission SRM and the ACRS recommendations, particularly those associated with the defense-in-depth and mitigation.**
  - **What impact any potential changes under the new rule may have on the seismic risk**
  - **Guidance and acceptance criteria to demonstrate applicability of NUREG-1829 results to individual plants.**

Backup Slides

Approach – Unflawed piping



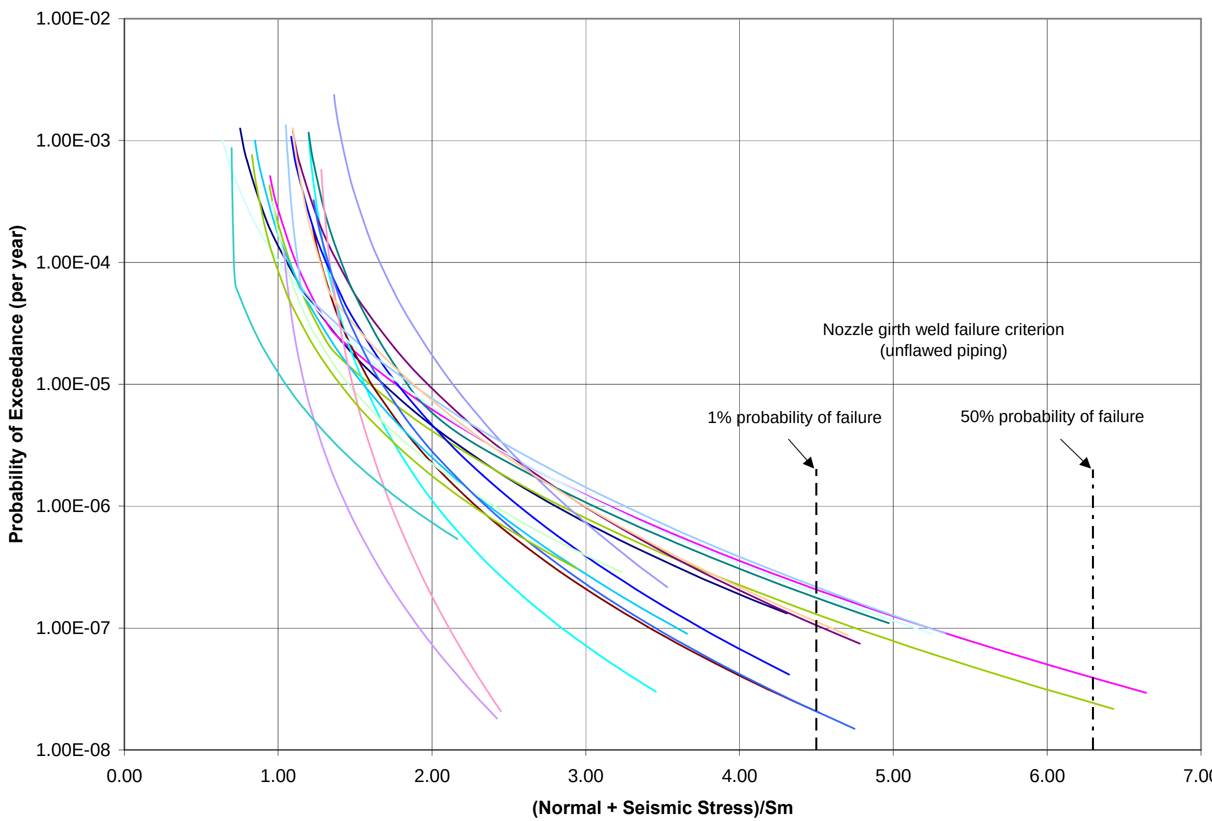
Note: **Scale factor is an approach to estimate more realistic seismic stresses at various ground motion levels**

Approach – Flawed Piping Surface Flaw Evaluation

- 52 large-diameter piping systems examined in 27 PWR plants
  - ASME allowable flaw size using actual or Code strength properties, and
  - Critical flaw size analysis for a 10^{-5} (or 10^{-6}) annual probability of exceedance seismic event using a number of corrections for best-estimate evaluations.
    - Used all stresses pressure, dead-weight, seismic inertial, SAM, and thermal expansion,
    - Flawed piping analysis based on fracture criteria that assumes nonlinear behavior, so additional correction applied to elastic stress analyses, and
    - More realistic account for material strengths and toughness values for 10^{-5} (or 10^{-6}) critical flaw.
    - Excluded cast stainless steels that might be much lower in toughness due to thermal aging.

Results for Unflawed Piping

Probability of Exceedance vs. (N + Seismic) / Sm
Reactor Coolant Loop Piping at 27 PWRs



Unflawed piping failure criterion based on an EPRI test program which was used to develop a technical basis for the ASME section III design rule changes

Passive System LOCA Frequencies for Risk-Informed Revision of 10 CFR 50.46

Robert L. Tregoning
Lee Abramson
NRC\RES

Paul Scott
Battelle

Advisory Committee on Reactor Safeguards
December 6, 2007

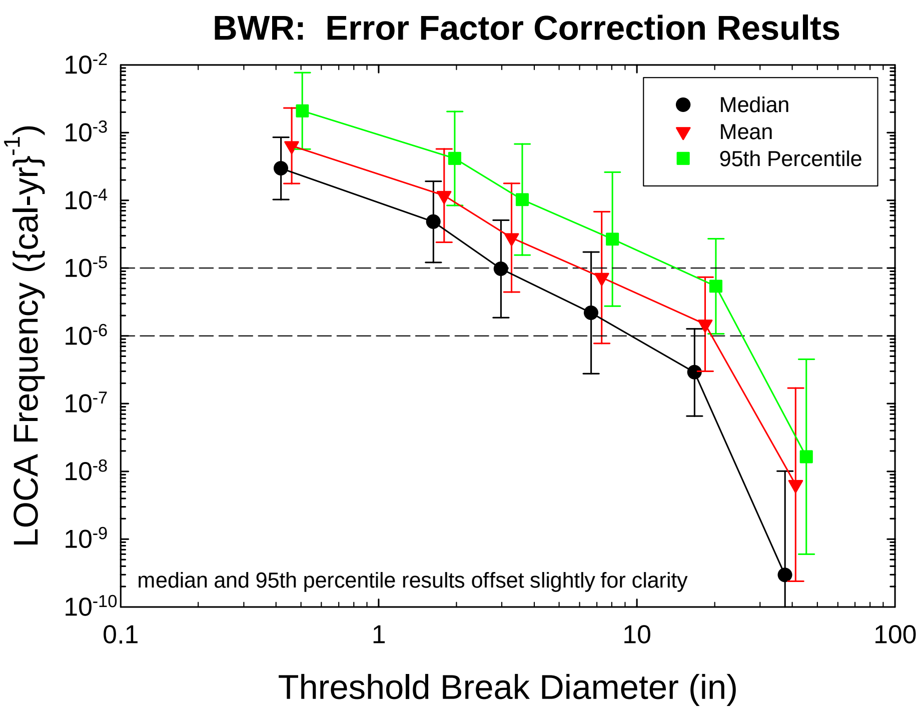
LOCA Frequency Reevaluation

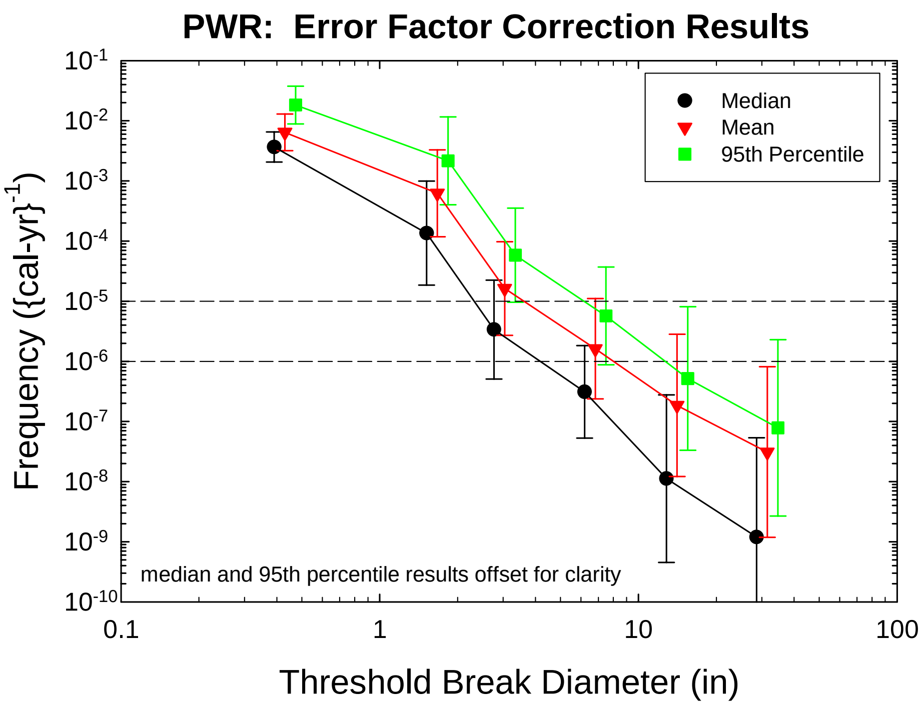
- Commission direction (SRM-02-0057)
  - "The staff should provide the Commission a comprehensive 'LOCA failure analysis and frequency estimation' that is realistically conservative and amenable to decision-making ... with appropriate margins for uncertainty ...".
  - "The staff should use expert elicitation to converge (whenever possible) service-data and PFM results ...".
- ACRS request
  - Letter stating that NUREG-1829 sufficiently meets the Commission direction and should be published

Executive Summary

- Formal elicitation process used to estimate generic BWR and PWR passive-system LOCA frequencies associated with material degradation.
- Panelists provided quantitative estimates supported by qualitative rationale in individual elicitations for underlying technical issues.
  - Generally good agreement on qualitative LOCA contributing factors.
  - Large individual uncertainty and panel variability in quantitative estimates.
- Group results for the LOCA frequency distribution parameters (i.e., 5^{th}, 50^{th}, 95^{th}, and mean) determined by aggregating panelists' estimates.
  - Geometric mean aggregated results are consistent with elicitation objective and structure; they are also generally comparable with NUREG/CR-5750 estimates.
  - Alternative aggregation schemes can result in higher LOCA frequencies.

Total LOCA Frequencies



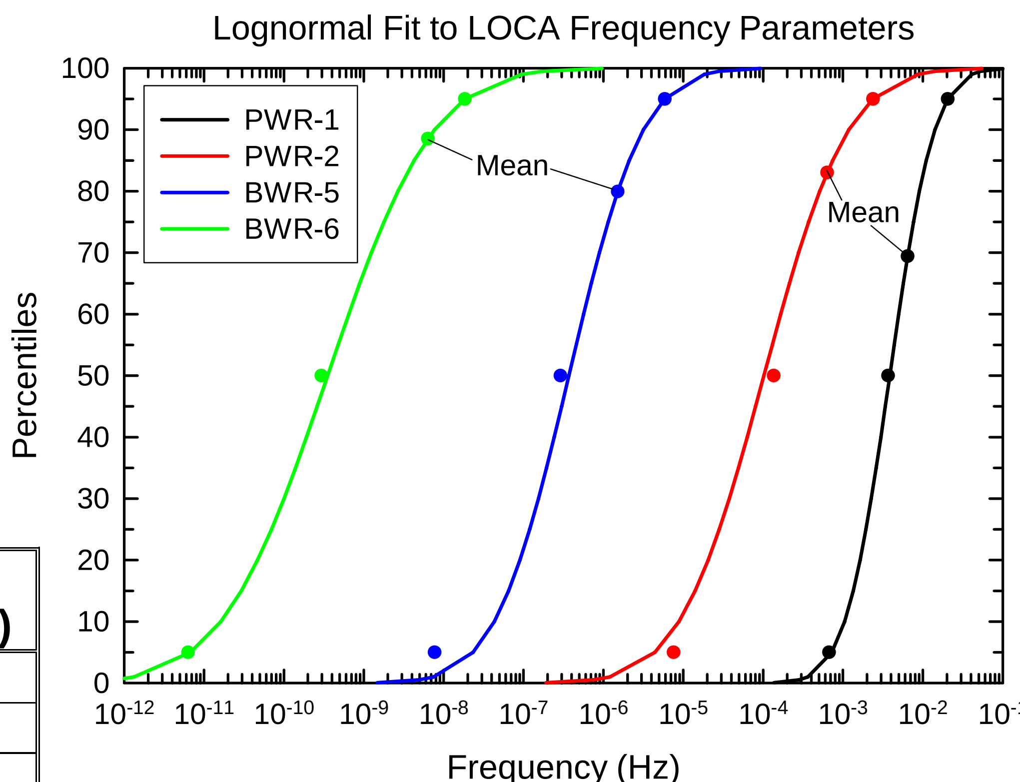


- Individual results adjusted for overconfidence and aggregated using geometric mean
- 95% confidence bounds (i.e., error bars) reflect diversity among panelists
- Differences between medians and 95th percentiles reflect individual panelist uncertainty

Lognormal Fit to LOCA Frequency Parameters

- Fits to 95th percentile and mean provide a reasonable representation
- Less than 30% error in the median
- 50% error or less in 5th percentile, except for BWR-5 case

Type	Cat.	5th (% Error)	50th (% Error)
PWR	1	8	5
PWR	2	-42	-25
BWR	5	200	28
BWR	6	7	20

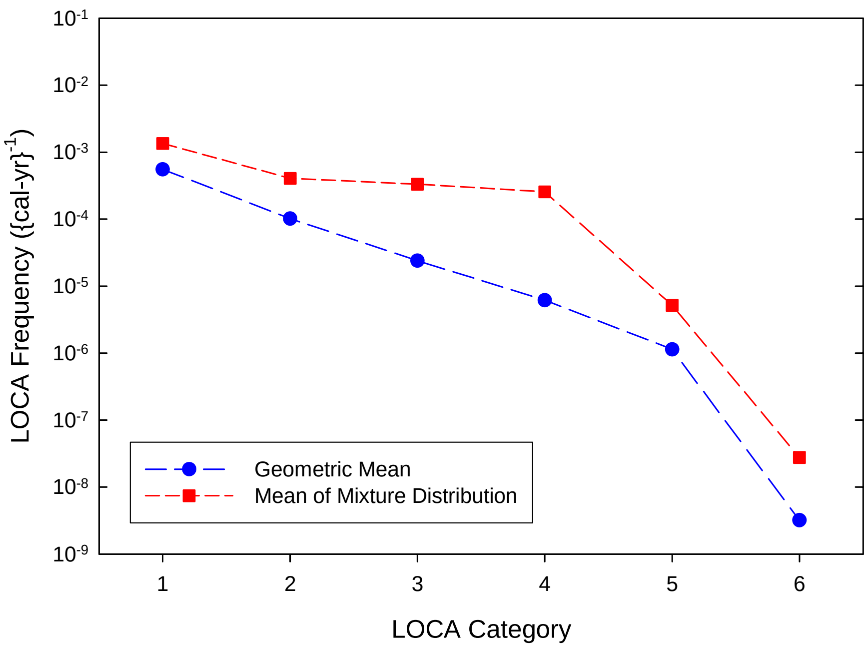


Analysis of Elicitation Responses: Sensitivity Analyses

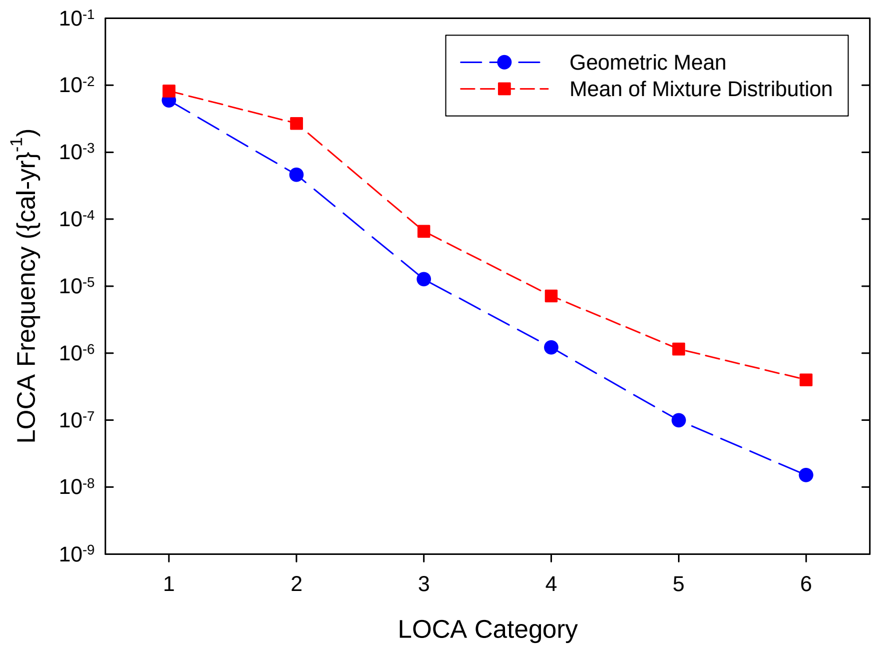
- Determine effect of assumptions on the LOCA frequency estimates
- Sensitivity analyses conducted in five broad areas of analysis.
  - Determination of calculated means
  - Overconfidence adjustment
  - Correlation structure of panelist responses
  - **Aggregation of individual results**
  - Measurement of panel diversity

Aggregation of Individual Results: Mixture Distribution vs. Geometric Mean

BWR Current Day Estimates



PWR Current Day Estimates



- Group estimates can be significantly affected by aggregation method!

Internal and External Reviews

- NUREG-1829 has been extensively reviewed
- Expert panel
  - Individual responses
  - Calculations and analysis
  - General qualitative and quantitative findings and conclusions
- External peer review (decision analyst and statistician)
  - General elicitation structure
  - Analysis procedure and framework
  - Aggregation and sensitivity analyses
  - Review reports are publicly available
- ACRS review
  - Elicitation process, structure, analysis, results, and application for 50.46
- Internal staff review
  - Analysis procedure and framework, aggregation and sensitivity analyses, and application to 10 CFR 50.46
- Public review and comment

Public Comment Schedule and Results

- Draft NUREG-1829 issued June 2005
- Public comment period closed November 2005
- Identified 29 comments from public
  - Bill Galyean (elicitation panelist)
  - Penn State University – Professor Larry Hochreiter
  - Palo Verde Nuclear Power Plant staff
  - BWR Owners Group
  - Westinghouse Owners Group
  - Nuclear Energy Institute
- NRR staff provided additional comments in parallel with public comment period
- In total, 101 separate comments were identified

Public Comment Summary

- Public comments identified additions and clarifications to improve the exposition and facilitate the use of NUREG-1829
- No comments presented a significant challenge to the appropriateness of the objective, elicitation approach, analysis, or results
- Most passionate controversy remains the proper method for aggregating individual estimates to produce group estimates

Public Comment Example: Comparisons with Service Experience

- NUREG-1829 SB LOCA estimates too high
  - Approximately 1 order of magnitude higher than NUREG/CR-5750 results
  - Implies one SB LOCA every 4 years for US reactor fleet
  - Using NUREG-1829 estimates in existing PRAs would lead to unwarranted impacts that are not supported by operational experience

- Related comments: GC12, 7-1, 7-3, 7-7, 7-8, 7-9

Comparisons with Service Experience: Response

- NUREG-1829 SB LOCA and NUREG/CR-5750 estimates are generally consistent
  - SGTR estimates are virtually identical
  - BWR SB LOCA estimates are similar (within 20%)
  - PWR SB LOCA estimates are higher (by approximately a factor of 5)
- NUREG-1829 SB LOCA estimates are consistent with operating experience
- Differences that do exist are supported by the quantitative estimates and qualitative rationale provided by panelists
- Resulting NUREG modifications
  - Provided separate PWR SGTR and SB LOCA estimates (Section 7.8)
  - Provided more extensive comparisons between NUREG-1829 estimates and historical results (Section 7.9)
  - Compared estimates with operational experience (Section 7.10)

AREVA Stability Methodologies: DIVOM & Enhanced Option III

Dr. Tai L. Huang (NRR/ADES/DSS/SRXB)

ACRS Committee Meeting

Dec 6, 2007

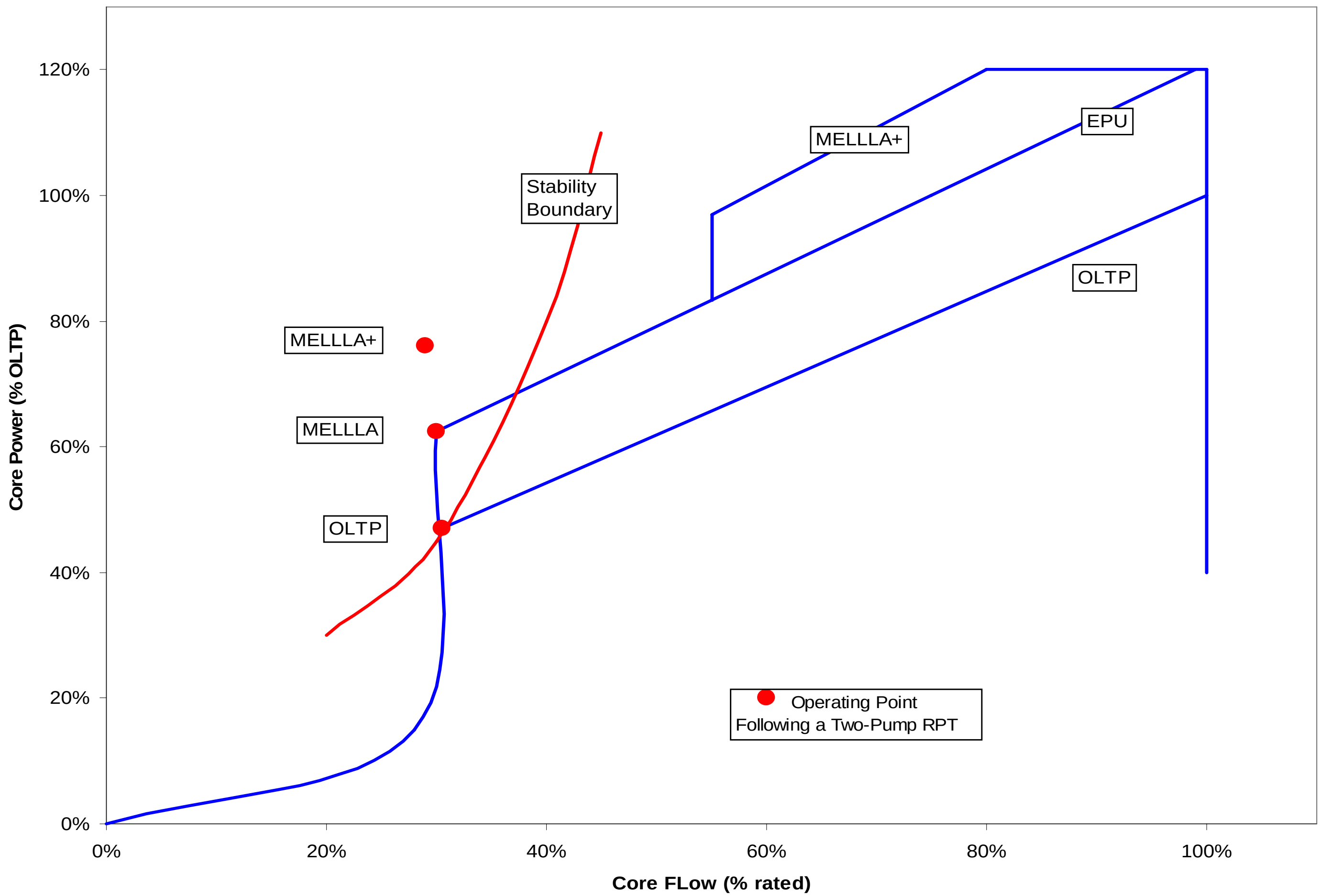
OPEN SESSION

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Scope of Staff Review

- This staff review applies to two AREVA reports in the area of stability:
  - **ANP-10262(P)**, Rev 0, Enhanced Option III Long Term Stability Solution. Framatome ANP. January 2006
    - A new long term stability solution algorithm applicable to extended flow domains (EFD's) like MELLLA+
  - **BAW-10255(P)**, Rev 2, Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code. Framatome ANP. January 2006
    - AREVA's methodology for calculating the DIVOM correlation, which is a required component of detect and suppress solutions

Extended Operating Domains Pose New Challenges to Stability



Long Term Stability Solutions for Original Thermal Power

- Options were developed by BWROG and publicly available
  - Documented in NEDO-31960A “BWR Owner's Group Long-Term Stability Solutions Licensing Methodology,” Nov 95
  - Approved for operation at Original Licensed Thermal Power (OLTP) operation
- Prevention (anticipatory scram)
  - Option E1A
  - Option ID
- Detect & Suppress
  - Option II
  - Option III

LTS for Extended Operating Domains

- Two LTSs address stability challenges for operating in extended operating domains (e.g., MELLLA+)
  - DSS-CD
    - NRC reviewed and approved for MELLLA+
    - GE Proprietary
  - Enhanced Option III (EO-III)
    - Focus of current staff review
    - Areva Proprietary

Enhanced Option III

- Enhanced Option III (EO-III) is an evolutionary step relying on the existing methodology and hardware for Solution III.
- EO-III introduces measures for addressing the reduced stability associated with extended flow window conditions and the higher probability of single channel hydraulic instability excitation
- The new elements, introduced as enhancements to the existing Option III solution are
  – Introduction of a calculated exclusion region on the power/flow map designed to preclude single channel instabilities.
  – Calculation procedures consistent with the introduction of the channel instability exclusion region

AREVA Cycle-Specific DIVOM Methodology

- The DIVOM curve is a relationship between the hot bundle relative oscillation magnitude and the limiting fractional change in critical power ratio
- This review addresses the capabilities of the RAMONA5-FA system code to model neutron-coupled density wave oscillations of the regional mode type, and the range of input data defining the state points within the reload cycle for which the DIVOM curve is generated.
- It also addresses the procedure for post-processing the system code output to generate the DIVOM data consistent with their intended application

Review Conclusions: EO-III

- The staff concludes that EO-III is an acceptable methodology to detect and suppress oscillations should they occur and, thus, satisfies General Design Criteria GDC-12
  - The EO-III Solution features provide protection up to and including MELLLA+ conditions

Review Conclusions: DIVOM

- The AREVA DIVOM Methodology is consistent with the previously approved BWROG methodology
- RAMONA5-FA is an integral part of the AREVA DIVOM Methodology. RAMONA5-FA is capable of:
  – Computing power, flow, and void oscillations with consistent phase lags and of a frequency representative of unstable oscillations
  – Estimate the loss of critical power ratio (CPR) induced by these oscillations

- AREVA has committed to support the staff review of the RAMONA5-FA code for DIVOM calculations

RAMONA5-FA Limitation

- EFW operation (e.g. MELLLA+) poses additional challenges to the calculations; therefore, the staff imposes the following conditions:
  - The application of RAMONA5-FA to calculate the DIVOM curve under extended flow window operating domains (such as MELLLA+) is restricted to stability solutions having a scram protected exclusion region that substantially reduces the potential severity of power oscillations.
  - A penalty of 10% must be added to DIVOM slopes calculated by RAMONA5-FA for extended flow window operating domains. This penalty is equivalent to a penalty of 10% added to the calculated relative CPR response for a given power oscillation magnitude.
- The above restrictions shall remain in effect until the staff completes a detailed review of the RAMONA5-FA code and its ability to calculate DIVOM curves in extended flow window operating domains.

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Enhanced Option III Long Term Stability Solution and DIVOM Methodology using RAMONA5-FA Code

Presentation to the Advisory Committee on Reactor Safeguards

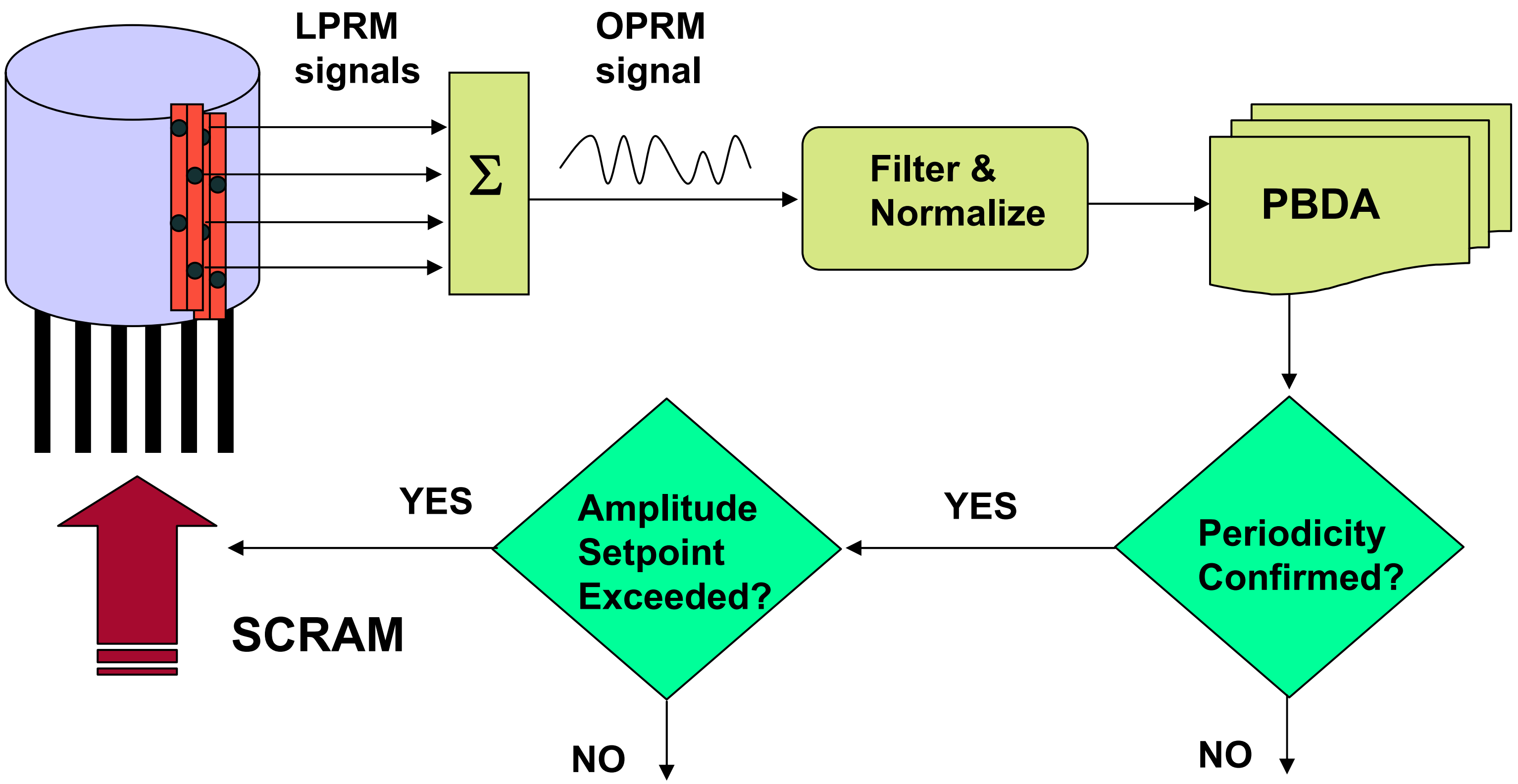
December 6, 2007

Presentation Roadmap

> Overview of Option III Detect & Suppress Solution
  - Part 21 Report and recovery
    - Short Term: Cycle-Specific DIVOM
    - Long Term: Include MELLLA+
> The Enhanced Option III Solution
  - The single (few) channel hydraulic instability exclusion
> Codes and Methods supporting EO-III
> Questions, Discussions, and Conclusions

Overview of Original Option III Solution

- Detect & Suppress
- Scram to Protect CPR Safety Limit



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Overview of Original Option III Solution

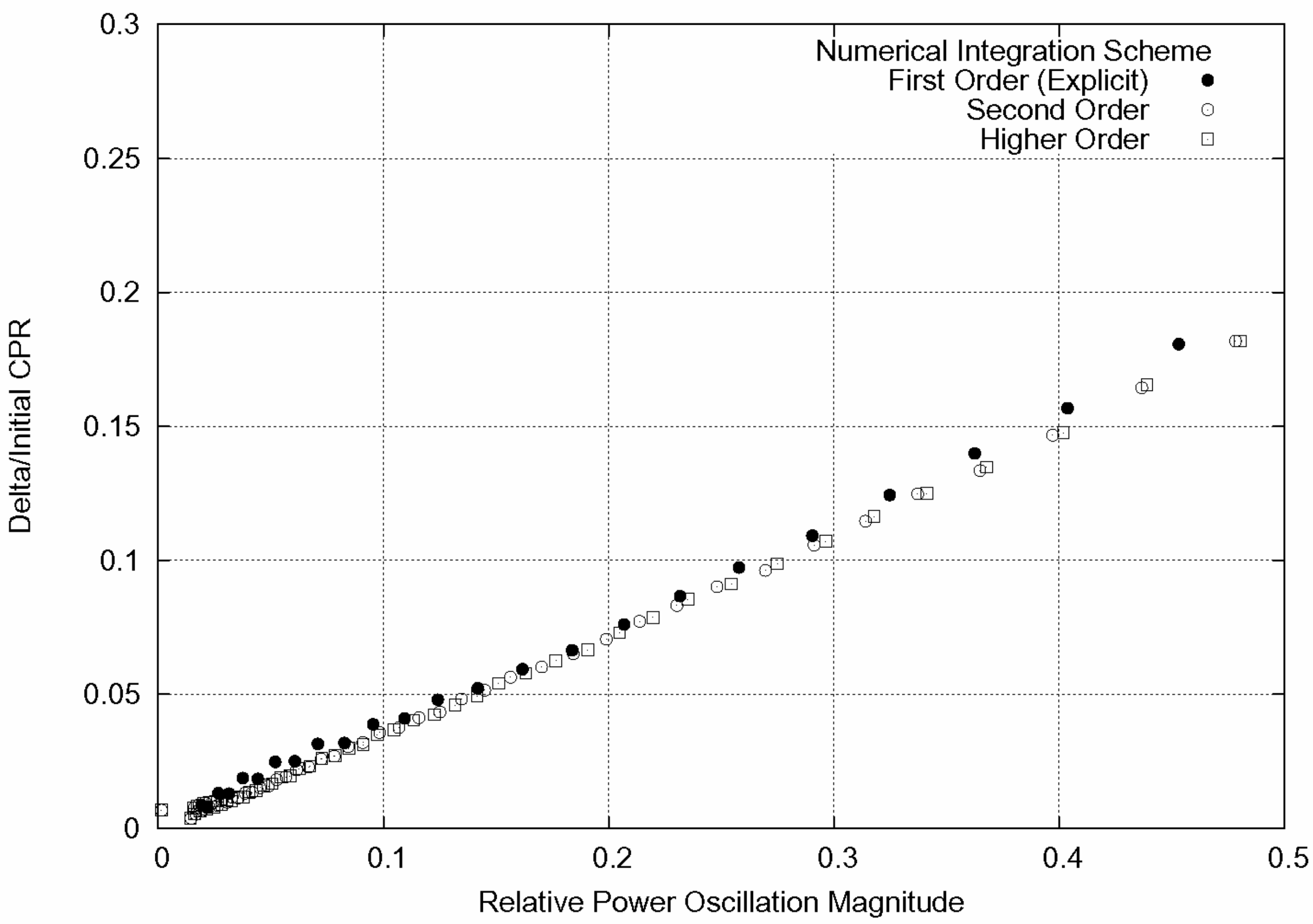
- > System designed to suppress oscillations at a preset amplitude to protect CPR safety limit
- > A relationship between oscillation amplitude and CPR response is required → DIVOM curve
  - Based on relative CPR response versus relative oscillation magnitude
  - Calculated with Time-Domain codes
  - Originally a generic DIVOM is applied

Original Option III Problem and Resolution

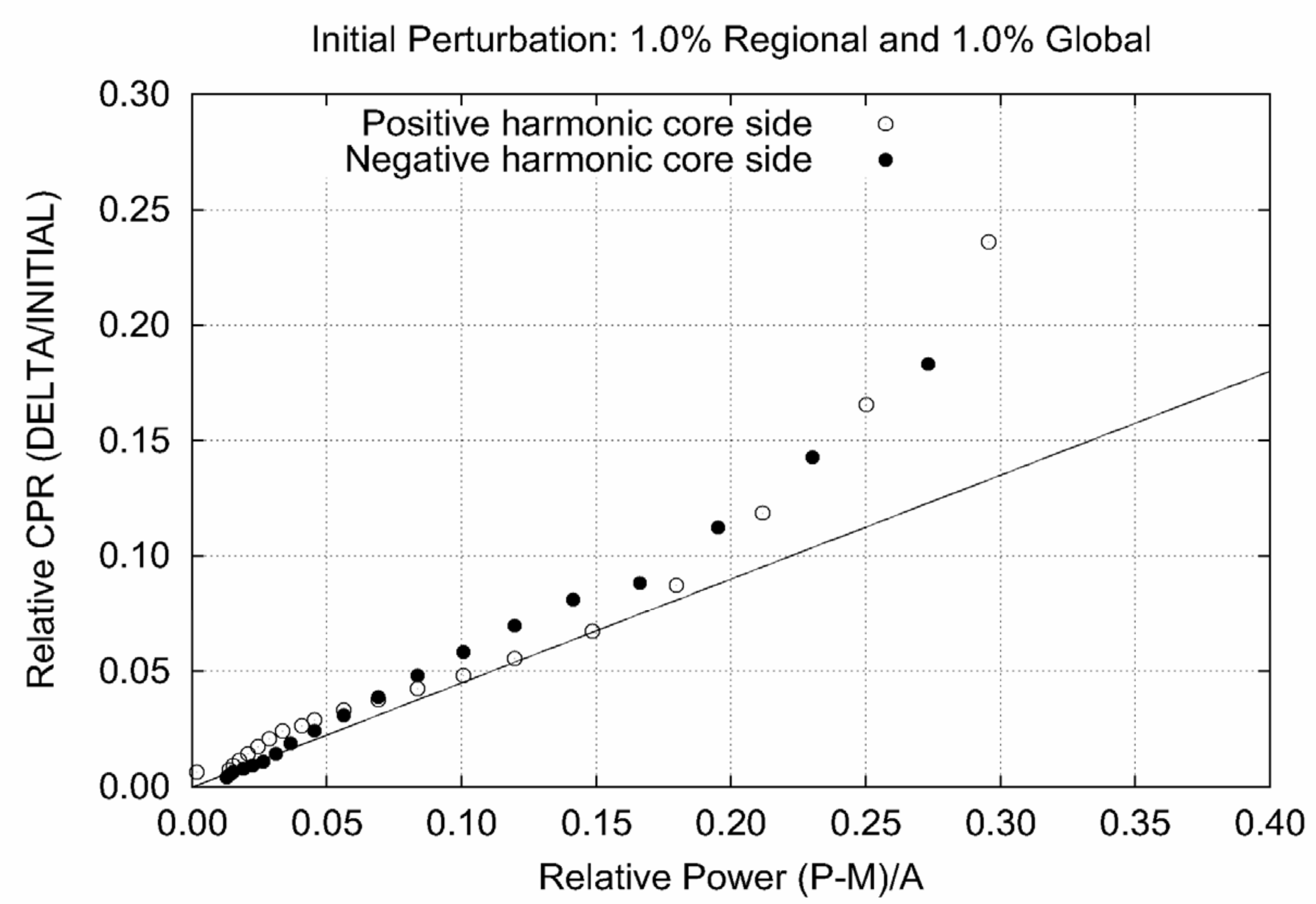
- Part 21 Report: Generic DIVOM curve is non-conservative
  - Occurs at high radial peaking and high power-to-flow ratio
  - DIVOM slope may reach as high as double the generic value

- Resolution
  - Short Term:
    - Cycle-specific DIVOM calculations instead of generic
    - Follow BWROG procedure
  - Long Term:
    - Improved solution not susceptible to DIVOM problems
    - Extend applicability to MELLLA+
  - AREVA long term solution is the Enhanced Option III

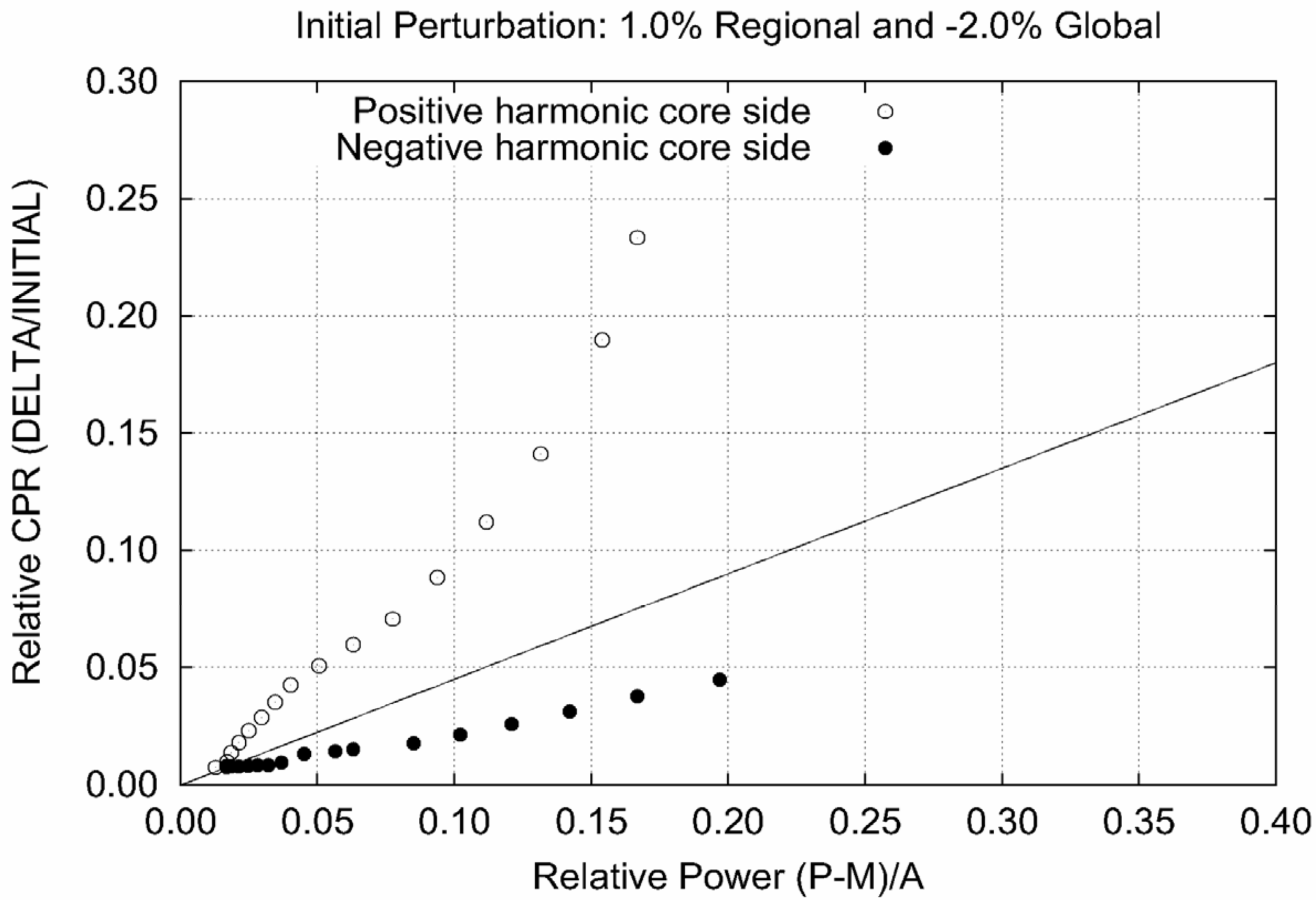
Example of Well-Behaved DIVOM Curve Calculated with RAMONA5-FA



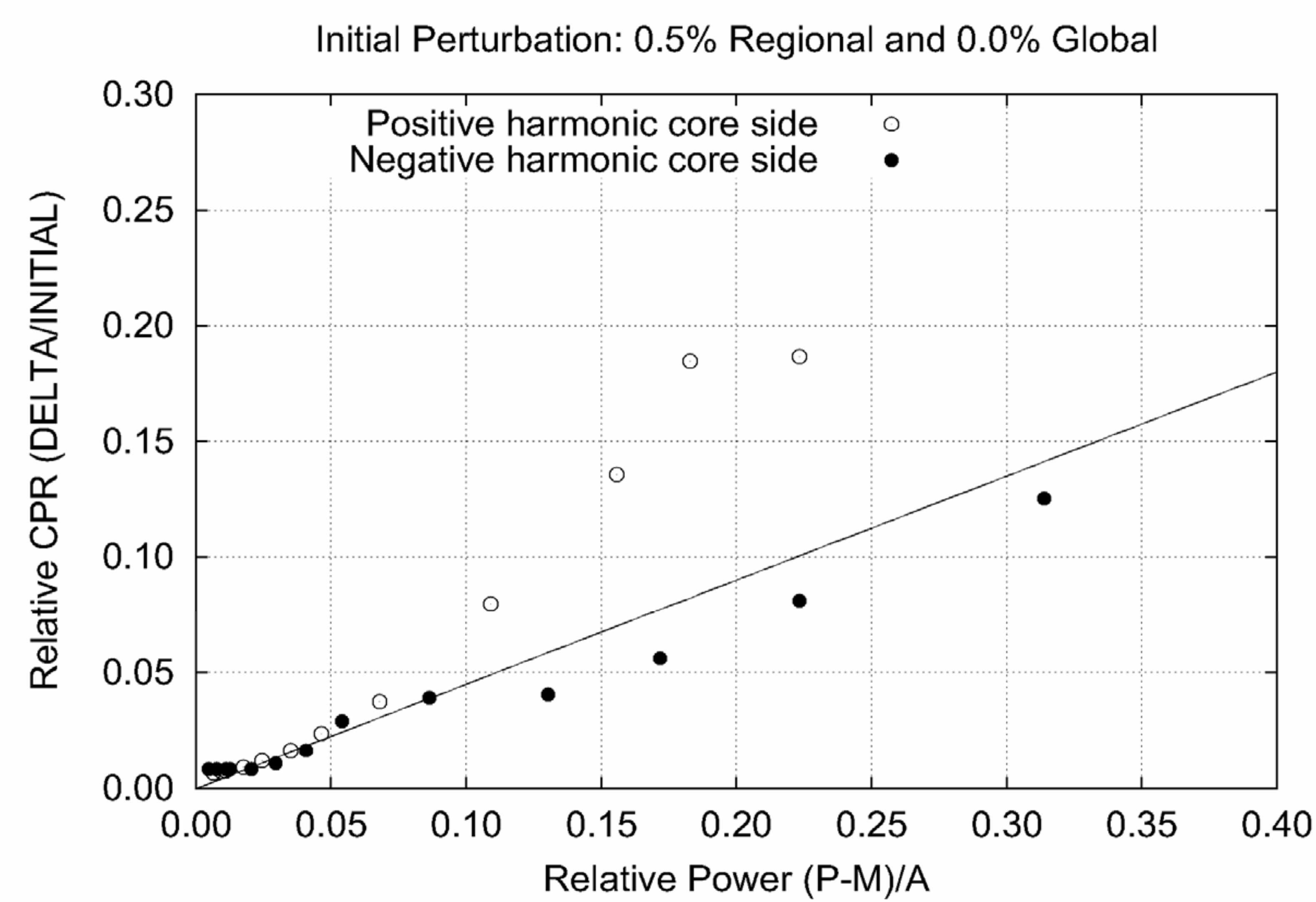
Examples of Irregular DIVOM Curves



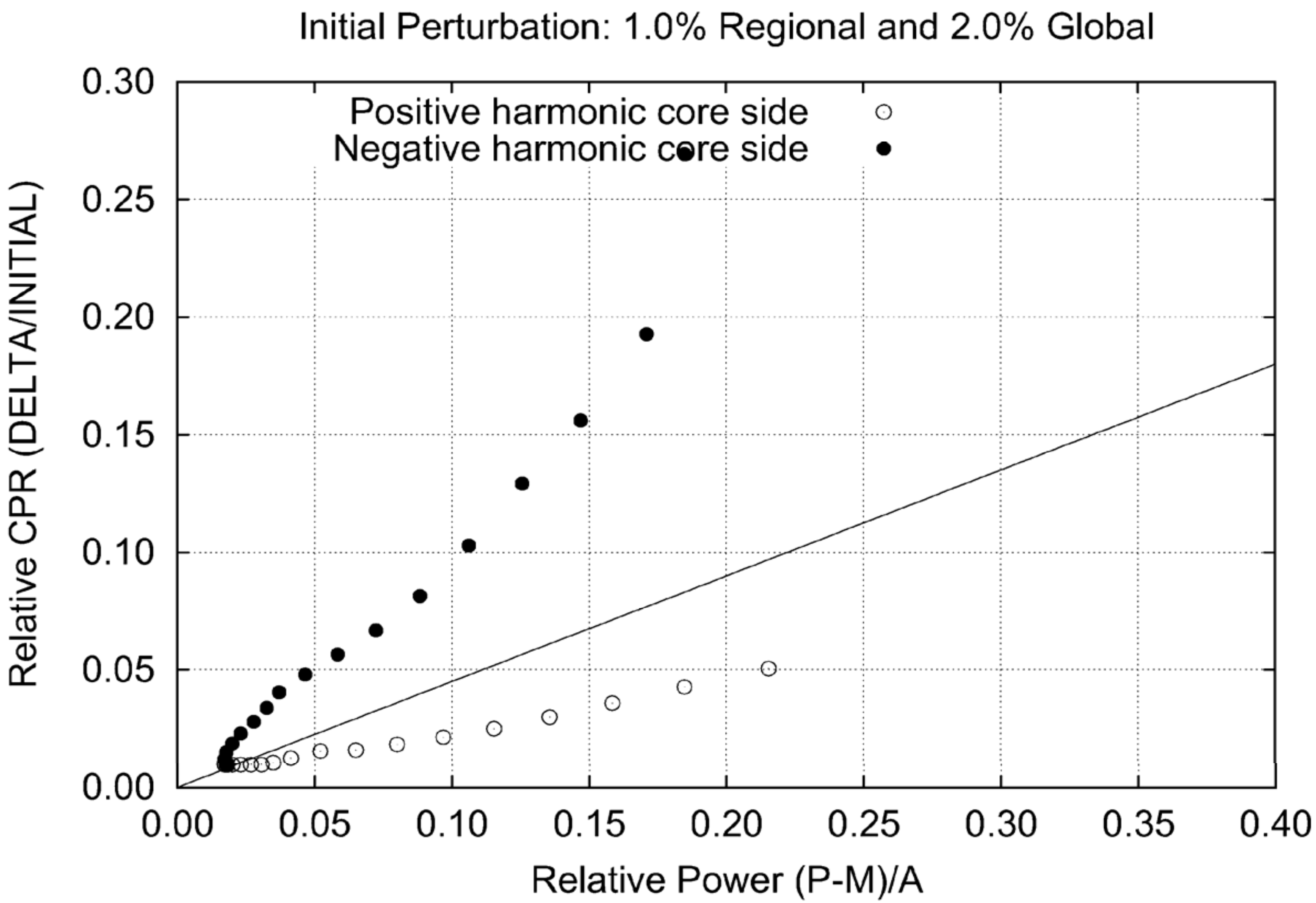
Examples of Irregular DIVOM Curves



Examples of Irregular DIVOM Curves



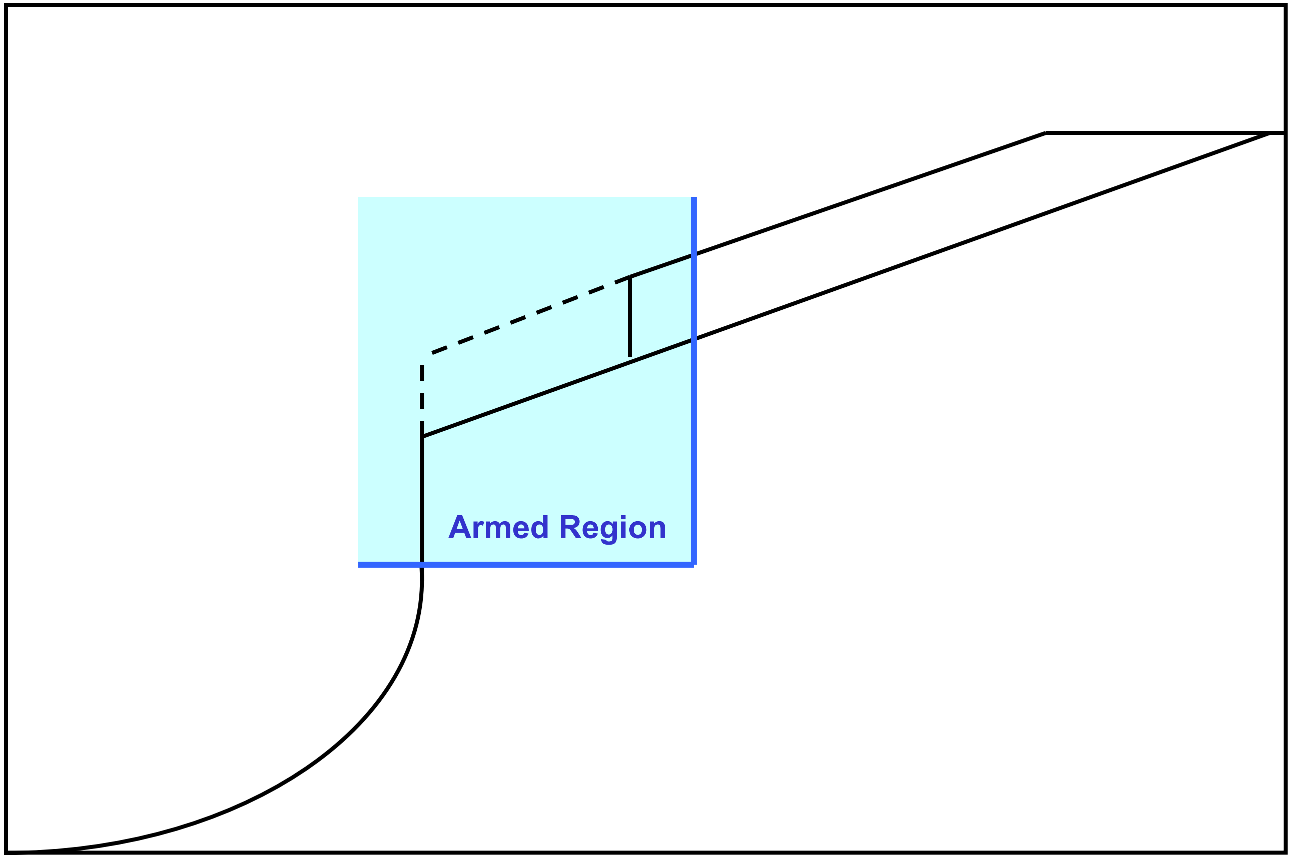
Examples of Irregular DIVOM Curves



Enhanced Option III

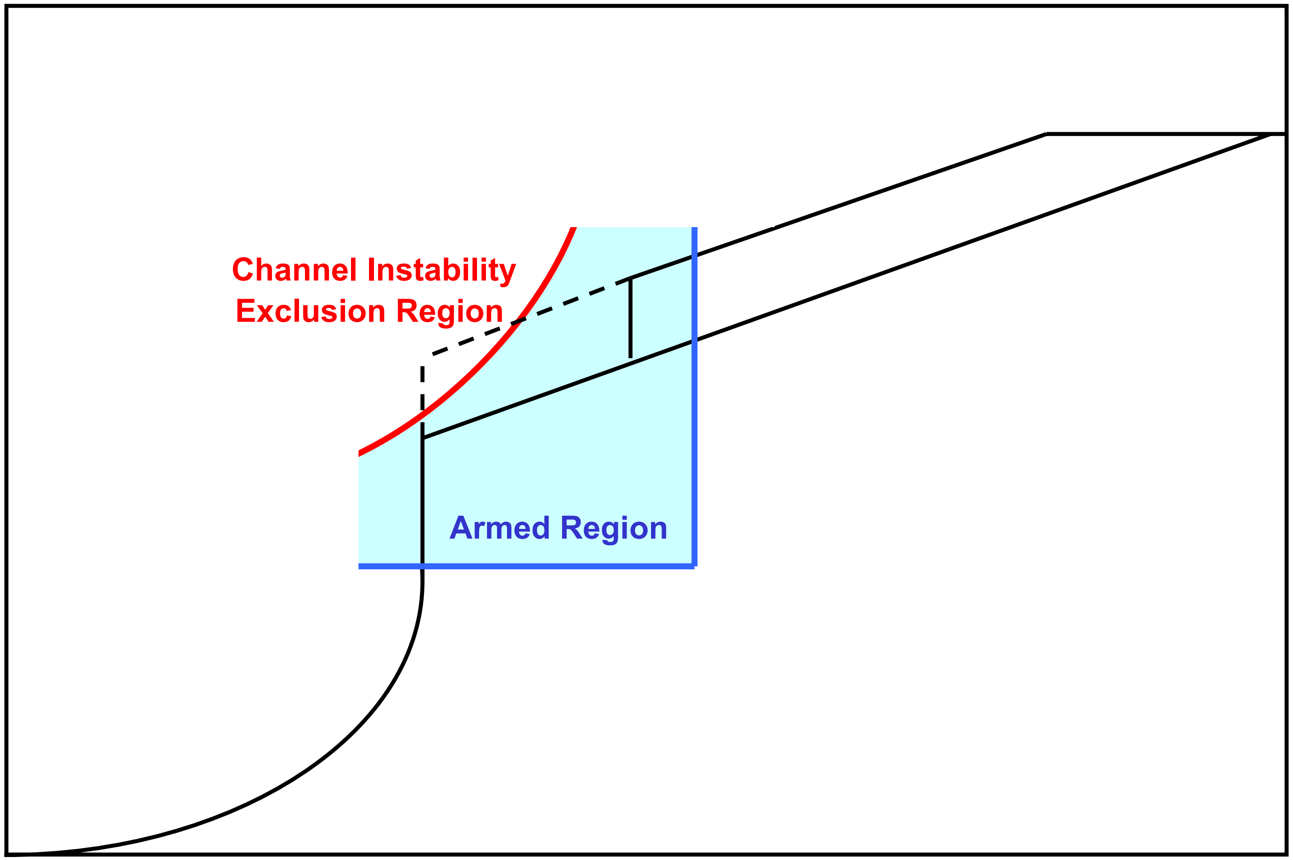
Applicability Domain of Original Option III

- Armed region where instabilities are possible
- Not qualified for MELLLA+



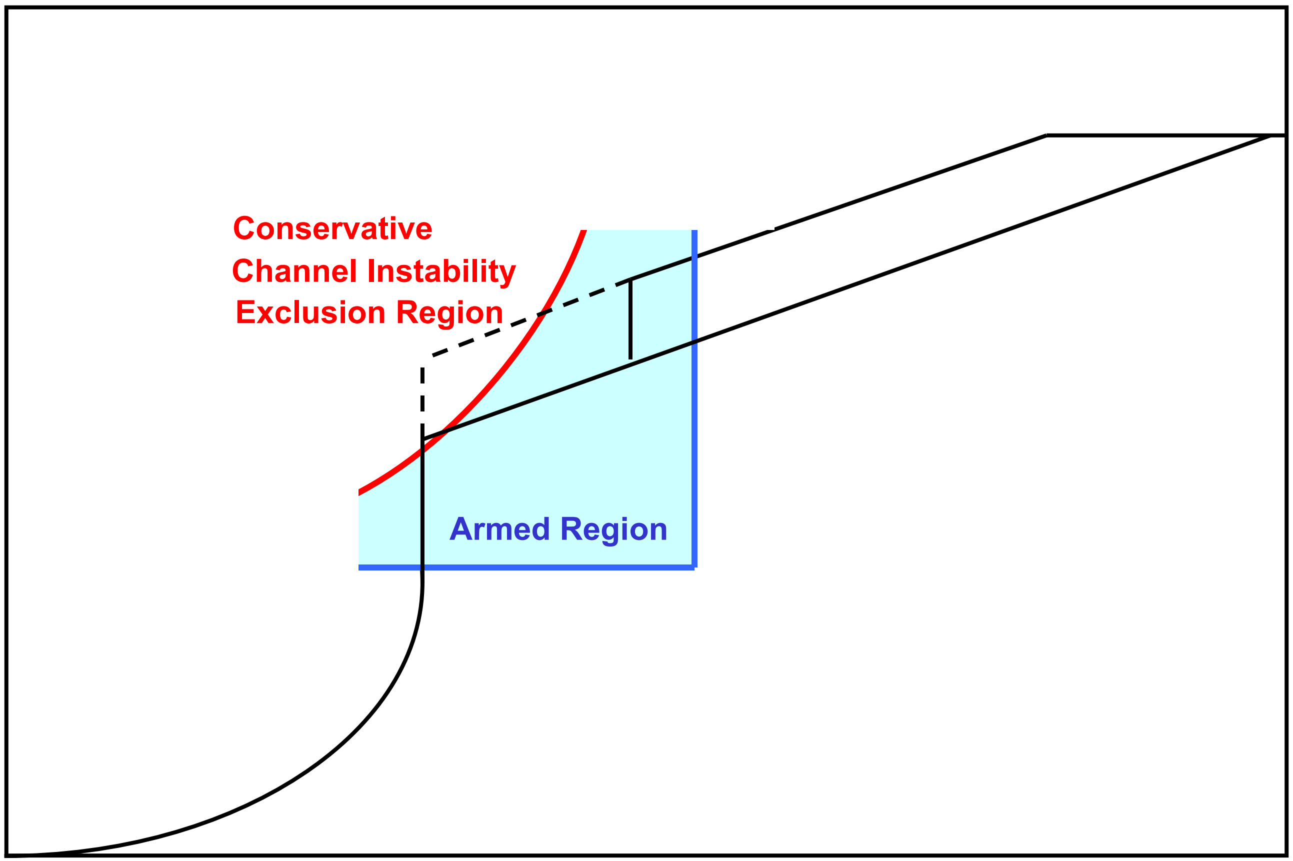
Applicability Domain of Enhanced Option III

> DIVOM problems in the high power/flow corner
> Exclusion region enables extension to MELLLA+



Applicability Domain of Enhanced Option III

> Conservative calculation of exclusion region using STAIF frequency domain code



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Review of the DIVOM Challenge

- DIVOM curve is fairly linear and well-bounded under
  - Conditions:
    - Power, flow, fuel types, loading and control rod patterns…
    - Initial perturbation
  - Modeling methods:
    - CPR correlation, steady state simulator
  - Transient code
  - Exception: Mixed mode oscillations
- Irregular DIVOM curves with elevated slopes observed when single channel decay ratios exceed unity

Elements of the Enhanced Option III Solution

> Define new boundary of applicability of the Option III Solution
  - Exclude conditions for unstable single channel interference
  - Imposing this restriction assures robust DIVOM curves
    - Fairly linear
    - Bounded slope
    - Invariant regardless of initial perturbation

> Protect the single channel instability exclusion region (immediate scram upon entry)

> Maintain all Detect & Suppress functions of Option III outside the channel instability exclusion zone

> Define cycle-specific DIVOM curve for reactor states with all channels stable

Transient System Code for DIVOM: RAMONA5-FA

> Perform Well-Defined Numerical Analyses to Provide Data for DIVOM Relationship

> Studsvik-Scandpower RAMONA5-2.4 ⟶ RAMONA5-FA

- Thermal-hydraulic balance equations unchanged
- Modal Kinetics (similar to STAIF)
- Updated Closing Relations & Correlations (similar to MB2)
- Benchmarking & Sensitivity
  - Integral Benchmarks
  - Separate Effects
  - Hydraulic loop testing

Conclusions
Advantages of Enhanced Option III

1. Maintains the basis of the original Option III solution with many years of operational experience
2. Clear physical basis for the proposed enhancements
3. Channel exclusion region based on approved frequency domain stability code (STAIF)
4. Small channel exclusion region should not interfere with normal operational flexibility

Conclusions
Advantages of Enhanced Option III

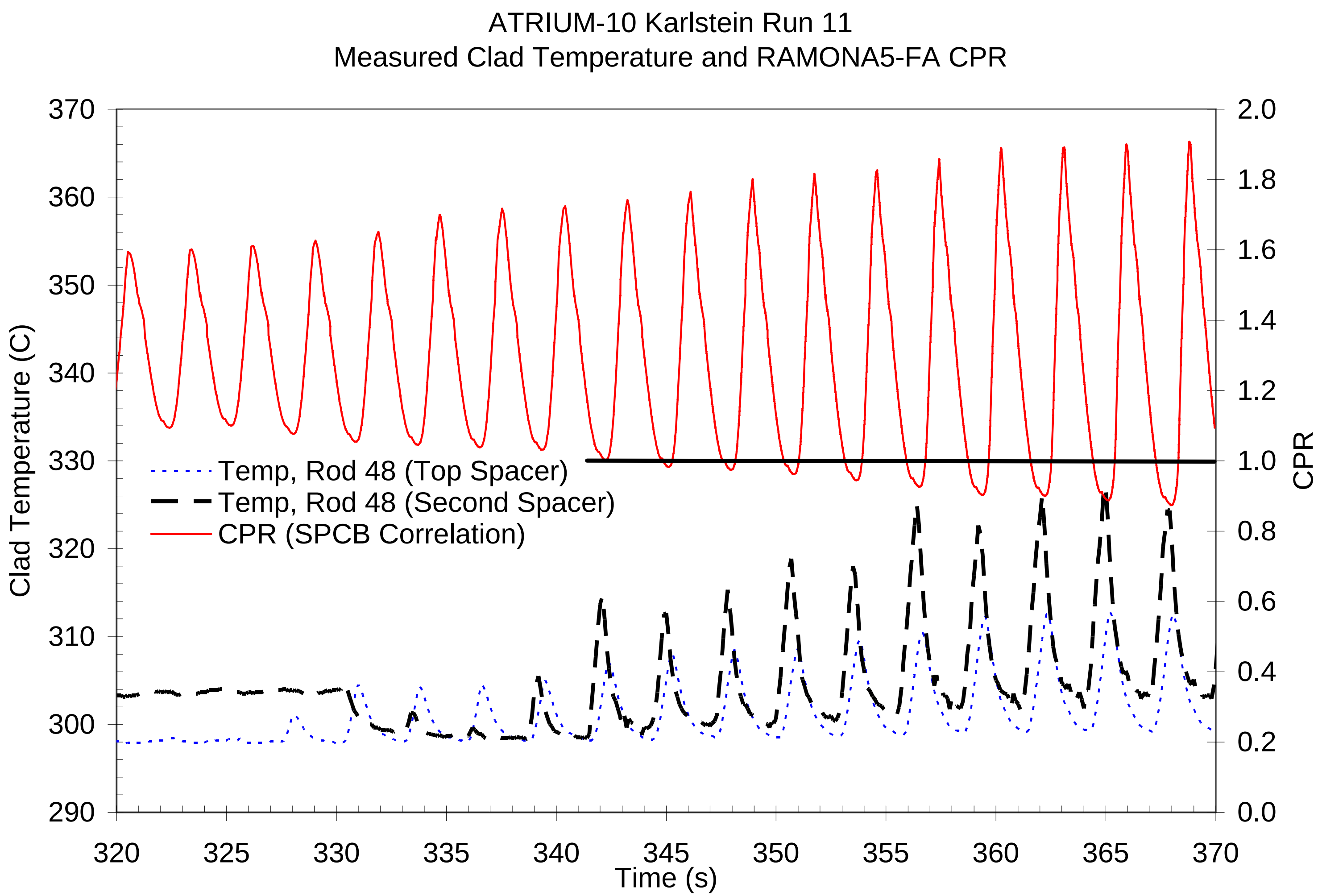
5. **Amplitude trip setpoint will be not be restricted by single channel interference leading to elevated DIVOM slope**
   - **Reduces the probability of spurious scram due to the period-based algorithm response to LPRM noise**
   - **DIVOM curve will always be regular and bounded**
6. **The enhanced solution covers extended flow operating domains up to MELLLA+**
7. **Explicitly addresses single channel instabilities**
8. **Simple application procedure**

Thank You!

Backup slides for additional topics

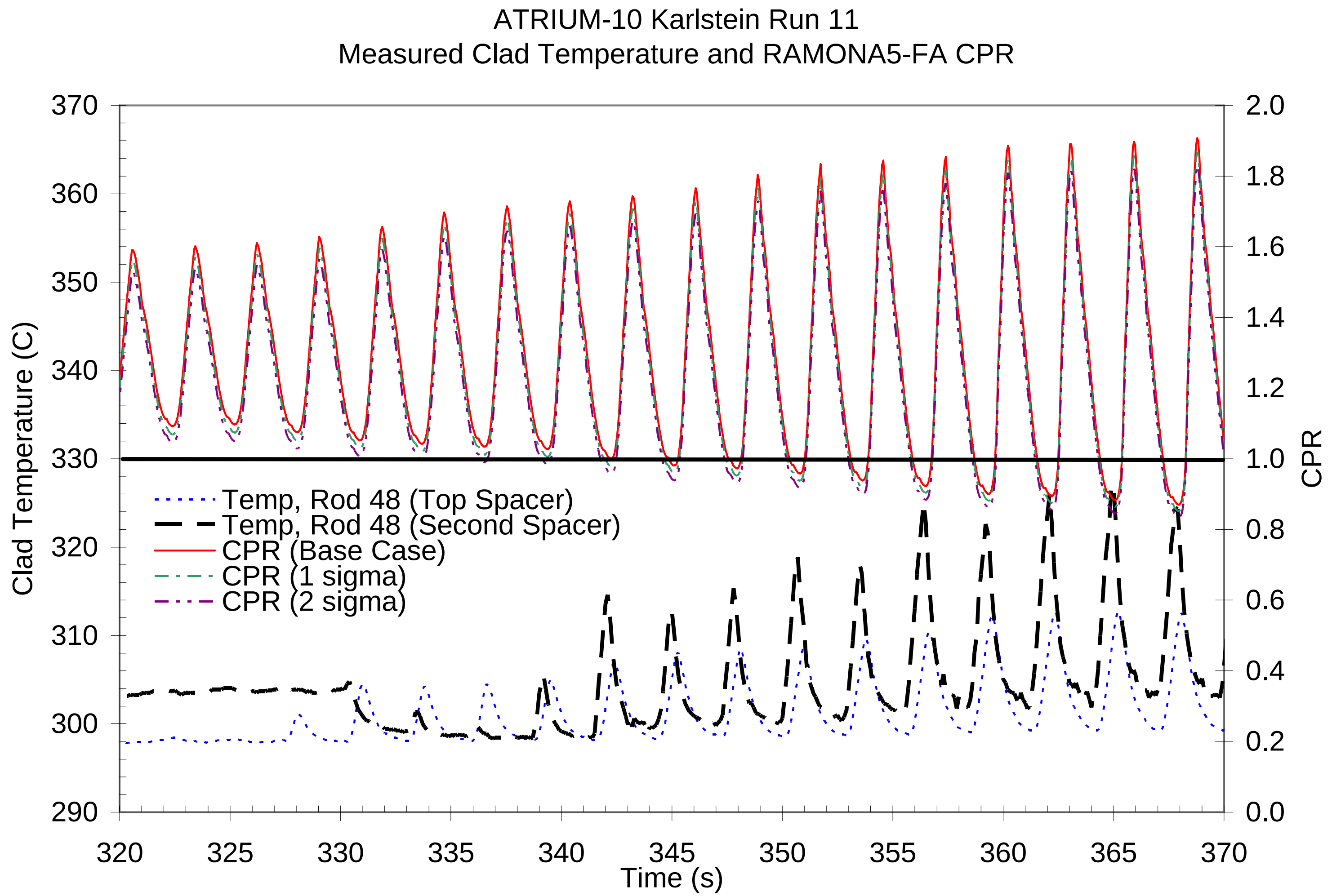
[KATHY Dryout/Rewetting Run 11]

- **[MCPR points coincide with rod temperature peaks]**

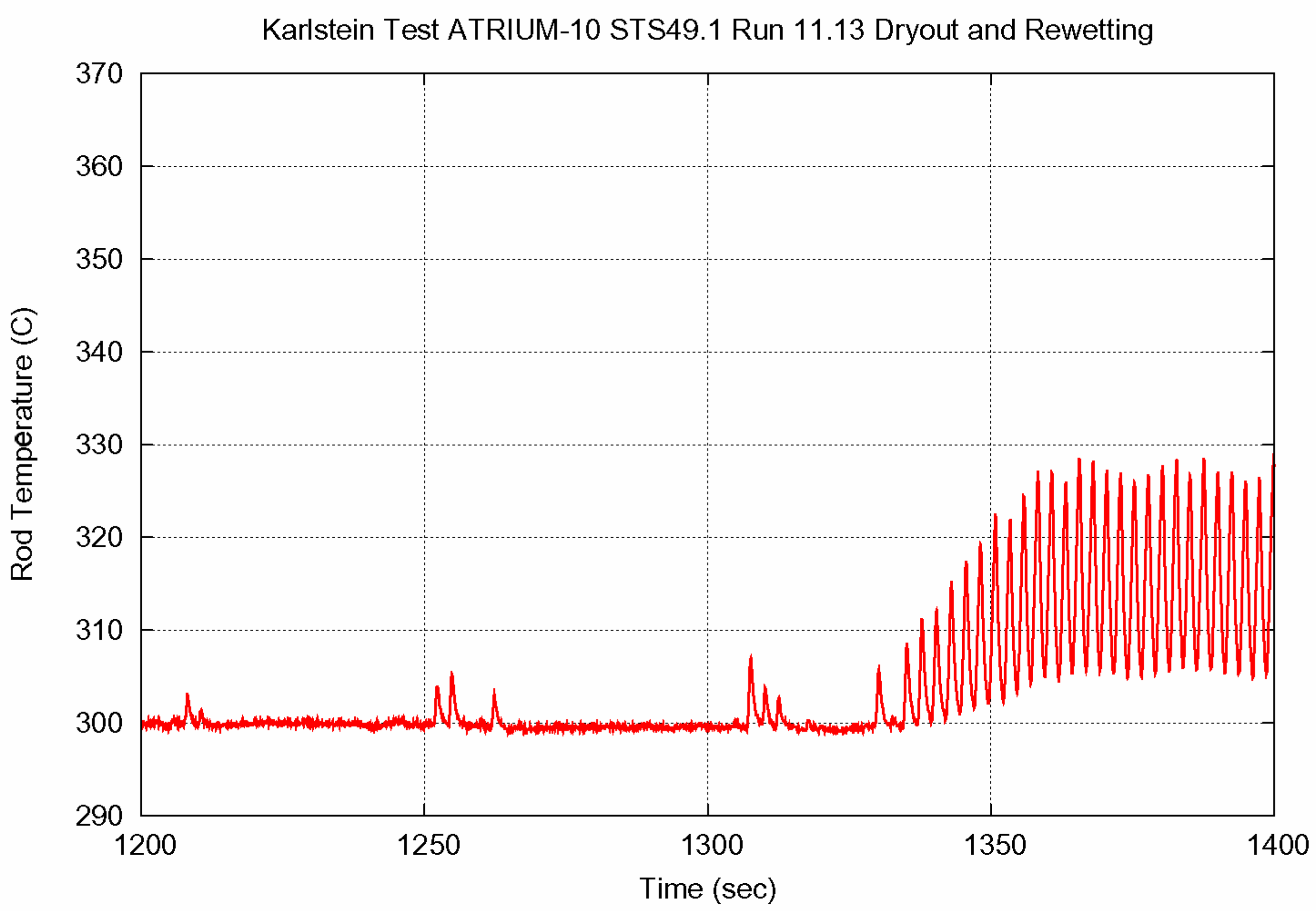


[*KATHY Dryout/Rewetting Run 11*]

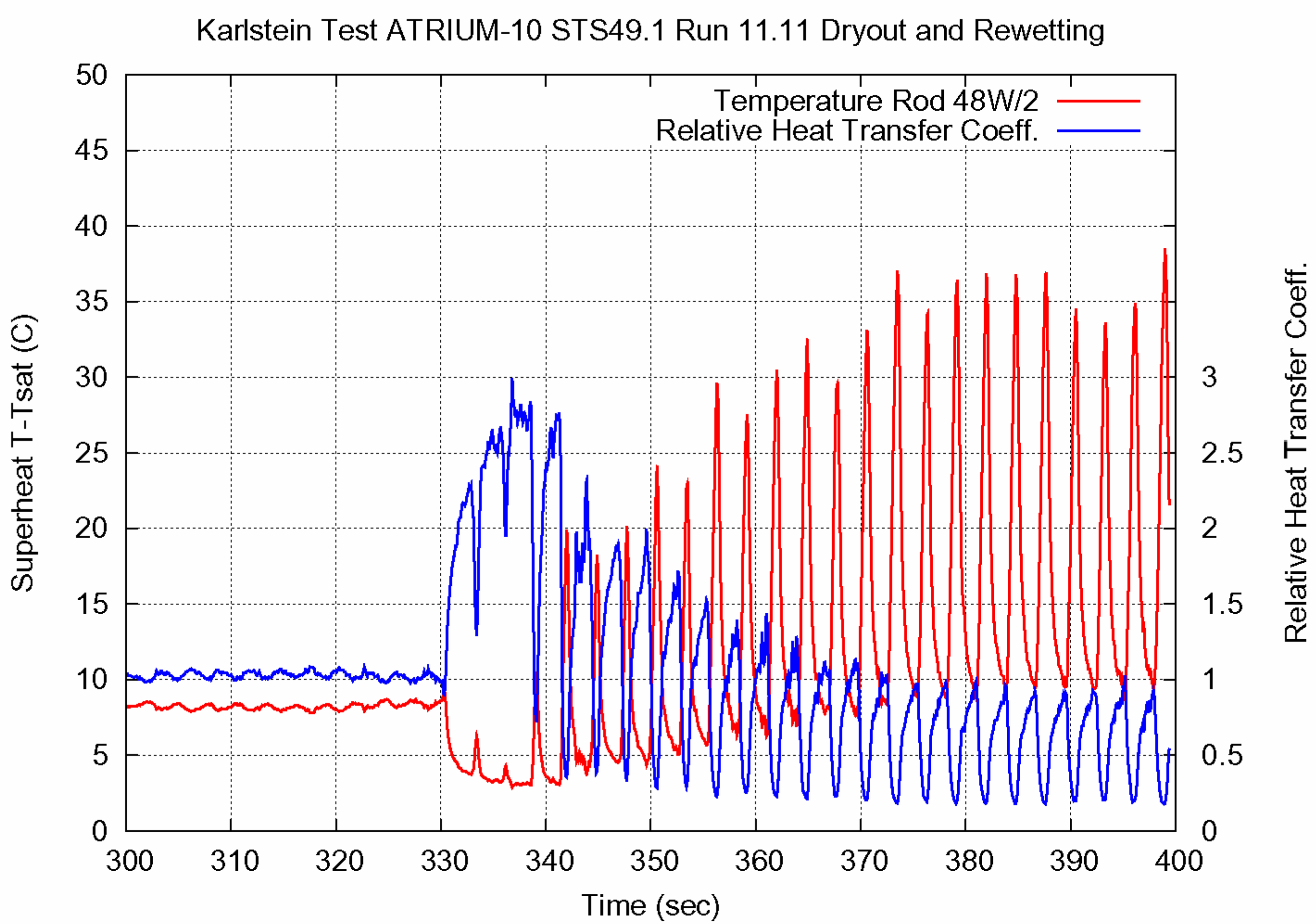
- [Include 1 sigma and 2 sigma uncertainties]



[*KATHY Dryout/Rewetting Non-limiting Rod*]



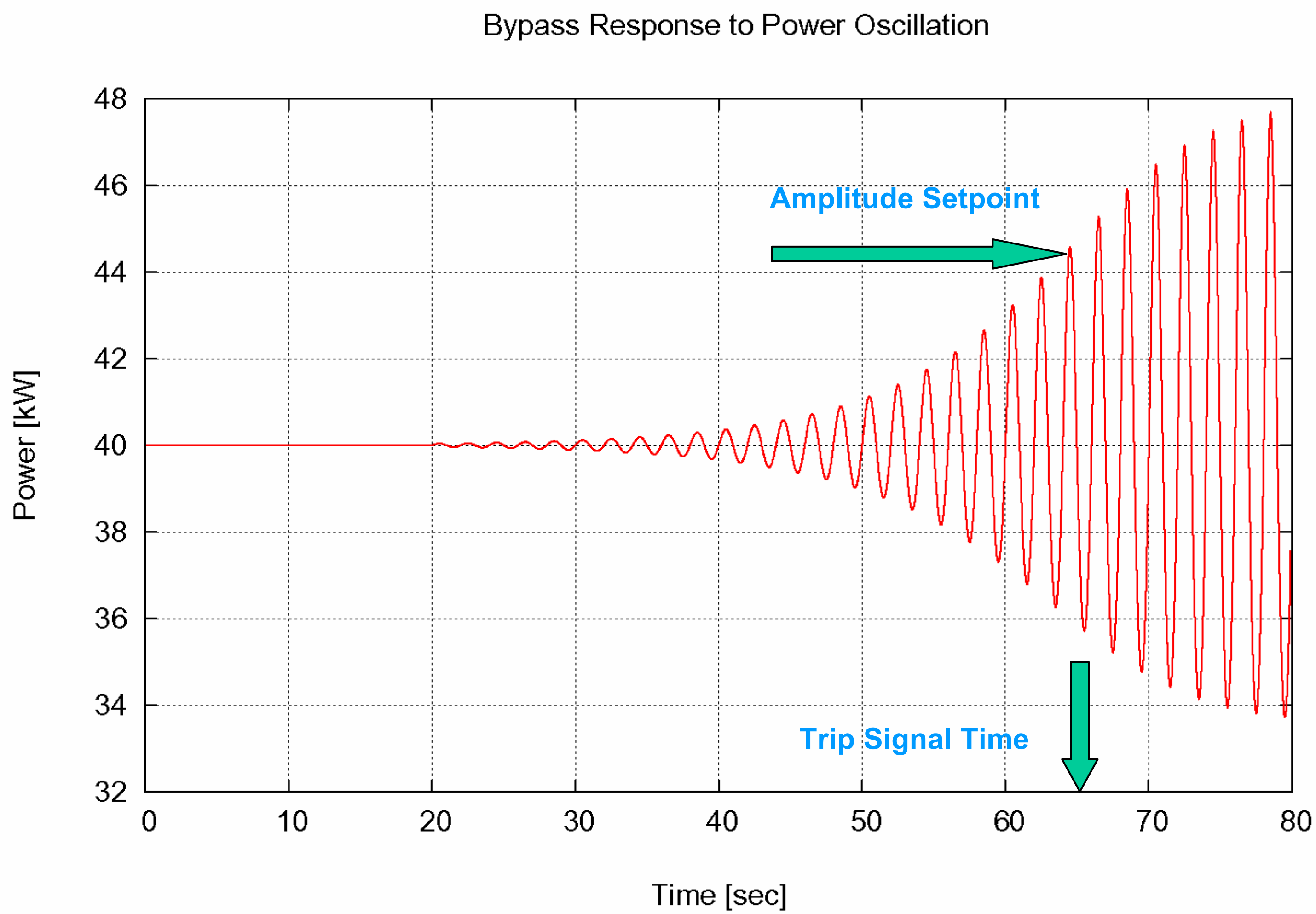
[KATHY Dryout/Rewetting Run 11]



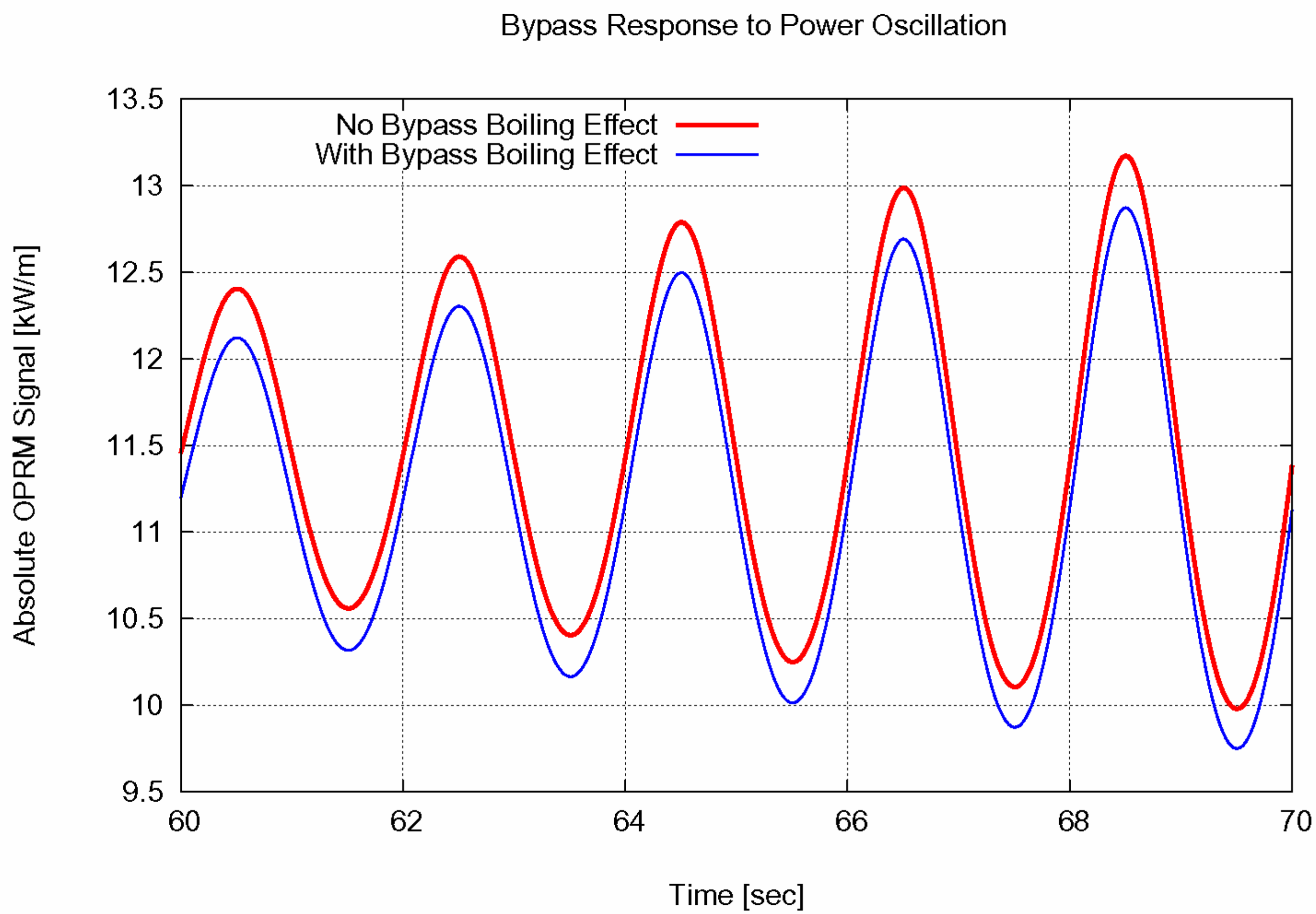
RAMONA5-FA Reactor Stability Benchmarks
Global and Regional Decay Ratios and Frequencies

	Decay Ratio			Frequency (Hz)		
	Measured	Calculated		Measured	Calculated	
		STAIF	RAM5		STAIF	RAM5
CGS Cycle 8 Global Instability Event	1.07	1.081	1.049	0.5	0.575	0.526
GUNC Cycle 1 Regional Instability Test	1.06	1.053	1.070	0.36	0.358	0.345
GUNC Cycle 13 Regional Instability Test	~1.0	0.848	0.806	0.633	0.657	0.635
KKK Cycle 3 Regional Instability Test	>1.0	1.154	1.120	0.40	0.398	0.385

Bypass Boiling Effect



Bypass Boiling Effect



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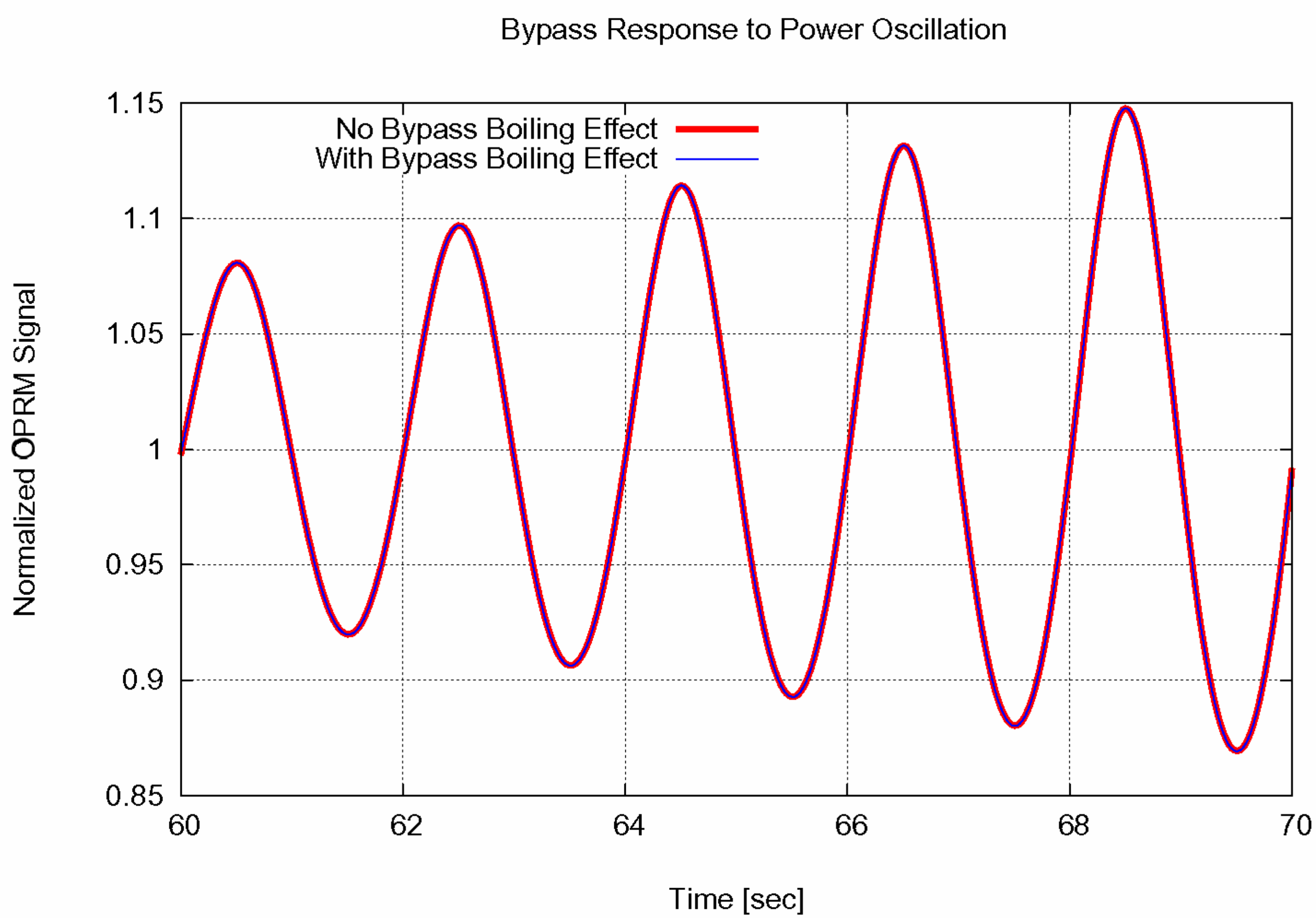
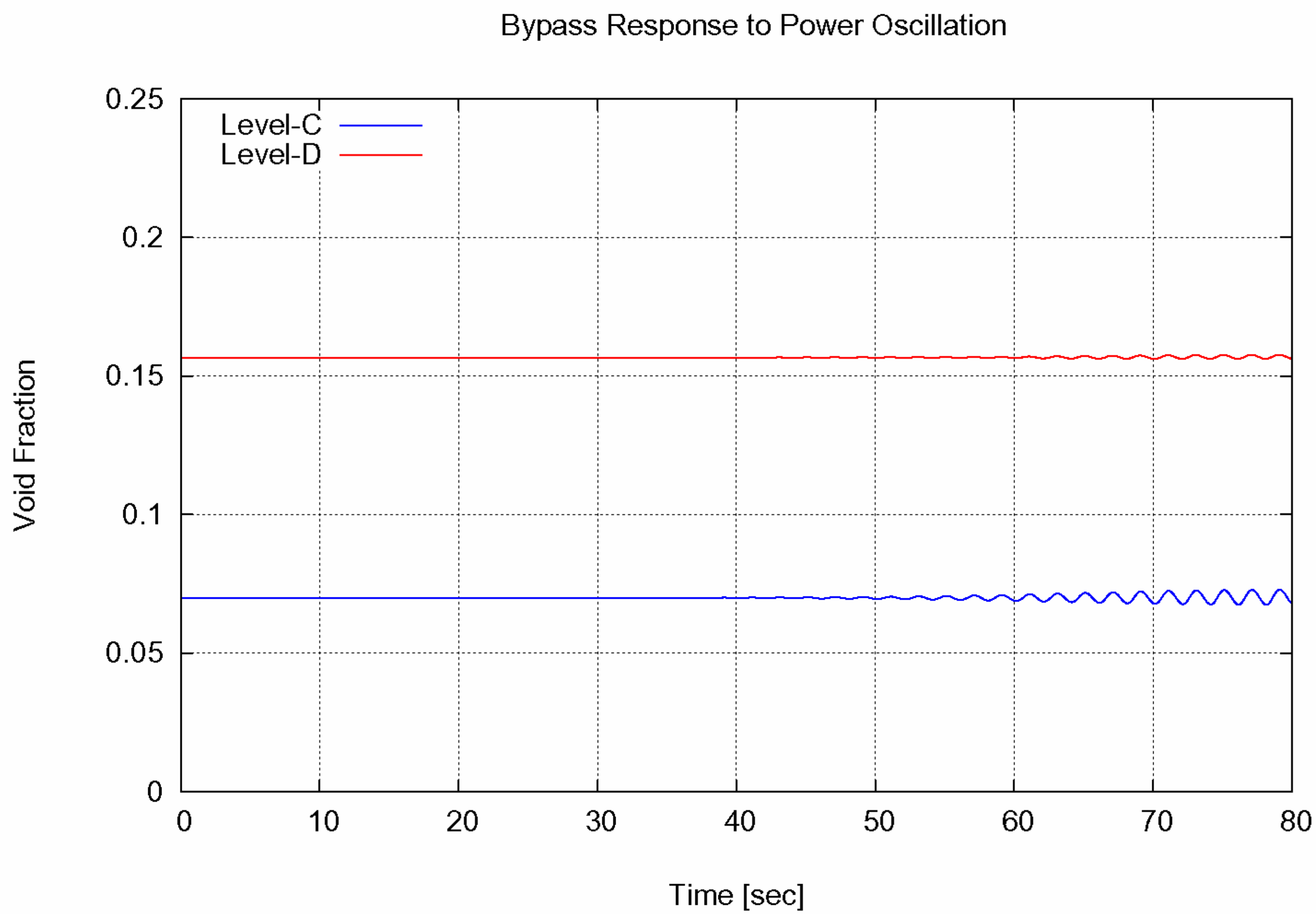
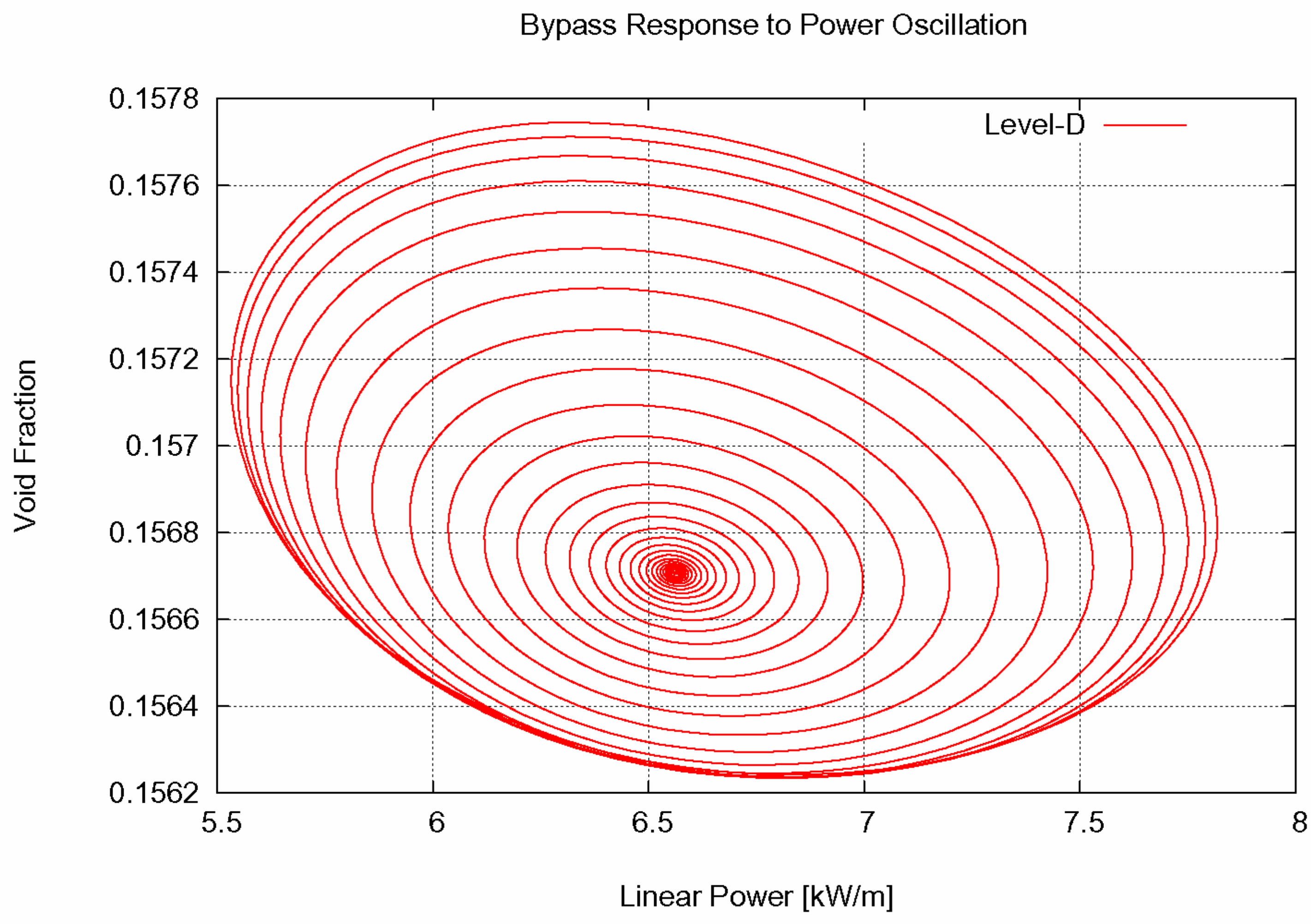


Bypass Boiling Effect



Bypass Boiling Effect



STATE-OF-THE-ART REACTOR CONSEQUENCE ANALYSES

Advisory Committee on Reactor Safeguards Briefing
December 6, 2007

AGENDA

- Project Overview
- Accident Sequence Selection
- Containment System States
- Mitigative Measures
- MELCOR
- MACCS2
- Emergency Preparedness
- Peer Review
- Sample Sequence
- Reporting Latent Cancer Fatalities

SOARCA Objectives

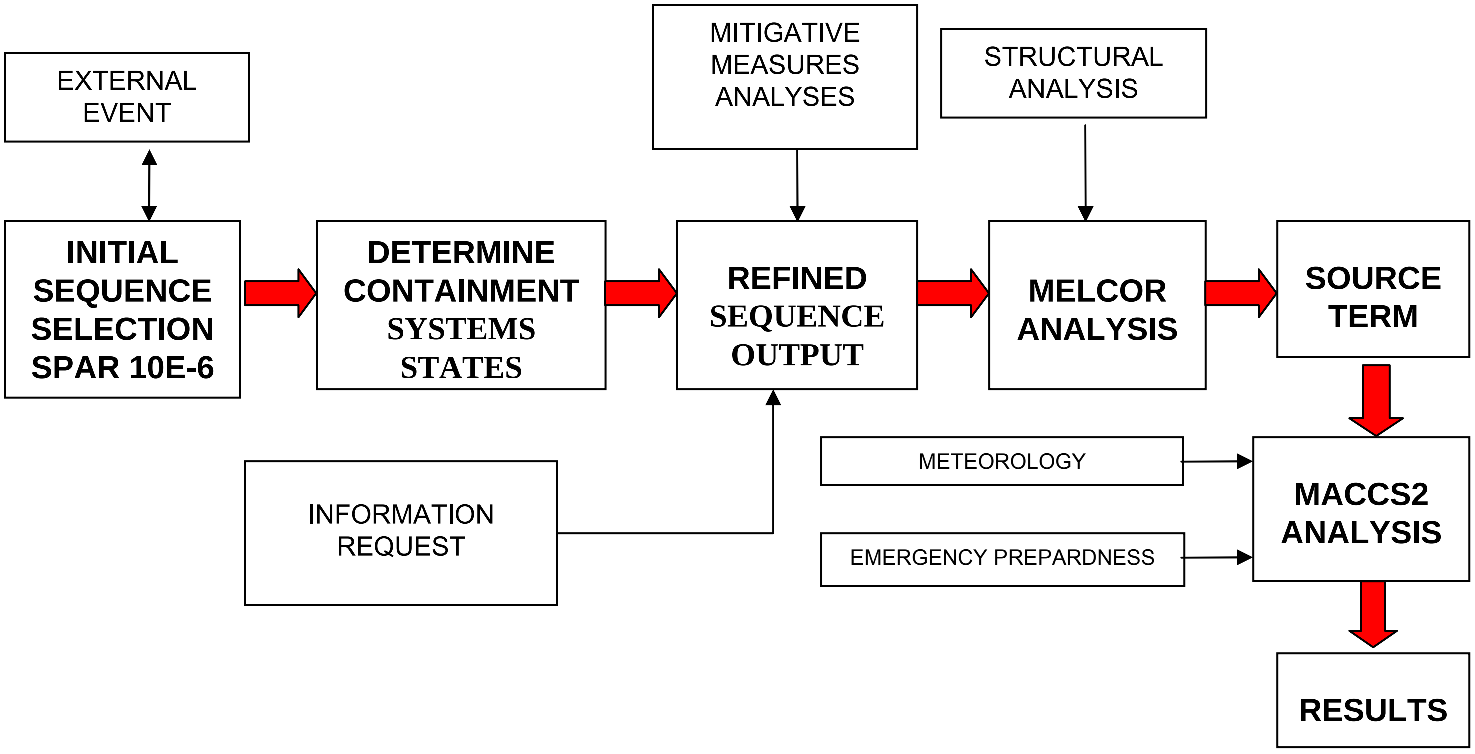
- Perform a state-of-the-art, realistic evaluation of severe accident progression, radiological releases and offsite consequences for frequency dominant core damage accident sequences
- Provide a more accurate assessment of potential offsite consequences to replace previous consequence analyses

Severe Accident Improvements

- 25 years of national and international research
- Regulatory improvements reduced the likelihood of severe accidents
- Improved modeling capability
- Improvements in plant design
- Other plant improvements

SOARCA OVERVIEW

SOARCA PROCESS



SOARCA Approach

- Full power operation
- Plant-specific sequences with a $CDF \geq 10^{-6}$ ($CDF \geq 10^{-7}$ for bypass events)
- External events included
- Consideration of all mitigative measures
- Sensitivity analyses to assess the effectiveness of different safety measures
- State-of-the-art accident progression modeling based on 25 years of research to provide a best-estimate for accident progression, containment performance, time of release and fission product behavior
- More realistic offsite dispersion modeling
- Site-specific evaluation of public evacuation based on updated Emergency Plans

SOARCA Insights

- Sequences dominated by external events, primarily large seismic events (PWR also includes bypass events)
- Previously used sequences have a significantly lower probability of occurrence or are not considered to be feasible
  - Alpha mode failure
  - High pressure melt ejection
  - ATWS
- Mitigative measures are proving to be effective at preventing core damage or containment failure

Sequence Screening Process

(Internal Events)

- Initial Screening - use enhanced SPAR models to screen out low CDF sequences with an overall CDF ≤1.0E-7 and sequences with a CDF <1.0E-8. This step eliminates <10% of the overall CDF (typically about 5%)
- Sequence Evaluation – identify and evaluate the dominant cutsets for the remaining sequences (~90% of initiator CDF). Determine system and equipment availability / unavailability and accident sequence timing
- Scenario Grouping - group sequences together that have similar times to core damage and equipment unavailability
- Select bounding sequences based on most limiting mitigative measures available

Sequence Screening Process

(External Events)

- Identify dominant externally initiated event sequences based upon available probabilistic risk assessment documentation from NUREG-1150, IPEEE submittals, as well as any additional and available supporting documentation
- Identify potential mapping between dominant external events and internally initiated events identified by the SPAR analysis
- Where mapping between external and internal events are not possible or appropriate, a unique externally initiated event or sensitivity study was recommended
- The resulting limited set of scenarios obtained for each SOARCA plant was used for subsequent accident progression and consequence analysis

Containment Systems States

The availability of engineered systems that can impact post-core damage containment accident progression, containment failure and radionuclide release

- Determine the anticipated availability of containment and containment support systems not considered in the Level 1 core damage analysis
- Determine the availability of non containment and non containment support systems such as low pressure injection that can impact containment accident progression

Mitigative Measures Analysis

- The mitigative measures analyses are qualitative, sequence-specific systems and operational analyses based on licensee identified mitigative measures from EOPs, SAMGs, and other severe accident guidelines that are applicable to, and determined to be available during a sequence groupings whose availability, capability and timing will be utilized as an input into the MELCOR analyses

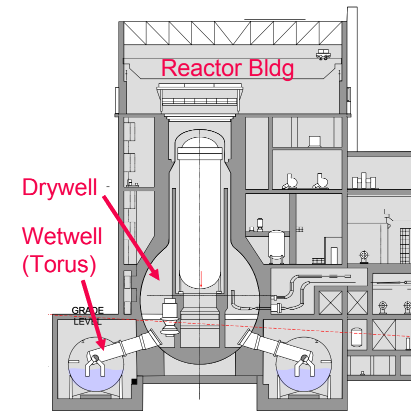
Mitigative Measures Analysis Process

- For those dominating sequences / sequence groupings within the scope of SOARCA, determine the potentially available mitigative measures
- Perform a system and an operational analysis based on the initial conditions and anticipated subsequent failures
- Determine the anticipated availability, capability and the time to implementation
- MELCOR will determine the effectiveness of the mitigative measures based on capability and estimated time of implementation

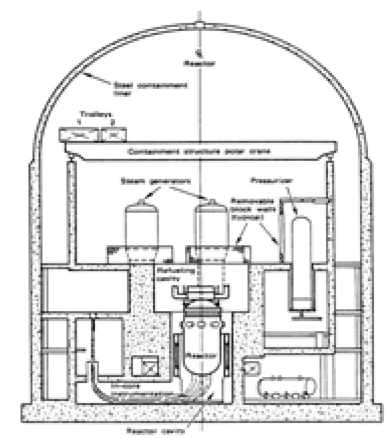
Structural Analyses Objective

Evaluate the behavior of containment structures under unmitigated severe accident conditions to predict the following performance criteria at the selected sites:

- Functional Failure Pressure - Leakage
- Structural Failure Pressure - Rupture
- Develop Leakage Rate and/or Leakage Area as a Function of Internal Pressure



Peach Bottom "Mark I – Steel Containment"



Surry "Reinforced Concrete Containment"

MELCOR Analyses

- MELCOR Code Improvements
  - MACCS2 Output Interface
  - Implement Fuel Collapse Model Logic
  - Update MELCOR Defaults
  - Pool Scrubbing Model
- Develop a plant-specific model
- Perform accident progression analyses for each plant using MELCOR computer code to determine source term, potential containment failure state, and time of release as input in the MACCS2 analyses

MACCS 2 Analyses

- MACCS2 Code Improvements
  - Increased number of evacuation cohorts
  - Alternative models for latent cancer fatality dose response
  - Increased angular resolution
  - More plume segments
  - Enable network evacuation model
  - KI ingestion
  - Evacuation speed modifiers by grid element and for precipitation
  - Enable parameter uncertainty
- Perform consequence analyses for each plant using MACCS2 computer code to determine early fatalities, and latent cancer fatalities

MACCS2 Assumptions

- No contaminated food or water consumed
- Latest federal guidelines used for dose conversion factors
- KI ingestion by half the 0 – 10 mile population, suboptimum timing
- Median values from US/CEC study of uncertainty for non-site specific parameter
- Site-specific population and meteorology
- Projected dose during emergency period, 5 rem relocate in 1 day; 2 rem, 2 days
- Return criteria: 0.5 rem in 1 yr for Peach Bottom, 4 rem in 5 yr for Surry
- In general,1-hr plume segments are used

Emergency Preparedness

- Model the protective response afforded by current site-specific Emergency Preparedness (EP) Programs to improve realism
- Used site-specific evacuation time estimates for evacuation of EPZ
- Used OREMs to model evacuation of 10 to 20 mile area
- Modeled cohort data
  - Population
  - Evacuation timing
  - Travel speed
  - Roadway network
- Data was used in MACCS2 to develop consequence estimates

Peer Reviews

- Internal
  - Staff
  - ACNW&M
  - ACRS
- External
  - National Experts
  - International Experts

Peach Bottom Accident Sequences

- PRA models indicate core damage frequency dominated by seismic event, which is functionally a long-term SBO (1×10^{-6} to 5×10^{-6} /yr)
  - Fire and flood events would be similar in terms of core damage progression
- Internal events were all < 10^{-6}/yr
- Bypass events were very low frequency: <<10^{-7}/yr

Surry Accident Sequences

- Dominant PRA events
  - Long-term SBO ($1x10^{-5}$ to $2x10^{-5}$/yr)
  - Short-term SBO ($1x10^{-6}$ to $2x10^{-6}$/yr)
  - ISLOCA ($7x10^{-7}$/yr)
  - SGTR ($5x10^{-7}$/yr)
- SBO events are due to seismic, flooding and fire initiators, and are modeled as seismic event
  - Internal fire and internal flood events are less challenging, more mitigation available
- ISLOCA and SGTR are due to random equipment failures followed by operator errors

Sample Sequence
Loss of Vital AC Bus

- This sequence was selected and assessed for demonstration purposes, not within the scope of SOARCA, CDF $<10^{-6}$
- MELCOR analysis showed that this event can be mitigated

Sample Sequence
Loss of Vital AC Bus – cont.

- Initiator: Loss of Div IV dc power resulting in
  - SCRAM, MSIV closure, containment isolation
  - RCIC automatically starts, 1 CRDHS pump active
- Operator actions (base case):
  - Load shed to maximize duration of DC power
  - Maximize flow from single CRDHS pump
  - Depressurize RCS at 1.5 hours
  - Secure CRDHS from 4 – 7 hrs to prevent RPV overfill
- Sufficient to prevent core damage

Sample Sequence
Loss of Vital AC Bus – cont.
Insights

- Sufficient injection capability to prevent core damage
  - SPAR does not credit CRDHS for coolant makeup
- RPV depressurization and maximizing CRDHS flow are important operator actions to optimize recovery
- SLC also available for high pressure injection
- Battery duration is important for RCIC operation and instrumentation

Reporting Latent Cancer Fatalities

- Commission Paper
- Options
  - Range of thresholds (0 – 5 rem)
  - Linear no threshold (LNT)
  - Estimate point value from Health Physics Society
    - 5 rem in one year, 10 rem in a life time
- ACNWM Full committee Meeting
  - Presentation on MACCS2
  - Initial suggestions included reporting dose and risk versus consequences
- In staff review

NRC'S SOARCA PROGRAM: UCS CONCERNS

Dr. Edwin Lyman
Senior Staff Scientist
Union of Concerned Scientists

Remarks to NRC Advisory Committee on Reactor Safeguards, December 6, 2007

The bottom line

- The Union of Concerned Scientists (UCS) is supportive of an authoritative and independent study that improves the technical credibility and accuracy of analyses of the consequences of severe reactor accidents
  - Improved protective actions
  - Better siting decisions for new reactors
- However, the “State-of-the-Art Reactor Consequence Assessment” (SOARCA) does not appear to be on track to fulfill such a role
  - Political goals of the project threaten to overwhelm the technical goals

What is the real point of SOARCA?

- Knocking down a “straw man:” the 1982 CRAC2 study (NUREG/CR-2239)
  – Ignores the more than 20 years of refinement of severe accident analyses performed for NUREG-1150 and subsequent work and commonly used in regulatory applications
- “The SOARCA project may show that a LER [large early release] may not credibly exist” --- Randy Sullivan, NRC, ACRS 544th meeting, July 12, 2007

Inappropriate focus on “risk communication”

- “Risk communication:” results “will be presented and documented using risk communication techniques to achieve public understanding”: NRC SOARCA Project Plan, June 2007
- The development of a “risk communication plan” years before the study’s results will be available raises the suspicion that the public relations aspects of this project are its main purpose
  - We say, “Just the facts, ma’am!” – the best way to achieve “public understanding” is to clearly present all assumptions and arguments in a step-wise fashion, so that the impact of the various changes to CRAC2 and NUREG-1150 can be readily observed; not to bundle all of them in a black box that generates an obscure “best estimate”
  - the public should be given the whole picture and the opportunity to make independent judgments of the level of risk it is willing to accept
  - The original CRAC2 risk communication fiasco occurred because NRC was preparing to release only the mean consequence values over the weather sequence distribution ; when the “peak values” were later leaked, it appeared that NRC had tried to conceal data from the public

Excessive secrecy

- Important information about the framework of SOARCA remains secret
  - SECY-05-0233 and the corresponding SRM remain withheld from the public in their entirety
  - The public has been excluded from much of the discussion of SOARCA, in some instances with an apparently inappropriate rationale

SOARCA:
The good ...

- Updated MELCOR accident progression and source term development using reactor-specific data and latest experimental insights
- Improved understanding of containment performance in severe accidents
- More accurate modeling of protective actions

…the bad …

- Improper truncation of low-CDF sequences
  - 10% of CDF screened out --- not insignificant
  - Inconsistent treatment of external events, low-power and shutdown risks
- Credit for unregulated measures like SAMGs

… and the ugly

- Use of thresholds in dose-response modeling would directly contradict the recommendations of established scientific authorities like the National Academy of Sciences BEIR VII Committee:
  - “Mechanistic uncertainties remain, but the weight of available evidence would argue against the presence of a low dose threshold for tumor induction based on error-free repair of initial DNA damage. In summary, the committee judges that the balance of scientific evidence at low doses tends to weigh in favor of a simple proportionate relationship between radiation dose and cancer risk” (NAS, BEIR VII Phase 2, 2006, p. 246).

What was so bad about CRAC2?

- CRAC2
  - used census data from 1970
  - assumed that the entire 10-mile emergency planning zone would be completely evacuated within at most six hours after issuance of a warning
  - assumed aggressive medical treatment for all victims of acute radiation exposure
  - employed a now-obsolete correlation between radiation dose and cancer risk that underestimated the risk by a factor of 4 relative to current models;
  - sampled only 100 weather sequences out of 8760, a method which we find underestimates the peak value occurring over the course of a year by 30%.
- UCS MACCS2 calculations of the consequences of a large, early release using more recent source term (based on NUREG-1465) generally confirm CRAC2 results for Indian Point for early fatalities and find CRAC2 underestimated latent cancer fatalities by a significant factor

Source terms for early containment failure

Source term derived from NUREG-1465

Plume	Release time (hrs)	Duration(hrs)	Energy release (MW)	Kr	I	Cs	Te	Ba	Ru	Ce	La
1	1.8	0.06	28	1	0.4	0.3	0.05	0.02	0.0025	0.0005	0.0002
2	1.86	2	1.6	0	0.27	0.37	0.25	0.1	0.0025	0.005	0.005

Entergy source term for Indian Point derived from MAAP

Plume	Release time (hrs)	Duration(hrs)	Energy release (MW)	Kr	I	Cs	Te	Ba	Ru	Ce	La
1	3.66	22.9	1.08	0.7	0.24	0.23	0.23	0.046	0.09	0.0048	0.0008

MACCS2 results for large early release in 2034

Consequence within 50 miles	UCS result	IP License Renewal Environmental Report (Table E.1-14)
Mean early fatalities	860	Not reported
Mean latent cancer fatalities	38,500	Not reported
Mean population dose (person-Sv)	4.97×10^5	1.58×10^5
Peak early fatalities	70,800	Not reported
Peak latent cancer fatalities	695,000	Not reported
Peak population dose (person-Sv)	7.34×10^6	Not reported

Conclusions

- If the main impact of SOARCA is to reduce potential severe accident consequences by eliminating consideration of large early releases, then it merely will be an exercise in circular reasoning
- Inclusion of thresholds in the dose-response curve used for SOARCA without authoritative technical justification for rejection of BEIR VII conclusions will further undermine the credibility of the report
- An “apples-to-apples” comparison with previous studies will be necessary to truly evaluate the effect of improved technical understanding, better data and code improvement

Recommendations

- The best way to establish the technical credibility of SOARCA is for
  – NRC to immediately submit the methodology and interim results for external, independent peer review
  – NRC to submit the final results for publication in a reputable, peer-reviewed journal

NUREG-1635, Vol. 7

Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program

A Report to the U. S. Nuclear Regulatory Commission

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

NUREG-1635, Vol. 7

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Manuscript Completed: March 2006
Date Published: May 2006

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
U. S. Nuclear Regulatory Commission
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ABSTRACT

This report to the U.S. Nuclear Regulatory Commission (NRC) presents the observations and recommendations of the Advisory Committee on Reactor Safeguards (ACRS) concerning the NRC Safety Research Program being carried out by the Office of Nuclear Regulatory Research (RES). These observations and recommendations focus on that portion of the NRC research program dealing with the safety of existing nuclear reactors and advanced light water reactor designs, such as the Economic Simplified Boiling Water Reactor (ESBWR) submitted for certification. The research strategy for more advanced reactors that are not based on water reactor technology such as the Generation IV reactors being studied by the Department of Energy is also discussed. In its evaluation of the NRC research activities, the ACRS considered the programmatic justification for the research as well as the technical approaches and progress of the work. The evaluation identifies research crucial to the NRC missions. The ACRS also attempts to identify research that had progressed sufficiently to meet current and anticipated regulatory needs so that it could be curtailed in favor of more important activities. This report does not address research on the security of nuclear power plants. Comments on such research will be reported separately. Also, the ACRS does not comment on research activities dealing with nuclear waste issues. The Advisory Committee on Nuclear Waste (ACNW) will report on these research activities.

TABLE OF CONTENTS

TABLES

ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
ACNW	Advisory Committee on Nuclear Waste
ACR-700	Advanced CANDU Reactor-700
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASP	Accident Sequence Precursor
ATHEANA	A Technique for Human Event Analysis
ATWS	Anticipated Transients Without Scram
BWR	Boiling Water Reactor
CAMP	Code Applications and Maintenance Program
CFR	Code of Federal Regulations
CRDM	Control Rod Drive Mechanism
CSARP	Cooperative Severe Accident Research Program
DOE	Department of Energy
ECCS	Emergency Core Cooling System
EMI	Electro Magnetic Interference
EPIX	Equipment Performance and Information Exchange System
EPR	Evolutionary Power Reactor
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling Water Reactor
FY	Fiscal Year
GDC	General Design Criterion
GSI	Generic Safety Issue
HERA	Human Event Repository and Analyses
HRA	Human Reliability Analysis
HSST	Heavy Section Steel Technology
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IASCC	Irradiation Assisted Stress Corrosion Cracking
ICET	Integrated Chemical Effects Tests
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
INL	Idaho National Laboratory
INPO	Institute of Nuclear Power Operations
IPEEE	Individual Plant Examination of External Events
IRIS	International Reactor Innovative and Secure
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
LANL	Los Alamos National Laboratory
LERF	Large Early Release Frequency
LOCA	Loss-of-Coolant Accident
LPSD	Low Power and Shutdown
LWR	Light Water Reactor
MACCS	MELCOR Accident Consequence Code System
MOX	Mixed Oxide

ABBREVIATIONS (Cont'd)

NDE	Non-Destructive Examination
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NIST	National Institute os Standards and Technology
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSIR	Office of Nuclear Security and Incident Response
OECD	Organization for Economic Cooperation and Development
PARCS	Purdue Advanced Reactor Core Simulator
PBMR	Pebble Bed Modular Reactor
PIRT	Phenomena Identification and Ranking Table
PRA	Probabilistic Risk Assessment
PSF	Performance Shaping Factor
PSHA	Probabilistic Seismic Hazard Analysis
PTS	Pressurized Thermal Shock
PUMA	Purdue University Multidimensional Integral Test Assembly
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
ROP	Reactor Oversight Process
SDP	Significance Determination Process
SMIRT	Structural Mechanics in Reactor Technology
SNAP	Symbolic Nuclear Analysis Package
SPAR	Standardized Plant Analysis Risk Model
SRM	Staff Requirements Memorandum
SSHAC	Senior Seismic Hazard Analysis Committee
TRACE	TRAC-RELAP Advanced Computational Engine
UNM	University of New Mexico
U.S.	United States

1 INTRODUCTION

The Nuclear Regulatory Commission (NRC) maintains a Safety Research Program to:

- Ensure its regulations and regulatory processes have sound technical bases.
- Prepare for anticipated changes in the nuclear industry that could have safety implications.
- Develop improved methods to carry out its regulatory responsibilities.
- Maintain an infrastructure of expertise, facilities, analytical capabilities, and data to support regulatory decisionmaking.

These essential missions for the research effort were defined when the NRC was established and there was limited experience with the operation of light water nuclear power plants. The need for research remains today, despite the growth of experience with existing power plants, because:

- Nuclear power plants age and encounter challenges of material degradation not anticipated when the plants were designed.
- The NRC considers applications for extending licenses, uprating the operating power levels of plants, and new plant licenses.
- Reactor fuels are used to higher levels of fuel burnup and new cladding alloys for the fuels are introduced.
- Mixed-Oxide (MOX) fuel is considered for the disposal of excess weapons-grade plutonium.
- The NRC evolves its regulations from a deterministic foundation to a risk-informed basis that makes ever greater use of best-estimate analyses to assess safety.
- New technologies including software-based digital instrumentation and control (I&C) systems are backfit into the existing nuclear power plants.
- New water reactor designs such as the ESBWR, which uses passive systems, have been submitted for certification.

There are on the horizon new power reactor concepts that are not based on the water reactor technologies used in the current fleet of power reactors. The U.S. Department of Energy is studying power reactors that use gas cooling, liquid metal cooling, and molten salt cooling. Reactors that use fast rather than thermal neutrons for fission are being studied with the intent of development. These new reactors make it important for the NRC to consider evolution of its regulatory system from one that is specific to water reactor technologies to one that is not specific to particular reactor technologies, but still lead to adequate protection of the public health and safety. This will require substantial research not only for the early development of technology-neutral regulations, but also, in the longer term, for the development of technology-specific regulatory guidance and plans for reviewing specific license applications.

In this report, the Advisory Committee on Reactor Safeguards (ACRS) presents its observations and recommendations concerning that portion of the NRC Safety Research Program devoted to regulation of existing light water reactors (LWRs) and the certification of advanced water reactor designs submitted for certification such as the ESBWR. The ACRS also makes observations on the need for research in anticipation of more advanced power reactor concepts. Observations and recommendations on research dealing with the security of existing nuclear power reactors and nuclear facilities will be provided in separate reports and are not discussed here. The ACRS does not comment on research activities dealing with nuclear waste issues. The Advisory Committee on Nuclear Waste (ACNW) will address such research separately.

In its review of the NRC Safety Research Program, the ACRS considered the programmatic justification for the research as well as the technical approach and progress of the work. The ACRS supports research that:

- Provides support to the identification and resolution of current safety and regulatory issues.
- Provides the technical basis for the resolution of foreseeable safety issues.
- Develops the capabilities of the agency to independently review risk-significant proposals and submittals by licensees and applicants.
- Supports initiatives of the agency such as the development of "technology-neutral" regulatory systems.
- Improves the efficiency and effectiveness of the regulatory process.
- Maintains technical expertise within the agency and associated facilities in disciplines crucial to the agency mission and that are not readily available from other sources.

This review of the NRC Safety Research Program identifies some research activities that have made valuable contributions to the agency mission in the past, but now have reached the point where additional research is not needed for efficient and effective safety regulation. This review also identifies research activities that could benefit by greater collaboration with research activities elsewhere in the world, including collaboration with researchers in Asia and Europe.

General observations and recommendations concerning NRC research activities are presented in Chapter 2. Observations and recommendations regarding research activities in specific technical disciplines are discussed in detail in Chapters 3 through 14:

- Advanced Reactor Research
- Digital Instrumentation and Control Systems
- Fire Safety Research
- Reactor Fuel Research
- Neutronics and Criticality Safety
- Human Factors and Human Reliability Research
- Materials and Metallurgy
- Operational Experience
- Probabilistic Risk Assessment
- Seismic Research
- Severe Accident Research
- Thermal-Hydraulics Research

2 GENERAL OBSERVATIONS AND RECOMMENDATIONS

The NRC Safety Research Program is largely focused on addressing near-term regulatory needs of the agency. Current activities are especially concentrated in three disciplines:

- Materials and Metallurgy
- Probabilistic Risk Assessment
- Thermal Hydraulics

This is an appropriate focus of the current NRC research activities. These activities are discussed further below along with other major aspects of the research program.

The incident at the Davis-Besse Nuclear Power Plant has emphasized, among other things, how important it is for the agency to have a better understanding of the corrosion of metallic systems in the aging fleet of currently operating nuclear power plants. Aging degradation research is necessary to ensure effective aging management for plants operating for extended periods under license renewal and to assess the effect that operation under extended power uprate conditions may have on margins against degradation. Continued challenges posed by stress corrosion cracking of steam generator tubes in pressurized water reactors (PWRs) and systems within boiling water reactor (BWR) vessels further support such focus in the research effort.

Probabilistic risk assessment (PRA) is the basic technology for the risk-informed regulatory system envisaged by the Commission. Research activities are focused now largely on the application of current PRA technology to reactor regulation through the Reactor Oversight Process (ROP). PRA insights are essential to develop and implement revisions to such central regulations as 10 CFR 50.46. They also will play a key role in the development of "technology-neutral" regulatory systems that will have applications to power reactors that are not based on the LWR technology used in the current fleet of operating plants.

The Standardized Plant Analysis Risk (SPAR) models are fundamental tools for risk-informed regulation. A stronger commitment should be made to the improvement of these models and their extension on a timely basis to include fire, seismic, and shutdown risks. The development of these capabilities for the SPAR models will not only provide a regulatory capability but will also encourage industry to more aggressively develop their own capabilities in these areas.

The quality of PRA results depends on good phenomenological models and there are important areas where such models still need further development. Approximate and often bounding risk analyses done for individual plants suggest that the risk of core damage as a result of events initiated by fires can be comparable to risks from other accidents initiated during normal operations. It is important to know if similar results would also be obtained using fire risk assessments of sophistication comparable to the risk assessments possible for normal operations. Such a finding would have ramifications on both regulatory attentions and licensee attentions to safety. The ACRS continues to believe that based on the potential risk significance of fires, fire safety research merits strong consideration in the NRC research program. The collaboration with Electric Power Research Institute (EPRI) is providing a good understanding of the current state-of-the-art methodology for fire risk assessment. This work provides a basis for determining the need for further development.

Thermal hydraulics is a fundamental feature of safety analyses of nuclear power plants. The NRC allows licensees to do either bounding or best-estimate analysis of plant thermal hydraulics for design basis accidents.

Confirmatory review of licensee analyses requires that the agency have high quality thermal-hydraulic analytical tools. Need for such tools is even greater for the analysis of advanced light water reactors that rely on passive systems to achieve safe configurations following accidents.

NRC has consolidated several models of the thermal-hydraulic transient analysis codes into a single code called TRACE. The TRACE code should be subjected to an independent technical review to assess its range of validity. The TRACE code then should be at a point at which it can be used as the primary thermal-hydraulic tool for regulatory analyses. A plan should be developed for its integration into the regulatory process. This integration will require strong support from the management of the NRC user organizations since such a change in the short run will create additional burden on the staff.

The potential for blockage of sump screens by debris dispersed into the sumps during depressurization of the reactor coolant system during an accident remains an unresolved issue. The complexity of the interactions between fibrous and particulate debris, as well as the chemical interactions that can occur among debris materials and solutes in the coolant, make predictions of blockage and consequently screen size requirements difficult. Research needed to reach a prompt resolution of this issue should receive the required resources.

International Collaboration

Reactor safety is an international undertaking. It is important that there not be great differences in safety regulations among the nations making major use of nuclear power generation. The NRC research is making good use of collaborations with other countries on reactor safety research. Much of this collaboration has been in the nature of information exchange. Such information exchanges are important and should continue to be encouraged and supported. They provide access to information and a kind of peer review that might not otherwise be obtained. However, there are other important cases where NRC has gone farther and formally partnered with other countries to leverage resources for experimental investigations of important reactor safety research issues. Such collaborations are especially noteworthy in the disciplines of reactor fuel research and in severe accident research. The combined resources of the partners in these collaborations are yielding higher quality and more extensive results than would be possible in research programs sponsored by individual countries.

Other areas of NRC research could benefit from more extensive collaborations. Such areas include fire safety research and thermal-hydraulics research. The benefits of such collaborations become more apparent as NRC moves to more realistic analyses which may require validation by costly large-scale, integral tests. Collaborations of this type may become even more important in the future as new types of reactors are proposed for certification internationally. To be effective and efficient in dealing with future challenges, NRC should look for opportunities to increase significantly collaboration with other countries. The ongoing collaborative efforts are very extensive with European countries. More collaboration with Asian countries having active nuclear power plant programs should be pursued.

Support for Future Licensing Activities

There has been a recent resurgence in interest in the use of nuclear reactors for the generation of electrical power. Innovative reactor designs are being suggested to sustain uranium resources and to generate electrical energy at much greater efficiency. The U.S. Department of Energy is studying very high temperature gas reactors, supercritical water reactors, sodium-cooled reactors, lead-bismuth cooled reactors, and molten salt cooled reactors. Some of these reactors will use fast neutrons rather than moderated neutrons for fission. These reactors use technologies quite different than those used for the currently operating fleet of reactors. The current regulatory framework is not well suited for the licensing, regulation, or monitoring of such different reactor technologies. Several years ago, it appeared that a substantial portion of NRC resources might need to be devoted to the development of the capabilities to address these very advanced reactor technologies. Today, this is not the case. NRC advanced reactor research resources are focused on addressing issues associated with advanced water reactors such as the ESBWR and the EPR.

This seems to be an appropriate use of NRC's limited resources for advanced reactor safety research. Very advanced reactor concepts have not reached a sufficient state of development that productive use of regulatory research resources can be made. However, work should continue on the development of a technology-neutral framework for regulation, although the development of technology- specific guides can be delayed until it is clearer which alternate reactor technologies will be of the greatest interest.

Development of the framework is not only important for the licensing of non-light-water reactors, but also may provide insights that are useful in developing a more efficient regulatory program for advanced reactors of all types.

There are some indications that certifications may be sought for advanced designs with minimal experimental study of plant response under accident conditions. NRC needs to provide clear guidance on its expectations for the experimental validation of computer models used in the licensing of advanced reactors that do not use familiar technologies. Development of such guidance is an area of advanced reactor research that can be pursued at relatively low cost, but which can play an important role in timely and efficient licensing of advanced reactors with new technologies.

Opportunities for Independent Research

In recent years, a strong effort has been made to ensure that NRC research is supportive of the needs of the line organizations. Focusing NRC research entirely on the immediate needs of the line organizations does, however, entail an important risk. This focus reduces the opportunities for independent thought by the research staff and the opportunities to conduct research that could make more dramatic improvements in the regulatory process, for example, in the tools that support it at a time when there is a rapid increase in workload. The risk is magnified by the diversion of so much research talent to address issues of security of nuclear facilities. There is the further risk of a loss of prestige in the research program focused as it is on issues of implementation. This could eventually lead to a loss in the credibility of the technical basis that underlies regulatory decisions.

It is important that NRC research stay abreast of technological developments that can enhance safety. Areas where developments in the larger technical community can be important to the NRC include reactor fuels, corrosion and materials degradation, man-machine interfaces, technologies for monitoring component performance, inspection techniques, and virtual facility inspections. Where NRC can adopt or adapt developments in other industries, safety can be improved and the efficiencies of NRC reviews enhanced.

One mechanism for RES to interact with the larger technical community is by sharing its own research plans. This has been done for research into digital instrumentation and control. Investigators did creditable reviews of the state-of-the-art, presented them at appropriate professional society meetings as a kind of public peer review, and developed from these state-of-the-art reviews a research plan that is well directed to address agency needs. Sharing research plans with a larger technical community is a strategy that would benefit other NRC research activities. Such interactions also help provide visibility for and help sustain the prestige of the NRC research program.

A Vision for the Future

Nuclear energy will remain an important and perhaps growing component in the mix of energy generating technologies used in this Country. There is the potential that many new reactors could be built in the next 15 to 20 years. It is unlikely that agency resources of either manpower or funds will experience a similar growth. Indeed, the experience level of the NRC staff is likely to decrease due to retirements just when the new plant licensing activities accelerate. A portion of the research program needs to be devoted to the development of a regulatory infrastructure for regulatory work in the next 20 years that supports a staff with less experience dealing with more tasks. Computerization will be undoubtedly an important element of such an infrastructure. The ACRS can foresee, for example, a time when regulatory staff have routine access to superior analysis tools for systems analysis, phenomenological analysis, and risk assessment. Development of such validated and verified tools for routine use by non-specialists will require a research program that is not tied exclusively to the near-term issues of the regulatory process. Appropriate attention will have to be paid to the agency's analytical tools, its access to facilities, and its ability to provide recently recruited staff with a sound understanding of past safety decisions. Availability of good infrastructure will enhance safety and allow for much more efficient and effective NRC review of new reactor designs and licensing applications based on realistic evaluations of safety.

Observations and Recommendations on Specific Research Activities

NRC research has made substantial progress since the last ACRS report, NUREG-1635, Vol. 6, on the research program. This progress has occurred despite the diversion of substantial research talent in the agency to address issues of reactor security that are not reviewed here. Notable accomplishments of the research program in recent years include:

- Multidisciplinary review of pressurized thermal shock criteria
- Performance of high-burnup fuel during reactivity transients
- Embrittlement of zirconium alloy cladding when taken to high burnup.

The ACRS applauds these high technical quality research accomplishments. The ACRS is, however, disappointed at the pace with which these important research results are being used to modify regulations.

Other major observations and recommendations concerning the NRC research activities are summarized below and also discussed in more detail in individual Chapters.

Advanced Reactor Research

Highest priority should be given to those research activities that support the ESBWR design certification process. The importance of tasks associated with the ACR-700 or a related design with higher power depends on whether the certification review for such a reactor is resumed.

Digital Instrumentation and Control Systems

Software-based digital electronic systems are inevitable for both current and more advanced design nuclear power plants. The staff has developed a research plan that addresses the challenges associated with the use of digital technology that will face the agency in the next five years.

The ACRS has recently reviewed and reported favorably on the research plan for digital systems. The ACRS was impressed by the technical quality in the development of the research plan, the scope and content of the plan, and the prioritization of activities in the plan. The ACRS recommends a number of improvements to an already quality research plan, including addition of an explicit element to the plan to study the acceptability of international standards in comparison to Institute of Electrical and Electronics Engineers (IEEE) standards for meeting regulatory requirements concerning digital instrumentation and control systems. This study will be an important element of efforts to develop a multi-national design approval process.

Fire Safety Research

There have been a number of important accomplishments by NRC research in the area of fire protection since the last ACRS report on NRC safety research program in 2004. Fire safety research continues to merit emphasis in the NRC research program.

RES, in cooperation with EPRI has taken some important steps to consolidate the fire PRA research and development activities, conducted over the past few years, into a single state-of-the-art methodology for fire risk assessment.

There are a variety of methods that can be used to model the progression of fires. Some of these have been used in fire protection programs for non-nuclear facilities for many years. The ranges of applicability of these methods have not been well studied or documented. In cooperation with EPRI, a program is in progress to verify and validate a set of fire progression modeling tools. The accuracies of these tools are being examined for different fire conditions and applications by comparison with benchmark tests.

RES has worked closely with the U.S. industry in undertaking generic fire risk research activities. Fire risk is, however, an issue of world-wide concern. RES has not aggressively sought collaborations with the international community to advance NRC capabilities for fire risk assessment. Collaborations with other countries especially in experimental studies may be essential to leverage resources of all partners sufficiently to achieve fire risk assessment capabilities commensurate with what can now be done for risk from normal plant operations.

Reactor Fuel Research

The NRC research on reactor fuel has been concentrated in recent years on the confirmation of regulatory decisions that allow licensees to take light water reactor fuels to burnups of nominally 62 GWd/t. The research on high-burnup fuel is reaching a substantial level of maturity. Some major confirmatory experiments remain to be done - notably experiments on reactivity insertion to be done in a water loop at the CABRI reactor. Since the last ACRS report on NRC safety research program, plans for these experiments have been revised so the experiments which are part of an international collaborative effort now better meet the agency needs. It is important that this work that is so well coordinated both with agency needs and with international partners be taken to completion. Still major findings of the research effort can be reduced to regulatory practice now. This reduction to regulatory practice needs to be initiated and pursued aggressively.

It is evident that high-burnup fuel research will soon achieve results that are adequate for agency needs. The NRC has made clear that it will expect the nuclear industry to provide necessary safety analyses and experimental data should the industry want to take fuel to burnups that exceed the current regulatory maximum. NRC needs to make these expectations more explicit, particularly its expectations for the experimental data needed to support the analyses of high-burnup fuel behavior under accident conditions.

Neutronics and Criticality Safety

The neutronics and criticality safety research program is small but appears adequate to ensure that the NRC has capabilities to meet immediately foreseen regulatory needs.

In the future, more innovative core designs for advanced reactors may be submitted to the NRC. Confirmatory analyses of reactor core physics will be an essential part of the regulatory process for these advanced reactors. The capabilities now available to the NRC in the area of core physics may well be stretched. It will be useful to the agency to understand these future needs. If long-term development activities are identified, such as those that might be needed for analysis of the PBMR, additional research may be needed in this area.

Human Factors and Human Reliability Research

As new reactor designs, likely dependent on a higher degree of automation than the current fleet, are introduced, the need for revised guidance and tools for the NRC staff in human factors and human reliability analysis will increase. RES has initiated a project to develop regulatory guidance and analytical techniques to review human factors for advanced nuclear power plants. The ACRS views this five-year project essential for preparing the staff in reviewing advanced reactor designs.

The quantification of human reliability continues to be a challenge in risk assessments. Human reliability modeling introduces large uncertainties in probabilistic risk assessments. The NRC staff needs guidance in its review of the human reliability models used by the industry in risk-informed licensing applications. Progress has been made with the publication of NUREG-1792, “Good Practices for Implementing Human Reliability Analysis (HRA).” Still, further guidance is needed for reviewers of licensing applications.

Materials and Metallurgy

The NRC is investing heavily in the better understanding of materials degradation issues in the currently operating fleet of nuclear power plants. Such investment is justified in view of significant agency regulatory activities that aging degradation research supports.

The current program is well focused on improving the agency’s ability to independently evaluate licensees’ efforts to prevent, detect, and mitigate environmentally assisted stress corrosion cracking.

The nuclear industry and the NRC have often been surprised by unexpected material degradation problems. As a result, they have responded to such problems in a reactive mode which has proven to be inefficient. The Proactive Materials Degradation Assessment project is an effort to identify potential material degradation problems before they manifest in operating nuclear power plants. The ACRS admires the vision of this undertaking and supports its continuation. The ACRS looks forward to reviewing the initial results of this ongoing effort soon and learning whether the admirable goal of this project is, in fact, feasible.

RES needs to reevaluate the need for continued research into heavy section steel components. This research may be justified if there is a clear need for NRC to develop its capabilities in the area of probabilistic fracture mechanics (PFM) so that it can evaluate licensees' applications. If this is the case, the research needs to be clearly focused on this objective and not the research that the industry should perform to meet its responsibilities to ensure reactor pressure vessel integrity. It appears now, however, that it is NRC that is advancing the state-of-the-art and making available information that allows licensees to reduce conservatism in their analyses.

Operational Experience

The ACRS is supportive of the research activities in the area of operational experience and recommends that these activities be continued. In light of the limited resources allocated to these tasks, RES has done a commendable job in producing outputs in well-documented and thorough fashion. Tasks that are currently in the 2005 research plan related to operational experience should remain funded and should be continued for the foreseeable future.

Probabilistic Risk Assessment

Altogether the scope and the number of activities in the NRC's PRA research program is quite impressive. The ACRS cautions, however, that NRC should not allow its work in such a crucial technology as risk assessment to become totally devoted to the support of line activities. Methods development is still important. As an example, the ACRS notes that considerable research is being reported in the literature regarding Binary Decision Diagrams as tools for solving large fault tress without resort to cutoff frequencies as is now done. The staff needs to review the literature concerning Binary Decision Diagrams and evaluate the need to adopt this technology. The growing importance of the SAPHIRE code and the SPAR models in the regulatory process warrants such an investigation.

Seismic Research

Seismic hazard analysis and structural response are not areas where NRC must maintain state-of-the-art expertise. Such expertise is available to the NRC on a contractual basis. As ACRS noted in its previous report on NRC safety research program, research activities at the agency can be confined to support needed updates to regulatory guides and collaborative work with the international community to stay abreast of developments in other countries. The current research program is, indeed, largely focused on needs of the regulatory process and a few important international collaborations.

Severe Accident Research

The ACRS is very supportive of the strategy NRC has developed to maintain and update its capabilities for severe accident analyses. The leveraging of resources through international collaborative experimental research is especially important. The planned extensions and continuations of current collaborations are well worth the investment.

Thermal-Hydraulics Research

Highest priority should be given to the integration of TRACE code into the regulatory process. As this integration progresses, the research staff can continue its efforts to improve and further develop TRACE on a "time available" basis. The ACRS is concerned now that efforts to improve TRACE lack prioritization and defensible organization. Prioritization of technical improvements might be aided substantially by commissioning a detailed peer review of TRACE. To do this, the staff will have to have available code documentation of outstanding scope and quality. Such high quality code documentation will also be needed if the code is to become part of the regulatory process. Code documentation, then, is a task that ought to take precedence in the thermal- hydraulic research effort.

3 ADVANCED REACTOR RESEARCH

The agency is already engaged in various activities related to a number of new plant designs, including ESBWR, PBMR, IRIS, and ACR-700. The staff has begun its review of ESBWR design certification application. It is anticipated that requests for design certification reviews will be received for EPR, and PBMR. Of these, the ESBWR, ACR-700, IRIS, and EPR can be certified in all likelihood under the current requirements in 10 CFR Part 52 using the design basis accidents as they are now defined. Nevertheless, there will be the need for NRC to verify the thermal-hydraulic assessments made by the applicants for the various designs. This will require review and approval of the computer codes that were used by the applicants for assessing the design basis accidents. Confirmatory analyses will require that design-specific versions of the computer codes TRACE and CONTAIN be available to the staff for audit calculations and independent assessment of separate effects and integral system experiments. Highest priority should be given to those research activities that make such tools available for the ESBWR design certification review. This includes tasks Y6857, Y6898, N6018, and Y6804. The importance of tasks associated with the ACR-700 or a related design with higher power, Y6831, Y6812, Y6899, Y6489, Y6748 and Y6933, depends on whether the certification process for such a reactor is resumed.

Certification reviews for designs such as the PBMR and the 4S that do not use water reactor technology will be more challenging. Although significant efforts were undertaken in the past to license such non-LWR designs under the current regulatory system designed for light water reactors, it would be far more appropriate, effective, and efficient to have the "technology-neutral-framework" for certification of such designs. For timely application to these reactor types (and possibly even more unusual designs in later years), the development of the technology-neutral framework needs to be given high priority and provided sufficient resources to complete the job in 2006 and to allow two years for rulemaking. High priority, then, should be given to the tasks N6205 and Y6487 that will develop a technology-neutral framework for the regulation of advanced nuclear power plants.

The Commission has expressed a desire for "enhanced safety" for new reactor designs. To ensure that new designs have reached enhanced levels of safety, the NRC will require each of the applicants for design certification to submit a full-scope PRA with consideration of uncertainties. The staff must be prepared to review these PRAs, to validate the results and to compare the results with acceptance criteria for "enhanced safety." This evaluation will include undoubtably a complete Level-2 evaluation of accident source terms since LERF (large early release frequency) will no longer be an appropriate safety metric. To review and independently assess the Level-2 analyses of source terms, the regulatory organizations will need design-specific versions of the MELCOR computer code. There is, then, the potential need to develop MELCOR versions specific for the PBMR and 4S designs. Development of such code versions will take time. Second priority should be given then to tasks K6703, Y6801, and Y6619. Again, the importance of developing an accident progression model for ACR-700 depends on resumption of its certification process.

Table 1. Advanced Reactor Research Activities

Job Code	Title	Comment
Y6857	***ESBWR Input Deck Development***	Analysis of DBAs in ESBWR using the TRACE code; This project should have high priority.
Y6898	***ESBWR Design Certification Report***	Support for review of PRA for ESBWR; This project should have high priority.
N6018	***Separate Effects Experiments***	Separate effects tests in support of TRACE model development for ESBWR; This project should have high priority
Y6804	***ESBWR Containment Support***	Analysis of experiments with CONTAIN and MELCOR. This is a high priority task for ESBWR design certification review.
Y6489	***PRA for ACR-700***	Support for review of ACR-700 PRA. This project can be deferred until certification application becomes active again.
Y6899	***ACR-700 Design Certification Support***	Support for review of PRA for ACR-700. This project can be deferred until the certification application becomes active again.
Y6748	***Review ACR-700 Support***	Support for thermal hydraulics review of ACR-700. This project can be deferred until the certification application becomes active again.
Y6831	***Methods Development for ACR-700***	TRAC code upgrades needed for ACR-700 certification calculations. This project can be deferred until the certification application becomes active again.
Y6812	***ACR-700 Input Model Development***	Develop RELAP5 and TRAC-M input models for ACR-700. This project can be deferred until the certification application becomes active again.
Y6933	***Evaluate Severe Accident Phenomena in ACR-700***	Analysis of risk dominant sequences for ACR-700. This project can be deferred until the certification application becomes active again.
K6703	***Coop. Agreement with Center for Advanced Nuclear Energy Systems***	Improve NRC's knowledge and information on advanced reactors. This project is useful but can have a second level priority.
Y6619	***Advanced Reactor PRA Development***	Develop knowledge needed to review advanced reactor PRAs. Second priority work for non-LWR design certifications.

Table 1. Advanced Reactor Research Activities (Continued)

Job Code	Title	Comment
Y6801	***Advanced Reactor/Severe Accident Code Development***	Develop a version of MELCOR code for advanced reactors. This project can have a second level priority.
Y6755	***Materials Evaluations for Advanced LWR Reactors***	Research materials engineering issues for advanced LWRs especially effect of coolant environment on fatigue and in-service inspection and monitoring. This project can have a second level priority.
N6205	***Assistance for Development of a Regulatory Structure for New Plant Licensing***	Development of a technology-neutral regulatory framework. This project should have high priority.
Y6487	***Advanced Reactor Regulatory Framework Development***	Development of a regulatory framework for advanced reactors. This project should have high priority.
Y6741	***Environmental Effect on Containment***	Develop understanding of the properties of concrete in high temperature gas cooled reactors. This project can have a low priority.

4 DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

Software-based digital electronic systems are inevitable for both current and advanced design nuclear power plants. Already such software-based digital electronics appear ever more frequently in systems for plant control. Eventually, they will appear in safety systems. The reliability of digital systems especially when using commercial, "off the shelf" hardware and software has become an issue because they cannot be comprehensively tested. The quality of the requirements for the software cannot be assessed fully through testing. Quality in the software-based systems is achieved through the control of the process of software development. Particular attention has to be given to the requirements for the system software. Failure to specify adequate requirements has often been found to be the root cause of digital system failures. Review and approval of licensee applications to incorporate software-based digital systems in its facility is, then, time-consuming for both the regulator and the licensee. New failure modes that arise in digital systems need to be recognized. Such failures can depend on the operational state of the system at the time of failure. Indeed, testing and maintenance as well as normal operations of digital systems can create the opportunities for their own unique kinds of failures.

Security of digital systems has become a major concern and there needs to be regulatory guidance and acceptance criteria for the security aspects of digital systems. Codes, Standards, and regulations must prompt the designer of digital safety systems to avoid system communications outside of the controlled areas of the plant and the use of wireless technology must be carefully evaluated to prevent interception, interdiction, or interference in communications to digital systems.

Current licensing guidelines provide information on what to review in digital systems. They do not necessarily provide sufficient guidance on how to review submittals or the acceptance criteria to apply. The NRC staff needs a firm technical basis for deciding when review of submittals is adequate and when confirmatory analyses are necessary. The situation will get worse with time. Digital systems in nuclear power plants are expected to become more numerous. The complexity of these systems will increase. There is the potential for the consolidation of what are now discrete analog safety systems into a single digital system. At the same time, there is interest both within the agency and on the part of licensees to adopt risk-informed techniques for the review of digital software systems. NRC lacks the technical basis to support risk-informed reviews of digital systems. Currently, the ability to model the reliability of software-based digital systems in PRAs is very limited. Without quantitative risk information, a much less defensible, qualitative, "graded approach" to the review of digital systems is likely to emerge.

If the use of digital protection systems and control systems becomes as widespread as now predicted, review of digital systems as part of ITAAC (Inspections, Tests, Analyses, and Acceptance Criteria) may eventually become a burdensome, time-consuming aspect of the licensing process. Methods and tools to facilitate confirmation that "as built" systems conform to accepted designs are going to be needed. As use of digital systems becomes more extensive in nuclear facilities, NRC may find it necessary to reconsider its current positions on defense-in-depth and diversity in instrumentation and control systems.

The nuclear industry is not a major user of digital technology relative to many other industries. Yet, the consequences of failure of digital systems in nuclear power plants are likely to be less acceptable to the public than are failures of such systems in other industries even when consequences are significant. Greater rigor in the review of digital systems is necessary for nuclear applications of these systems. It is expected then that NRC will have to “blaze new paths” in this area through research. In particular, the usual industrial practice of separately considering hardware and software reliabilities may not be adequate for nuclear systems and a more integrated or systems approach may be needed.

The staff has developed a research program plan that addresses these challenges that will face the agency in the next five years. Critical reviews of the state of the art in several areas were completed, documented and presented before audiences in professional societies. Recommendations made to the NRC by independent bodies, including the National Academy of Sciences were considered in the development of the plan. Inputs from the program offices at NRC (NRR, NSIR, and NMSS) were also obtained. The research plan is well directed toward meeting the agency needs and is intended to provide:

- Improved technical guidance for review of digital systems
- Technical support for developing improved acceptance criteria for assessing the safety and security of the systems
- Tools and methodologies for improved review of digital systems
- Technical bases for including models of digital systems in PRAs

The research plan has six major elements:

- Systems aspects of digital technology
- Risk assessment of digital systems
- Emerging digital technology with application to nuclear facilities
- Software quality assurance
- Security aspects of digital systems
- Advanced nuclear power plant digital systems

Within each of these major elements of the plan, there are a number of subelement. The staff has prioritized work on the subelement basis. Now, the major focus of the work is on collection of data on the failure modes of digital systems, including international experience with digital system failures, software quality assurance, environmental stressors on digital systems, modeling digital systems in PRAs and cyber security of digital systems. Within the general element of emerging digital technologies applicable to nuclear facilities, attentions are on system diagnosis, prognosis and on-line monitoring as well as wireless technology. Research on digital systems for advanced nuclear power plants was given a low priority. Perhaps, future new orders for advanced plants (AP1000, ESBWR, etc.) may create new regulatory demands and cause this priority to be re-evaluated.

The ACRS has recently reviewed and reported favorably on the research plan for digital systems. The ACRS was impressed by the technical quality in the development of the research plan, the scope and content of the plan, and the prioritization of activities in the plan. Indeed, it would help better understanding of other research programs if they were also based on such thorough planning efforts. The ACRS recommends the following to further improve an already quality research plan:

- The plan is currently focused very much on the software aspects of digital systems. Eventually, the research will have to be expanded to recognize the entire system of interest. Though the focus on software is appropriate now, the plan should reflect the need for expansion in scope in the longer term.

- There should be an explicit element of the plan to study the acceptability of international standards in comparison to IEEE standards (such as IEC 60780 in comparison to IEEE 323) for meeting regulatory requirements concerning digital instrumentation and control systems. This study will be an important element of efforts to develop a multi-national design approval process.

- As data on digital system failures are collected and analyzed, the research staff should prepare episodic papers or presentations to professional societies of their interpretations and “lessons learned” for peer review by the larger digital system reliability community.

Table 2. Research Activities in Digital Instrumentation and Control Systems

Job Code	Title	Comment
N6116	***Secure Network Design Techniques***	Develop technical guidance for mitigating cyber vulnerabilities in secure networks
N6095	***Assignment Robert Edwards***	Support analysis of digital systems failures and consequences
Y6962	***Emerging Technologies***	Conduct periodic surveys of the state of the art for a wide range of technology issues in the I&C field
Y6873	***International Cooperative Research Program on Digital I&C***	Search for opportunities to collaborate in the safety assessment of digital systems
N6010	***COMPSYS***	OECD/NEA international program to develop data-base on digital systems failures
K6472	***Risk Importance of Digital Systems***	Develop methods to include digital systems in PRAs
Y6332	***Digital Systems Risk***	Develop a PRA method for modeling failures of digital I&C systems.
Y6591	***Software Reliability Code Measurements***	Large-scale validation of NRC methodology for predicting software reliability in digital systems
N6080	***Interactions with Industry on Standards***	Development of standards on EMI/RFI
Y6475	***Wireless***	Confirmatory research on effects of wireless communications
N6113	***Security of Digital Platforms***	Study in laboratory digital systems generically qualified for nuclear safety applications
N6114	***Site-specific Protocol Analysis***	Study power plant implementation of digital systems generically qualified for nuclear safety applications
N6124	***Digital System Dependability Performance***	Qualify safety of a digital system using a process developed in NRC research
W6851	***Review Guidance for Lightning***	Support for response to public comments on draft regulatory guide; Program completed.
Y6924	***SPACE Engineering Workstation for Review of TXC Applications***	Evaluate the use of the RETRAN tool for review of TELEPERM-based digital instrumentation and control upgrades
Y6349	***Halden Environmentally Assisted Cracking*** (The title of this program is amazingly misleading!)	Despite the name this is research on COS operating experience, ranking software engineering practices and testing digital reliability assessment methods

5 FIRE SAFETY RESEARCH

The fire safety research program can be divided into three technical areas:

- Fire Risk Assessment
- Fire Modeling
- Fire Testing

Each of these areas is discussed below.

Fire Risk Assessment: The nuclear industry has made substantial progress over the past thirty years in the development and standardization of internal events risk assessment. Progress in the development of the methods of fire risk assessment has been much slower. Only a few nuclear power plants currently have full-scope fire risk assessments. The requirements placed by the NRC on the industry for performing Individual Plant Examinations of External Events (IPEEE) permitted the use of simplified and qualitative techniques. Most analyses of fire risk at nuclear power plants were performed with these less quantitative techniques..

As the NRC moves from deterministic regulations to risk-informed and performance-based regulations, the need for quality risk information increases greatly. It is expected that many nuclear power plants will transition from their current fire protection programs to the risk-informed, performance-based fire protection programs that meet the requirements of 10 CFR 50.48(c) and the referenced 2001 Edition of National Fire Protection Association (NFPA) standard, NFPA 805, “Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Stations.” This is only possible if a full-scope fire risk assessment is performed for each transitioning nuclear power plant. NRC will need appropriate standard to assess the quality of such fire risk assessments and inspectors will need tools

203-912-All-Post

213-950-A-PstSpry

115-837-E-PstSpry

Confirmatory Testing of Hemyc/MT Fire Barriers

The Hemyc and MT electrical raceway fire barrier systems are used in a number of plants to provide a fire barrier between two trains of safe shutdown equipment within a fire area. In the performance of fire protection inspections at nuclear plants, questions raised regarding the fire resistance capability of these systems. NRC conducted confirmatory testing of Hemyc and MT fire barriers at the Omega Point Test Facility in 2005. All of the configuration tested failed to meet acceptance criteria. A Generic Letter was issued requiring licensees to identify where Hemyc and MT fire barriers are used in their plants and to provide a plan and schedule for corrective actions.

and the knowledge to assess the validity of changes to the licensing basis made at the plants.

RES in cooperation with EPRI has taken some important steps to consolidate the fire PRA research and development activities, conducted over the past few years, into a single state-of-the-art methodology for fire risk assessment. In 2005, the final NUREG/CR-6850, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities,” was issued. This document provides a structured framework for the overall fire risk assessment as well as specific recommended practices to address key aspects of the analysis. While the primary objective of the project was to consolidate state-of-the-art methods, in many areas the newly documented methods represent a significant advancement over those previously documented. Although some utilities have used parts of the improved approach, no utility has completed a fire risk assessment using the methodology and submitted the assessment for critical peer review.

Areas of fire risk analysis where further development in methodology is needed have been recognized by RES. These include spurious equipment actuations, post-fire human reliability analysis, aging effects, and low power and shutdown fire risk.

Fire Modeling: Deterministic criteria for fire protection are typically very conservative in their treatment of fire progression. Fire risk assessment, on the other hand, requires a realistic assessment of fire progression. There are a variety of methods that can be used to model the progression of fires. Some of these have been used in fire protection programs for non-nuclear facilities for many years. The ranges of applicability of these methods have not been well studied or documented. In cooperation with EPRI, a Project (Y6688) is in progress to verify and validate a set of fire progression modeling tools. The accuracies of these tools are being examined for different fire conditions and applications by comparison with benchmark tests performed by National Institute of Standards and Technology (NIST) The phenomena identification and ranking table (PIRT) process is being used by RES to identify potential limitations of the fire progression modeling tools. Preliminary draft of multi volumes NUREG-1826, “Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications,” was issued for Public Comment in January 2006.

Fire Testing: Confirmatory testing is another critical element of the fire safety program. During the past year, tests were performed at the Omega Point Test Facility on the Hemyc and MT electrical raceway fire barrier systems (see side column). The test results indicated that these fire barrier systems are not capable of satisfying regulatory requirements. It is somewhat distressing that confirmatory testing of these fire barriers did not occur until sixteen years after problems were identified with a similar fire barrier material, Thermo-lag, and five years after inspection teams raised specific concerns about the Hemyc and MT fire barriers. The results of these tests provide further evidence of the continuing value of NRC’s confirmatory testing program.

There have been a number of important accomplishment by NRC research in the area of fire protection since the last ACRS report on NRC safety research program in 2004. Fire safety research continues to merit emphasis in the NRC research program. Approximate, and often bounding risk analysis, performed for individual plants indicate that the risk of core damage from fire- initiated events is comparable to or greater than the risk from other accidents initiated during normal operations. It is important to know whether the same conclusion would be drawn if fire risk assessments were performed using tools of comparable sophistication as those used for assessing risk of accidents initiated by internal events. Conclusions based on more realistic fire risk assessments could have ramifications on both regulatory attention and licensee attention to safety. In the interim, risk-informed regulatory decisions are being made with an incomplete understanding of the impact of fire on risk.

RES has worked closely with the U.S. industry in undertaking generic fire risk research activities. Fire risk is, however, an issue of world-wide concern. France, for example, has recently initiated a fire research program in a multi-volume test facility. RES has not aggressively sought collaborations with the international community to advance NRC capabilities for fire risk assessment. Collaborations with other countries especially in experimental studies may be essential to leverage resources of all partners sufficiently to achieve fire risk assessment capabilities commensurate with what can now be done for risk from normal plant operations.

Table 3. Fire Safety Research Activities

Job Code	Title	Comment
N6107	***10 CFR 50.48C - related Technical Activities***	Develop fire PRA methods, tools, and data. Perform demonstration studies. This is a collaborative effort between NRC and EPRI.
N6108	***Fire Risk Assessment and Risk Applications***	Improve fire PRA approaches. Develop test plan to address spurious equipment actuation issues.
N6134	***LPSD Level 1 & Fire Risk Standard***	Supports NRC staff in the development of industry standards.
Y6651	***Effects of Switchgear Aging on Energetic Faults***	Assess the aging of medium voltage switch gear as it affects the potential for energetic electrical faults. Such faults are thought to contribute significantly to fire initiation. The work addresses how aging affects fire risk.
Y6688	***Fire Model Benchmarking and Validation***	Benchmark fire model computer codes against fire experiments performed by NIST. Such validation is necessary to ensure that appropriate tools are used for regulatory applications.
Y6817	***Fire Protective Wrap Performance Testing***	Test Hemyc and MT fire wrap materials. These important tests conducted in 2005 showed there to be significant issues associated with these fire barrier materials.
Y6955	***Fire Incident Records Exchange***	Collect and analyze international fire events data. This is a long-term collaborative effort with OECD.

6 REACTOR FUEL RESEARCH

Reactor fuel is an important element of safety technology. NRC must maintain expertise in the area of reactor fuel because of both the importance to safety and because of the limited availability of expertise outside the agency that is independent of licensees. Research is an important vehicle for maintaining expertise in reactor fuel. NRC research on reactor fuel during normal operations and design basis accidents has been concentrated in recent years on the confirmation of regulatory decisions that allow licensees to take light water reactor fuels to burnups of nominally 62 GWd/t. This research has largely resolved the issue of the vulnerability of high-burnup fuel and cladding to reactivity transients though some confirmatory tests need to be completed. Research results will allow regulatory changes to better reflect the degraded capacity of high-burnup fuel to sustain reactivity insertion events.

The reactor fuel research has remained quite productive as examinations of high-burnup fuel behavior under loss-of-coolant accidents have been initiated. An important discovery has been the synergistic effect on clad ductility of hydrogen absorption during normal operation and steam oxidation of the cladding during an accident. Based on the research, revised embrittlement criteria have been developed that could be incorporated into 10 CFR 50.46.

The research on high-burnup fuel is reaching a substantial level of maturity. Some major confirmatory experiments remain to be done - notably experiments on reactivity insertion to be done in a water loop at the CABRI reactor. Plans for these experiments have been revised since our last report on reactor fuels research so the experiments which are part of an international collaborative effort now better meet agency needs. It is important that this work that is so well coordinated both with agency needs and with international partners be taken to completion. Still, major findings of the research effort can be reduced to regulatory practice now. This reduction to regulatory practice needs to be initiated and pursued aggressively.

It is evident that high-burnup fuel research will soon achieve results that are adequate for agency needs. The NRC has made clear that it will expect the nuclear industry to provide necessary safety analyses and experimental data should the industry want to take fuel to burnups that exceed the current regulatory maximum. NRC needs to make these expectations more explicit, particularly its expectations for the experimental data needed to support the analyses of high-burnup fuel behavior under accident conditions.

Completion of NRC's research on high-burnup fuel raises the question of how NRC will maintain expertise in fuel. Continued evolution in fuel cladding alloys can be anticipated. Interest is developing within the industry in fuels with enrichments exceeding 5% ^{235}U. These higher enrichment fuels may necessitate NRC research. If use of MOX fuel becomes more widespread than the planned disposal of excess weapons-grade plutonium, additional research on MOX fuel with reactor grade plutonium may be needed. Research on both higher enrichment fuel and MOX fuel can be done with substantial collaboration with international partners. Such collaboration will further the ideal of international safety evaluations of nuclear power plants.

Table 4. Reactor Fuel Research Activities

Job Code	Title	Comment
Y6586	***Fuel Code Assessment for MOX***	Improve FRAPCON and FRAPTRAN for calculating the behavior of MOX fuel rods; An important activity for licensing core loads for excess weapons-grade plutonium disposal.
Y6580	***Fuel Code Applications for High Burnup Fuel***	Improve FRAPCON and FRAPTRAN for calculating the behavior of high burnup fuel rods; an important activity as licensees press limits on allowable fuel burnup.
Y6788	***Halden Fuel Experiments Under Transient Conditions***	Data on fuel behavior under operational transient conditions for code development.
N6074	***STUDSVIK Cladding Integrity Project***	Stress corrosion cracking, hydride embrittlement and delayed hydride cracking study of ZIRLO clad. Defueled clad segments provided for NRC research.
Y6849	***ZIRLO Cladding Performance***	Adequacy of criteria for ZIRLO cladding performance in a LOCA; an important study of cladding used for high burnup fuel and the synergism between hydriding and oxidation on clad ductility.
Y6850	***M5 Cladding Performance***	Adequacy of criteria for M5 cladding performance in a LOCA; an important study of cladding used for high burnup fuel and the synergism between hydriding and oxidation on clad ductility.
G6923	***Failure of Hydrided Zircaloy under Severe Loading Conditions***	Develop theoretical model of mechanical failure of hydrided Zircaloy cladding.
W6832	***CABRI Water Loop***	NRC support for the CABRI water loop for RIA testing of high burnup fuel; confirmatory testing of high burnup clad and fuel vulnerability to reactivity transient events.
Y6367	***High Burnup Cladding Performance***	LOCA testing of high burnup cladding behavior; important study of cladding used for high burnup fuel and the synergism between hydriding and oxidation on clad ductility.
Y6723	***International Agreement on Fuel Behavior and Materials Science Research***	Data report on BIGR pulse reactor tests.
Y6847	***Clad Performance in ATWS***	Determine the adequacy of criteria and analysis of clad performance in BWR power oscillations; NRC needs to see if this problem can be solved by analysis with minimal experimental confirmation.
Y6195	***Dry Cask Storage License for High Burnup Fuel***	Develop criteria for dry-cask storage and transportation of spent high burnup fuel.

7 NEUTRONICS AND CRITICALITY SAFETY

Neutronics and criticality safety are areas in which NRC must maintain exceptional capabilities through its research program. The neutronics and criticality safety research program is small but appears adequate to ensure that the NRC has capabilities to meet immediately foreseen regulatory needs. The current NRC research activities in neutronics analysis, core physics, and criticality safety are listed in Table 5. Maintenance of the SCALE suite of codes is essential for the analysis of reactor core physics. These codes are complemented by the PARCS code which is part of the TRACE code and is discussed in more programmatic detail in the Chapter 14 of this report dealing with Thermal Hydraulics Research. The availability of the NEWT lattice code is important to licensees since it will be essential for the use of more advanced computer models in future regulatory processes. Currently, this lattice code is being used for the analysis of reactor cores fueled in part with MOX fuel for the disposition of excess weapons-grade plutonium. Several other activities are under way to support the licensing of MOX fuel core at the Catawba reactor for this plutonium disposition activity. These are appropriate programs at the current time. It is noted that NRC is taking advantage, to the extent feasible, of the considerable European experience with MOX fuel made with reactor-grade plutonium.

In the future, more innovative core designs for advanced reactors may be submitted to the NRC. Confirmatory analyses of reactor core physics will be an essential part of the regulatory process for these advanced reactors. The capabilities now available to the NRC in the area of core physics may well be stretched. It will be useful to the agency to understand these future needs. If long-term development activities are identified, such as those that might be needed for analysis of the PBMR, additional research may be needed in this area.

Table 5. Research Activities in Neutronics Analysis, Core Physics, and Criticality Safety

Job Code	Title	Comment
Y6846	***SCALE Code Development for Reactor Physics***	Essential code for neutronics analysis to audit licensee submittals and other regulatory needs.
Y6320	***NEWT Lattice Code***	Generate lattice cross-sections for safety analysis of MOX cores to support licensing of cores for Pu disposal.
N6162	***MOX Benchmark***	Confirmation of uncertainties in PARCS code predictions of MOX core neutronics; Also supports the licensing of Pu disposal activities.
Y6403	***Reactor Core Analysis***	Analysis to predict details of reactivity transient in MOX core. Again, this research supports regulatory activities associated with the DOE program to dispose of excess weapons-grade plutonium.
Y6685	***Experimental Data for High Burnup Spent Fuel Validation***	This project provides NRC with foreign and domestic data on high burnup fuel and MOX fuel for assessment of analytical tools used to predict fuel inventories, decay heating, and radiation shielding.

8. HUMAN FACTORS AND HUMAN RELIABILITY RESEARCH

Human performance plays a critical role in the safe operation of nuclear plower plants. Human performance issues have been main contributors to accidents and unsafe conditions experienced by the current fleet of operating reactors. They can be expected to continue to have a major impact on nuclear power plant safety. As licensees increasingly rely on risk-informed licensing applications that require the quantification of human reliability under accident conditions, the staff needs to be able to evaluate the treatment of operator actions in such applications. As new reactor designs, likely dependent on a higher degree of automation than the current fleet, are introduced, the need for revised guidance and tools for the NRC staff in human factors and human reliability analysis will increase. Therefore, it is very important that the NRC maintain research programs in these areas.

The current NRC research activities in the areas of human factors and human reliability analysis are:

- Human Factors
  (B7488, N6207, Y6843, N6137, Y6529)

- Human Reliability Analysis
  (Y6497, Y6496, N6248)

Current research in the human factors area includes a continuing international collaborative research program at the Halden project (B7488). The ACRS is supportive of this collaborative program and recommends continued NRC participation.

The project "Development of a Regulatory Guide and Analytical technique for Assessing NPP Staffing" (N6207) supports the development of guidance for staffing exemption requests to 10 CFR 50.54(m). This project is almost complete. Guidance is now provided in the recently issued NUREG-1791, "Guidance for Assessing Exemption Requests from the NPP Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)." Publication of this guidance is a significant accomplishment that provides a more flexible approach to staffing of current and future reactors.

Human performance issues, including organizational issues are of great importance to nuclear reactor safety. Inspectors at nuclear power plants currently have limited guidance or means with the Reactor Oversight Process (ROP) to characterize problems associated with human performance. This issue has been highlighted in a recent report from the Inspector General. In response to a Commission request, the project Y6843, "Develop Human Performance Indicators,"has been initiated to study the feasibility of establishing the technical bases for indicators of human performance that would be used to supplement indicators currently used in the ROP. This research is appropriate and very important. It may lead to significant improvements in the NRC inspection program and the ROP.

There is evidence of degrading performance of operations personnel in the nuclear and other industries due to operator overload. The research project N6137, "Impact of Operator Workload on Human Performance," is a five-year effort to assess the impact of operator overload on performance. The plan is to develop licensing requirements as well as inspection guidance and techniques for reviewing the impact of workload on operator performance and plant safety. This is an important new project that deserves support both for the current fleet of operating reactors and for advanced reactor designs.

Advanced reactor designs are likely to introduce much greater automation than exists in current reactors. Certainly, advanced digital control and instrumentation methods as well as new human-system interfaces can be anticipated. These new features of plants are likely to have some effects on human performance. The NRC staff needs to prepare itself to review new concepts and designs proposed by licensees. The project "Human Factors of Advanced Reactors" (Y6529) has been initiated to address this issue and to develop regulatory guidance and analytical techniques to review human factors for advanced nuclear power plants. The ACRS views this five-year project essential for preparing the staff in reviewing advanced reactor designs.

The quantification of human reliability continues to be a challenge in risk assessments. Many approaches to the quantification of human reliability have been proposed. However, the benchmark exercise conducted by the Ispra Laboratory of the European Union demonstrated that the choice of model has a significant impact on the results obtained. Not much progress to improve this situation has been made since that exercise was performed. The NRC staff has recently completed an assessment of the strengths and weaknesses of the various methodologies now available for assessing human reliability. The ACRS has been quite impressed with this assessment and hope the work leads to the identification of best methods for the quantification of human reliability in PRA.

Human reliability modeling introduces large uncertainties in PRAs. The NRC staff needs guidance in its review of the human reliability models used by the industry in licensing applications. The project Y6497, "HRA Application and ATHEANA Maintenance," is intended to improve NRC's ability to independently model human reliability and to provide guidance concerning risk-informed regulatory applications. Progress has been made with the publication of NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)." Still, further guidance is needed for reviewers of licensing applications. The NRC has applied ATHEANA model to the human performance issues associated with its recent pressurized thermal shock study. The NRC is also planning to apply the ATHEANA model to a number of ongoing risk assessments, including those for fire and steam generator tube rupture to develop lessons learned on human reliability analysis and to develop guidance for the staff. If needed, modifications to the Standard Review Plan for licensee's applications will be devised. The ACRS believes that this effort is needed. ATHEANA is a state-of-the-art model of human performance and is complicated to use. Application of the tool will show whether benefits derived from the analyses are commensurate with the enhanced complexity. Application may also show how the complexity of ATHEANA can be reduced. Application of ATHEANA is, however, very much behind schedule. Resources and management attention are needed to either accelerate the efforts or to revise the scope of the application efforts.

Both ATHEANA and SPAR-H (the HRA model used in SPAR) quantify the probability that a human unsafe act will be committed. This probability depends on a number of performance shaping factors (PSFs) that determine the context within the crew operates. The available time for action is one of the PSFs estimated from thermal-hydraulic considerations. The evaluated failure probability ia understood to be the probability that the required action will not be completed within the available time.

An alternative approach to HRA is to recognize the importance of time taken by the crew to complete a task and to develop a probability distribution for this time. The failure probability, then, is calculated from this distribution as the probability that this time will exceed the available time.

Recent experiments performed at Halden, Norway, have shown that there may be significant variability in the time that crews take to perform a given task. Such evidence is very difficult to account for in ATHEANA and SPAR-H. The alternative approach could accommodate such evidence. In addition, the staff is currently supporting research at Idaho National Laboratory (INL) that develops “time lines” for past accidents. This evidence can also be accommodated in the alternative approach.

The staffshould evaluate the merits of an HRA model that focuses on the time required for action.

The project Y6496 is a continuing effort to develop an event database called Human Event Repository and Analyses. This database and analysis capability should significantly improve the treatment of human reliability in nuclear reactors and provide a realistic, performance-based database to assess licensee’s quantification of human performance. This effort should be sustained and made an ongoing part of the research program.

The project N6248, “Advanced Reactor HRA Development,” is the first year of a proposed five-year effort to develop HRA methods and tools to support an independent staff review of human reliability analyses submitted as part of new reactor licensing applications. Given the importance of human factors to reactor safety and the likelihood that new reactor designs may significantly alter the role of operators and the human-system interface, this project is valuable and should be continued to completion.

Table 6. Human Factors and Human Reliability Research Activities

Job Code	Title	Comment
Y6497	***HRA Application and ATHEANA Maintenance***	Apply ATHEANA to Fire Risk Requantification; upgrade and improve ATHEANA. ATHEANA is NRC's tool for analysis of human reliability. Application of this tool will allow assessment of its worth.
Y6496	***Human Event Repository and Analysis***	Develop a human event repository and analysis tools. This program develops a useful data-base for comparison to model predictions of human events.
B7488	***Halden Reactor Project***	International collaborative research project that addresses man-machine interaction and verification and validation of software, surveillance and support systems, advanced control rooms and fuels and materials. This international effort helps keep staff aware of international developments in human factors and human reliability.
N6207	***Develop Reg. Guide and Analytical Technique for assessing NPP staffing***	Support development of guidance for staffing exemption requests to 10 CFR 50.54 (m). This is an important program as licensees look at manpower costs associated with nuclear power plant operations.
Y6843	***Develop Human Performance Indicators***	Determine availability and viability of human performance indicators for assessing performance at nuclear power plants; This program was undertaken in response to a Commission SRM.
N6137	***Impact of Operator Workload on Human Performance***	An important new effort to assess the impact of operator overload on operator performance and plant safety.
N6248	***Advanced Reactor HRA Development***	The first year of a proposed five -year effort for addressing human performance issues for new reactors. This is a valuable project and should be continued to completion.
Y6529	***Human Factors of Advanced Reactors***	Develop regulatory guidance and analytical techniques to review human factors for advanced reactors. Essential work to prepare the staff in its review of advanced reactor designs.

9 MATERIALS AND METALLURGY

Research in the area of materials and metallurgy is an important focus of the NRC Safety Research Program. Current research activities are concentrated in five areas:

- Environmentally Assisted Cracking in Light Water Reactors
  (Projects K6266, K6202, Y6270, Y6388, N6007)

- Steam Generator Tube Integrity
  (Projects Y6536, Y6588)

- Non-destructive Examinations
  (Projects Y6534, Y6604, Y6649, Y6869, Y6867, Y6541, N6019)

- Proactive Materials Degradation Assessment
  (Project Y6868)

- Reactor Pressure Vessel Integrity
  (Projects W6953, Y6533, Y6378, Y6638, Y6951, N6204, Y6870, N6223, Y6485, Y6656)

These projects represent a significant investment by the NRC to better understand the issues of materials degradation in the currently operating fleet of nuclear power plants. Such investment is justified in view of significant agency regulatory activities that aging degradation research supports. As plants age, known degradation mechanisms will continue to affect components and new degradation mechanisms may develop. The current program is well focused on improving the agency's ability to independently evaluate licensees' efforts to prevent, detect, and mitigate environmentally assisted stress corrosion cracking. The Proactive Materials Degradation Assessment project is an effort to identify potential material degradation problems before they manifest in operating nuclear power plants.

Unfortunately, the planning of NRC's research in materials and metallurgy is not well documented in the way planning for research on digital instrumentation and control systems has been documented. It is, then, difficult to explain the role and priority of each task within each of the five project areas. In aggregate, the activities in the first four project areas (Environmentally Assisted Cracking, Steam Generator Tube Integrity, Non-destructive Examinations, and Proactive Materials Degradation Assessment) seem to be appropriate. These are the very areas that most challenge the industry and its ability to detect component degradation. The agency must develop the capabilities to assess the acceptability of the industry's initiatives to deal with these degradation challenges. The five project areas are further discussed below.

Environmentally Assisted Cracking

Environmentally assisted cracking is a complicated technical issue that continues to afflict the industry as components age and irradiation effect increases. In recent years, the industry has experienced irradiation assisted stress corrosion cracking (IASCC) of components internal to the vessels of boiling water reactors (BWRs) and stress corrosion cracking of reactor vessel head penetration assemblies in pressurized water reactors (PWRs). Although the industry has responded to these events with initiatives to prevent and mitigate these types of degradation, the event at Davis-Besse makes it readily apparent that the NRC staff must be capable of independently evaluating the adequacy of licensees' initiatives. The research projects now under way seem well designed to ensure that the NRC has the needed technical understanding of the stress corrosion cracking issues.

The project Y6388, "Environmentally Assisted Cracking of LWRs," evaluates environmental effects on fatigue of steels used in light water reactors and provides the NRC with technical data and analytical methods to assess licensees' plans concerning mitigation. The large effort includes tests of neutron-irradiated specimens to improve the understanding of IASCC initiation and stress relaxation. It also provides data on the performance of probes and monitoring techniques in radiation environments. This work is essential and should be continued. A new project, "Investigation of Stress Corrosion Cracking in Selected Materials" (N6007), will develop a better understanding of stress corrosion cracking in PWRs. Such cracking occurs typically in the reactor coolant system boundary. Understanding of such cracking in this boundary is essential for maintaining the defense-in-depth.

Environmentally assisted corrosion of reactor materials is an international concern. The CIR-II Cooperative Agreement (K6202) is a collaboration with the international community for studying the susceptibility of stainless steel to IASCC. Certainly, this collaboration should be continued.

Steam Generator Tube Integrity

Rupture of steam generator tubes in PWRs can lead to accidents that allow radioactive materials released from the core to bypass the reactor containment and enter directly into the environment. Severe accidents involving containment bypass can be risk dominant at some PWRs. Through the years, many modes of corrosion of steam generator tubes have been experienced. Regulations on the corrosion were developed when erosion was the dominant concern. Careful water chemistry control by licensees has largely eliminated erosion as a safety concern. But, now, stress corrosion cracking has emerged as the dominant threat to the integrity of steam generator tubes. Incipient stress corrosion cracking is much more difficult to detect. NRC has two research projects to deal with the degradation mechanisms in steam generator tubes, "Steam Generator Tube Integrity Program" (Y6588) and "PWR Primary System Components Severe Accidents" (Y6536). The first project, Y6588, deals with potential tube degradation modes, their resulting leak rates, and the effectiveness of in-service inspections. The second project, Y6536, seeks to improve methods and models used to predict the behavior of degraded steam generators and other PWR components under severe accident loads. Both of these research efforts are important and should be continued.

Non-destructive Examinations

Non-destructive examinations are relied upon to monitor the integrity of the reactor coolant system. The reliability and effectiveness of existing non-destructive examination techniques remain open to question. Certainly, a steam generator tube cracking incident at the Indian Point reactor emphasizes this point. Four projects are under way to improve non-destructive examination techniques (Y6534, Y6604, Y6649, and Y6869) and this work should continue. Two of these projects deal with the effectiveness and reliability of non-destructive examination of reactor vessel penetration assemblies. As the ACRS noted in NUREG-1635, Vol. 6, this is an area that needs increased attention. A third project will provide destructive examination data that should be of tremendous value for the validation of non-destructive examination methods. The project, N6019, will examine non-destructive methods and leak monitoring techniques and the requirements for light water reactor components that have experienced degradation or have been identified as being susceptible to future degradation. The project "Evaluate Reliability and Effectiveness of Advanced NDE," Y6541, will support continued investigation of innovative methods to detect incipient amounts of wastage of ferritic steel. All of these projects are responsive to the NRC's needs and should be continued.

Proactive Materials Degradation Assessment

The nuclear industry and the NRC have often been surprised by unexpected material degradation problems. As a result, they have responded to such problems in a reactive mode which has proven to be inefficient. Reactive response does not enhance public confidence in the safe operations of nuclear power plants. The project "Proactive Material Degradation Assessment" (Y6868) is an NRC initiative to identify materials and locations in light water reactors where degradation can reasonably be expected in the future. The goal of this project is to develop the technical bases needed to implement regulatory actions to proactively address materials degradation problems. Current inspection and monitoring programs at plants can be reviewed and modified as needed to provide earlier identification of incipient degradation before it affects plant safety. The ACRS admires the vision of this undertaking and supports its continuation. The ACRS looks forward to reviewing the initial results of this ongoing effort soon and learning whether the admirable goal of this project is, in fact, feasible.

Reactor Pressure Vessel Integrity

The integrity of the reactor pressure vessels has been studied for decades. Maintaining the structural integrity of the reactor pressure vessel in a nuclear power plant during both routine operations and during postulated upset conditions, including pressurized thermal shock situations, is a longstanding obligation of licensees. This obligation is codified in three general design criteria (GDC 14, GDC 30 and GDC 31) as well as in 10 CFR 50.61 and the appendices G and H to 10 CFR Part 50. Technical bases for these requirements were largely established in the 1980s. NRC is continuing to devote substantial resources to the study of pressure vessel embrittlement though there does not seem to be a comparable interest within the industry who will have most of the research benefits. Indeed, the number of projects in this area seems to have grown since the ACRS last reviewed the NRC research program and questioned the need for research in the area of reactor pressure vessel integrity.

Some of the activities in this programmatic area deal with the finalization of the NRC's work on pressurized thermal shock which is nearing completion. These activities will contribute to the potential revisions of Regulatory Guide 1.99 on radiation embrittlement of reactor pressure vessel materials and Appendices G and H to 10 CFR Part 50 on fracture toughness requirements and reactor surveillance needed to ensure low probability of reactor vessel failure.

The project "International Pressure Vessel Technical Cooperative Program" (Y6378) will ensure NRC participation in the International Atomic Energy Agency (IAEA) deliberation on reactor pressure vessel integrity.

The NRC's comprehensive program on reactor pressure vessel integrity has produced significant results by providing better understanding of the available margin in reactor pressure vessel components. Revisions to PTS screening criterion in the PTS rule and the associated regulatory guides and Appendices G and H to 10 CFR Part 50 are likely to provide great benefit to licensees by relaxing current requirements and allowing longer life of reactor pressure vessels. These activities should be completed soon.

RES needs to reevaluate the need for continued research into heavy section steel components. This research may be justified if there is a clear need for NRC to develop its capabilities in the area of probabilistic fracture mechanics so that it can evaluate licensees' applications. If this is the case, the research needs to be clearly focused on this objective and not the research that the industry should perform to meet its responsibilities to ensure reactor pressure vessel integrity. It appears now, however, that it is NRC that is advancing the state-of-the-art and making available information that allows licensees to reduce conservatism in their analyses.

Table 7. Research Activities in Materials and Metallurgy

Job Code	Title	Comment
Environmentally Assisted Cracking in LWRs		
K6266	***CIR-II Cooperative Agreement***	NRC contribution to international research on irradiation assisted stress corrosion cracking.
K6202	***Extension of CIR-II Cooperative Agreement***	Assess the susceptibility of stainless steels to Irradiation Assisted Stress Corrosion Cracking. This program allows NRC to stay abreast of international developments.
Y6270	***Environmentally Assisted Cracking***	Provide neutron irradiated specimens for NRC research programs.
Y6388	***Environmentally Assisted Cracking of LWRs***	Develop data on irradiation assisted stress corrosion cracking in PWRs and BWRs. This program provides NRC staff with the data and analytical methods to review licensees' activities and plans to limit corrosion.
N6007	***Investigation of Stress Corrosion Cracking in Selected Materials***	User need for a better understanding of stress corrosion cracking in PWRs. This program supports the regulatory process.
Steam Generator Tube Integrity		
Y6536	***PWR Primary System Components Severe Accidents***	Methods and models to predict PWR reactor coolant system component behaviors under severe accident loads; This is an essential research program.
Y6588	***Steam Generator Tube Integrity Program***	Wide-ranging program in support of the steam generator integrity action plan. ACRS supports this action plan and regularly monitors its progress.

Table 7. Research Activities in Materials and Metallurgy (Continued)

Job Code	Title	Comment
Non-destructive Examinations		
Y6534	***Piping NDE Reliability***	Program addresses Inconel cracking in weld metal and base metal. This is an essential program to ensure licensees adequately monitor nickel alloys in plants.
Y6604	***Evaluate Reliability of NDE Techniques***	Addressing the inspection of cast stainless steel components and dissimilar metal welds; evaluation of reliability and accuracy of in-service inspection. This is an essential program to facilitate NRC monitoring of licensee activities.
Y6649	***Phase II - Alloy 600 Cracking***	Independent assessment of industry analyses of CRDM nozzle cracking. This is a classic NRC program of confirmatory research.
Y6869	***Barrier Integrity Research Program***	Evaluate RCS leakage experience and leak detection capabilities. This is an essential program to facilitate NRC monitoring of licensee activities.
Y6867	***Cooperative Activities Reactor Coolant System Pressure Boundary Components***	Complete non-destructive examinations of nozzles from vessel heads. Plan destructive tests. This is an important program to validate analyses NRC uses in its regulation of licensee activities.
Y6541	***Evaluate Reliability and Effectiveness of Advanced NDE***	Identify innovative NDE techniques in coordination with industry and international community. This program allows NRC staff to stay abreast of international developments in NDE.
N6019	***NDE & Leak Monitoring Requirements***	Assess adequacy of current inspection and monitoring requirements. Assemble data on probabilities of failure of passive components. This is an essential program to facilitate NRC monitoring of licensee activities.

Table 7. Research Activities in Materials and Metallurgy (Continued)

Job Code	Title	Comment
Proactive Materials Degradation Assessment		
Y6868	***Proactive Materials Degradation Assessment***	Identify materials and locations in LWRs where degradation can reasonably be expected. This program is intended to better equip NRC to anticipate materials degradation problems at nuclear power plants. This program should be continued. The ACRS looks forward to reviewing the initial results.
Reactor Pressure Vessel Integrity		
N6204	***Review and Revisions of Pressurized Thermal Shock Reports NUREGs 1806 and 1809***	Support documentation of thermal hydraulics analyses for pressurized thermal shock, and document Calvert Cliffs RELAP5 calculations to support FAVOR calculations. This program should be completed.
Y6485	***Technical Support - Pressurized Thermal Shock Rulemaking***	Support for the pressurized thermal shock rulemaking effort. This is essential support for the regulatory process.
W6953	***Heavy-Section Steel Irradiation Program***	Evaluation of Master Curve methodology for reactor pressure vessels. The ACRS questions the need for the large investment in heavy section steel research.
Y6870	***Cooperative Program on Irradiation***	Development of a cooperative program with DOE to study reactor pressure vessel materials.
Y6378	***International Pressure Vessel Technical Cooperative Program***	International cooperative effort to understand embrittlement of reactor pressure vessels and other components. This program will keep staff aware of international developments in reactor pressure vessel integrity.
Y6533	***HSST-3 (Heavy Section Steel Technology)***	Development of fracture mechanics methodologies; The ACRS questions the need for the large investment in heavy section steel research.

Table 7. Research Activities in Materials and Metallurgy (Continued)

Job Code	Title	Comment
Y6951	***Fracture Mechanics Technology for LWR***	Fracture mechanics of heavy section steel. The ACRS questions the need for the large investment in heavy section steel research.
Y6638	***Statistical Analysis of RPV Steels***	Assist NRC staff in developing a revision to Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials." This research directly supports the regulatory process.
N6223	***FAVOR 4.1 Sampling Validation***	Validation of new features of the FAVOR computer code for fracture analysis of vessels. FAVOR is NRC's computer code for fracture mechanics analysis and is used extensively.
Y6656	***Risk Inform Appendices G & H***	Develop a risk-informed revision to 10 CFR 50, Appendix G on Fracture Toughness Requirements and Appendix H on Reactor Vessel Material Surveillance Program.
N6227	***SMIRT-18 Conference Registration***	Costs associated with presentation of papers on NRC research projects at the Structural Mechanics in Reactor Technology meeting.
N6097	***SMIRT 18***	Financial support to publish proceedings of the 18^{th} International SMIRT conference.

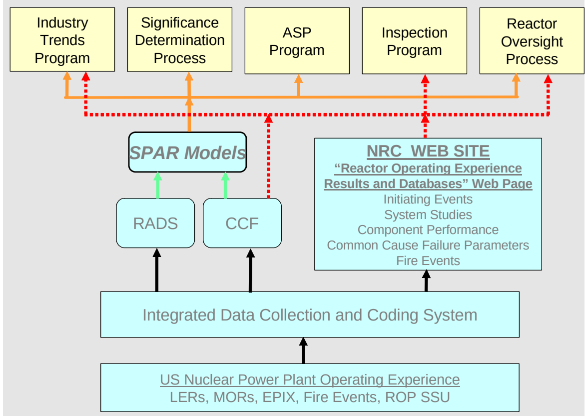
10 OPERATIONAL EXPERIENCE

The analysis of operating data is a cornerstone in the NRC's increased use of risk information in regulatory processes. Such analysis provides current information on initiating events, component failure data, and the risk profiles of licensees. Comparison of these results to goals in the agency's Strategic Plan provides a measure of regulatory effectiveness and inputs for the agency's annual report to Congress on significant operating events.

The NRC research activities associated with operational experience are listed in Table 8. The Accident Sequence Precursor (ASP) Program, Y6815, and the Industry Trends Program, Y6546, alert the staff and industry to component failures as old or replacement components age or operations change. Data derived from operating experience will validate or refute the assumption that aging management programs are sufficient to ensure the operability of both active and passive components. The operating experience programs provide data that can be the bases for regulatory decisions to improve safety. These programs also support the Reactor Oversight Process, including the determination of the safety significance of inspection findings and the development of industry performance indicators.

Two tasks in the research of operational events, "Method to assess Effect of Design and Operations Margins," N6082, and "Procedure Development for External Events," Y6814, are important efforts to extend the use of quantitative risk assessment into external events, including fire, and low power and shutdown operations.

ACRS is supportive of the research activities in the area of operational experience and recommends that these activities be continued. In light of the limited resources



Uses of Operational Data and Analyses in Regulatory Activities

allocated to these tasks, RES has done a commendable job in producing outputs in well-documented and thorough fashion. Tasks that are currently in the 2005 Research Plan related to Operational Experience should remain funded and should be continued for the foreseeable future.

Staff engaged in the collection and analysis of operating experience data might also be able to improve the state-of-the-art in PRA modeling. Specifically, they might be able to use operating experience data to derive higher resolution models of system and component operability. Currently, PRAs use success criteria models. A system or component that meets the success criteria is deemed operable This "go/no go" model is not entirely realistic. There is no assessment of margins, equipment aging, changing plant conditions, etc. Success criteria models may not provide adequate answers for some applications such as power uprates, containment overpressure credit, license

renewal, sump screen clogging, or any set of plant conditions that are in some way off-normal or even outside the design specifications of the equipment. There have been several events that were surprises because the phenomena that caused or contributed to the failure mode had not been realistically modeled. Certainly, the recent Davis-Besse event involving corrosion of the reactor pressure vessel head penetrations comes to mind. Staff granted a small extension to ordered shutdown date for reactor pressure vessel penetration inspections. They did so, in part, because the calculated risk was small. Unfortunately, the phenomenological modeling of the head penetrations and their corrosion was incorrectly used in the risk assessment.

Development of improved models of system and component operability models will require that choices be made concerning areas where improved modeling will yield useful improvements in the risk predictions. The issues of interest may themselves dictate where choices for improved modeling should be made. Some modeling improvements are being made now on an *ad hoc* basis. There is no need to continue to do so if a more structured approach could result in better models with wider applications.

Table 8. Research Activities in Operational Experience

Job Code	Title	Comment
N6082	***Method to Assess Effect of Design and Operations Margins***	Provides a methodology to assess the effects of changes to design and operation on plant safety margins. This program provides direct support for the regulatory process.
Y6468	***Reactor Operating Experience Data for Risk Applications***	Collect operational data for reactor systems, components, initiating events, common-cause failures and fire events. Data collected in this program is of use for validation of PRA models.
Y6546	***Industry Trends Program***	Includes grid concerns. This is an essential program for NRC.
Y6864	***Operating Event Technical Support***	Support for technical expertise in operating events.
Y6816	***SDP/ASP Standardization***	Develop analysis guidelines for operating events during low power/shutdown conditions. This program will extend the ASP program to include events during shutdown operations.
Y6815	***Accident Sequence Precursor Analysis***	Systematically screen, review and evaluate operating events. This is a flagship program at NRC.
Y6987	***Expert Elicitation Process - Accident Sequence Precursor Program***	Develop guidelines for obtaining and using expert opinion in ASP analyses. The useful elicitation of expert opinion is of growing importance in the risk-informed regulatory system.
Y6814	***Procedure Development for External Events***	Expand the scope of ASP analyses to include the calculation of risk from external events and from low power and shutdown modes of operation. This program will help extend the scope of the ASP program.

11 PROBABILISTIC RISK ASSESSMENT

Probabilistic risk assessment has become an essential technology for NRC as it evolves the regulatory system to make greater use of risk information. The NRC research activities in probabilistic risk assessment are shown in Table 9. Probabilistic risk assessment has become pervasive within the research program. Other activities nominally part of the development of PRA are addressed in other Chapters of this report. See especially the discussions of Digital Instrumentation and Control Systems (Chapter 4), Fire Safety Research (Chapter 5), Human Factors and Human Reliability Research (Chapter 8), and Operational Experience (Chapter 10). The staff involved in PRA research has been extraordinarily productive since the last ACRS report the NRC research program. A major focus of the current PRA research is to support the ROP, which uses risk information for monitoring the operations of nuclear power plants and acting on inspection findings and deviation of performance indicators from established thresholds.

The ROP makes heavy use of the SAPHIRE computer code and the SPAR models of specific plants. The SPAR model development program has become an essential element of the ROP. The ability to develop a SPAR model for each nuclear power plant has only been feasible because of the existence of Level I, internal events, PRAs for each plant. Each SPAR model begins with a basic model of a plant system for a generic category of plants (e.g., a BWR4 reactor with a Mark I containment). The SPAR model is then made plant specific through upgrades based on discussions with the licensee. NRC has found it essential to develop its own risk-assessment model for each plant as a matter of practicality. It would be difficult for the NRC staff to take a variety of plant PRAs, which use different platforms and approaches, make them operational at NRC, and have knowledgeable staff available to execute and update each plant model. NRC development of SPAR models for individual plants has also enhanced the plants' risk assessments.

A major issue that confronts the use of risk information in nuclear power plant regulation is the question of incompleteness of individual plant risk assessments. The Individual Plant Examination (IPE) program and subsequent evolutions at the nuclear power plants led to development of Level I, internal events, PRA models of all of the operating. These PRAs meet (or with modest effort can meet) the requirements of industry standards for internal events PRAs. The same is not true for the assessment of risk from fires, floods, seismic events and for plant modes of operation that differ from full power operations. Furthermore, the capabilities to assess risk at Level II, radionuclide release and source terms , lag far behind the Level I capabilities.

The NRC staff has plans to expand the scope of the SPAR models to include treatment of risks from fire-initiated events, seismic events and shutdown modes of operations. These plans are, however, not well developed. There is furthermore the question of availability of resources needed to undertake these efforts. The expansions of the scope of SPAR models will be challenging because all licensees do not have sophisticated risk assessments in these areas for comparison and validation of NRC's SPAR models with expanded scope. The NRC staff could develop generic models accounting for the major features of the plant designs, but the staff would not be able to upgrade the generic models to become plant-specific models as was done for the treatments of risk from internal events. In addition, fire and seismic risk assessments differ qualitatively from internal events risk assessments since the events occur in "areas" of a plant and affect multiple systems rather than just specific components in specific systems. Fire and seismic risk assessments require detailed knowledge of spatial relationships in addition to functional relationships. Spatial relationships, of course, vary substantially even among plants of the same generic type. Despite these challenges, the regulatory oversight value of full-scope SPAR models is very high. Over the next year, the staff should develop its approach and plans for the expansion of the scope of the SPAR models to treat external events, shutdown modes of operation and even to go to Level II analyses that include accident progression and the release of radionuclides to the environment. Even if it is not possible to have plant-specific models in the near term, the generic shells should be available and can be adapted to be plant specific in the future or can be upgraded in particular areas to address specific regulatory issues.

Another barrier to the greater use of risk assessment in the regulatory process is the question of uncertainty in the risk predictions. There are, of course, parametric uncertainties and the agency has active programs to better understand the important parametric uncertainties (See especially Chapter 10, Operational Experience). There are also issues of uncertainty in the models adopted in PRA. Uncertainties in the models of human reliability and passive system reliability are significant examples. It has become common now for the NRC and the licensee to agree upon a model appropriate for particular regulatory activities. This agreement can often be based on familiarity or expedience. The disturbing trend is for the staff to conclude, then, that there are no longer uncertainties associated with the results predicted by the agreed upon models. Staff needs to ensure that it treats uncertainty in risk assessments in a more defensible manner. Research needs to provide the tools and understanding so that this can be done.

The staff has also been revising 10 CFR 50.46 to account better for risk information. This is challenging and important work. Even more challenging is the effort to develop a "technology-neutral" alternative to the current regulatory framework. The ACRS views such a technology-neutral regulatory framework as essential in the future and feels that it needs more attention.

Altogether the scope and the number of PRA research activities are quite impressive. The ACRS cautions, however, that NRC should not allow its work in such a crucial technology as risk assessments become totally devoted to the support of line activities. Methods development is still important. As an example, the ACRS notes that considerable research is being reported in the literature regarding Binary Decision Diagrams as tools for solving large fault tress without resort to cutoff frequencies as is now done. Some researchers report that the unavailability of highly redundant systems could be underestimated significantly when cutoff frequencies are used for the analysis. Although no definitive evidence has yet been produced to show that methods used in the NRC's SAPHIRE code are inadequate, the staff needs to review the literature concerning Binary Decision Diagrams and evaluate the need to adopt this technology. The growing importance of the SAPHIRE code and the SPAR models in the regulatory process warrants such an investigation.

Table 9. Probabilistic Risk Assessment Research Activities

Job Code	Title	Comment
N6027	***PRA for Dry Cask Storage Follow Up***	A variety of tasks including uncertainty analysis and extension to multiple casks. This program supports licensing and inspection oversight of cask vendors.
N6105	***Guidelines for the Communication of Risk Information***	Complete the technical basis for the internal risk communication guidelines. This task completes the technical basis for internal risk communication guidelines. The ACRS remains concerned that publically available information on risk analyses may not be sufficient to ensure public confidence in a risk-informed regulatory process.
Y6842	***Guidance for the Development of Latent Errors***	Quantitatively assess the importance of latent errors and the treatment of latent errors in PRAs. This project has been deferred until FY2007. The ACRS cautions that operating experience shows that latent errors may be four times more common than active errors in important reactor events. The work should not be deferred further.
J8263 **Y6370**	***Reactor Oversight Process Support*** ***Development of Risk-based Performance Indicators***	Development of performance indicators to be incorporated into the ROP. Support for the Mitigating Systems Performance Index. These programs support the ROP.
Y6626	***Access to INPO's EPIX System***	Data-base on equipment performance and reliability.
J8258	***International Common Cause Exchange Project***	Sharing of data on common-cause failures with the international reactor safety community. This program keeps staff abreast of international findings concerning common-cause failures.
N6008	***Passive Components Conditional Core Damage Probability***	This program should prioritize passive components for consideration in the proactive materials degradation assessment (Project Y6868, Materials and Metallurgy, Chapter 9).

Table 9. Probabilistic Risk Assessment Research Activities
(Continued)

Job Code	Title	Comment
Y6153	***SPAR Model Development: Level2/LERF***	Develop SPAR models for evaluation of large early release frequencies.
N6090	***SPAR Model Development: Shutdown Models***	Develop logic models for analyzing low power and shutdown internal events.
W6355	***SPAR Model Development: Low Power Shutdown***	Identify methods to characterize risk during low power or shutdown operations.
W6467	***SPAR Model Development: Level 1 Rev. 3 Models***	Revision of Level 1 SPAR models to better reflect as built and operated plants.
Y6595	***SPAR Model Development: External Events Analysis***	Development models of external events for the SPAR codes
N6075	***SPAR Model Development: Enhanced Level 1, Revision 3 Models***	These are important programs to support the expanded scope of the SPAR models.
Y6394	***Maintain and Support SAPHIRE Code and Library of PRA***	Testing to ensure that SAPHIRE is a state-of-the-art PRA code.
N6172	***Participate in the MERIT Program (Maximizing Enhancements in Risk Informed Technology)***	Base program supports risk informing 10 CFR 50.46 and includes development of a probabilistic LOCA code, non-piping component degradation, and pressurized water stress corrosion cracking. This international program supports one of the important NRC initiatives.

Table 9. Probabilistic Risk Assessment Research Activities
(Continued)

Job Code	Title	Comment
N6111 **Y6538**	***Technical Support for 10 CFR 50.46 Task Order 3*** ***Technical Development of LOCA Frequency Distributions***	Quantification of the effect of break size reduction and alternative break locations on margin to existing alternate acceptance criteria Provide LOCA frequency estimates for use in revision of 10 CFR 50.46. These programs are needed to support risk informed revisions to 10 CFR 50.46.
K6081	***PRA Techniques in Risk-informed and Performance-based Regulation***	Develop methods for uncertainty analysis for risk-informed purposes. This is a cooperative agreement with a broad scope. In addition to potential methodological contributions it has an educational value.
N6107	***10 CFR 50.48c related Technical Activities***	In collaboration with EPRI, develop a comprehensive set of risk methods, tools and data to understand and evaluate risks from fires.
W6224 **Y6492**	***Risk-informing Part 50*** ***Assess Possible Part 50 Risk-informed Changes***	Develop recommendation on changes to 10 CFR Part 50 to make it risk-informed. Develop recommendations to specific requirements in 10 CFR Part 50 to make them risk-informed. These program support the initiative to risk inform 10 CFR Part 50.
W6970 **W6971**	***Support to Develop Consensus PRA Standards*** ***Support in Development of Consensus PRA Standards***	Provide guidance on the use of industry standards for PRA. Revise Regulatory Guide 1.200 based on industry pilots and Revision 1 to ASME PRA standard. These program support the Commission’s phased approach to PRA quality.

Table 9. Probabilistic Risk Assessment Research Activities
(Continued)

Job Code	Title	Comment
Y6103	***Low Power and Shutdown Risk Study - Level 2***	Program to extend the scope of SPAR models to include accident progression for accidents initiated during shutdown operations. Premature at this point.
N6133	***Development of Consensus on PRA***	Support for staff in development of ANS Low Power and Shutdown operations PRA Standard.
N6134	***Low Power/Shutdown Level 1 and Fire Risk Standard***	Project provides support for staff involvement in the development of ANS standards on PRA for low power/shutdown operations and fire-initiated events.
Y6371	***Risk Associated with Cable Aging***	Addresses the inclusion of aging effects into PRA.

12 SEISMIC RESEARCH

As the design of nuclear power plants improves, the seismic hazard and seismic response of the plants can make an increasingly important contribution to risk. Seismic hazard analysis and structural response are not areas where NRC must maintain state-of-the-art expertise. Such expertise is available to the NRC on a contractual basis. As noted in our previous report, seismic research activities at NRC can be confined to support needed updates to regulatory guides and collaborative work with the international community to stay abreast of developments in other Countries. The current research program is, indeed, largely focused on needs of the regulatory process and a few important international collaborations.

Table 10. Seismic Research Activities

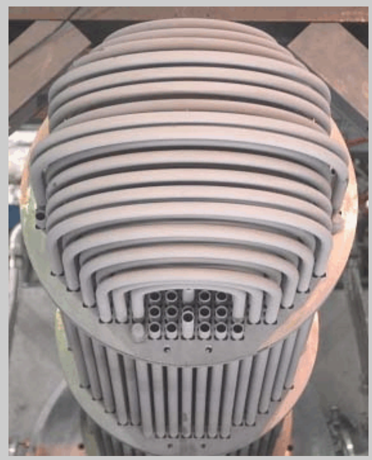
Job Code	Title	Comment
N6020	***Seismic-induced Passive Component LOCA Frequencies***	Review of work by national laboratories and industry on piping degradation and failure under earthquake loads; Work being done to upgrade Regulatory Guides.
Y6481	***SSHAC Method***	10-year update of the Probabilistic Seismic Hazard Assessment used in evaluation of early site permits; work to support update required by regulations.
Y6718	***Soil-structure Interaction for Buried Structures***	Review adequacy of current NRC guidelines concerning soil-structure interactions; work to update Regulatory Guides.
N6112	***Evaluation of Seismic Siting***	Review of ASCE Standard 43-05, "Seismic Design Criteria for Structures, Systems and Components in Nuclear Facilities."
N6076	***Japanese Collaboration on Seismic Issues***	Collaboration with Japan on seismic tests and analyses; Collaborative work give NRC access to extensive work underway in Japan.
W6081	***Japanese Collaboration on Seismic Issues***	Supports work in U.S. in connection with collaboration.
N6102	***Reg. Guide 1.165 Update Technical Basis***	Review of technical advances in the development of seismic response spectra; prepare draft revision to Regulatory Guide 1.165.
N6103	***Enhancement of the CARES Code (Computer Analyses for Rapid Evaluation of Structures)***	The CARES computer code is used to predict the free field and structural response to seismic input.
N6219	***Resolve Regulatory Guide 1.92 Public Comments***	Regulatory Guide provides up-to-date guidance for using the response spectrum and time history methods for estimating seismic response of power plants.
N6104	***Ground Motion Seismic Hazard Studies***	Collection and review of new data on the propagation of earthquake motion in the Central and Eastern U.S.; work to support required update in regulations.
Y6796	***IAEA Coordinated RES Project on Seismic Ground Motion***	NRC contribution to international effort to understand earthquake effects on nuclear power plants. Collaborative effort keeps NRC staff abreast of any international developments.
Y6757	***Containment Capacity Studies***	Confirmatory analyses of structural response and failure modes of containments under extreme loading including seismic loads.

13 SEVERE ACCIDENT RESEARCH

In the past, NRC invested heavily in the experimental and analytical characterization of severe reactor accidents. A substantial technology has been established to understand the progression of severe reactor accidents and the radiological consequences of such accidents. Once its immediate needs were met to understand severe reactor accidents sufficiently well to estimate risks to the level of confidence needed to provide assurance of adequate protection, the NRC substantially curtailed its investments in severe reactor accident research. The current NRC research activities in the severe accident area are listed in Table 11.

Research on severe accidents has been continuing in other countries. Substantial programs are under way in both Europe and Japan. NRC has developed an effective strategy to maintain the technology for severe accident analysis and to update this technology with research results from international programs. The body of knowledge coming the NRC's past work and the ongoing international work are systematized in the useable form in the MELCOR accident analysis code. At the same time, the NRC is entering into international cooperative research programs to obtain data for validating the MELCOR code and improving its accuracy and realism. NRC provides the Cooperative Severe Accident Research Program (CSARP) as a forum for the exchange of severe reactor accident information among Countries. One outcome of this focus of the NRC's research into severe reactor accidents is that many Countries and institutions have adopted the MELCOR code as the preferred tool for the severe accident analysis.

A new version of the MELCOR code has been released to users. NRC is collaborating with researchers in Russia to modernize MELCOR to use FORTRAN 95 coding. MELCOR is being used for licensing actions. The capabilities developed to perform detailed



Aerosol Trapping in a Steam Generator (ARTIST)

NRC is participating in ARTIST international cooperative research program to conduct an experimental study in Paul Scherrer Institute in Switzerland to measure the aerosol removal on secondary sides of steam generators during severe accidents at PWRs that bypass reactor containments. Such bypass accidents are often risk dominant for PWRs. The high risks associated with such accidents may stem from conservatism in the aerosol decontamination assumed in accident analysis models for steam generators. Test results are expected to provide the basis for more realistic analyses of these accidents.

parametric uncertainty analyses with the code are especially attractive.

RES is also maintaining the MACCS code for the analysis of consequences of accidents at nuclear facilities. This code is widely accepted in the U.S. as a tool for consequence analysis. Its maintenance at near the state-of-the-art is important to the agency and the ACRS is supportive of the current research programs.

Collaborative severe reactor accident research programs that NRC has joined are making good technical progress and there have been notable accomplishments in the last 2 years.

- **PHÉBUS - FP**

The Phébus-FP program consists of large-scale prototypic experiments involving the degradation of irradiated reactor fuel, release of fission products as vapors and aerosols, and transport of these fission products through a model of a reactor coolant system into a model of a reactor containment. These are the most prototypic and most comprehensive severe accident experiments that have ever been performed. The last of these tests was completed recently. The experiments have proved to be invaluable for the validation and improvement of the MELCOR code and the validation of the alternative source term used for a large number of licensing actions. The program has revealed a number of unanticipated phenomena and refined understanding of other phenomena. NRC has joined a second-generation program that will involve about 15 Nations to conduct separate effects tests to further understand the important accident phenomena revealed in the PHÉBUS-FP test program. This follow-on program addresses the containment chemistry of radioactive iodine, fission product chemistry in the reactor coolant system, the effects of boron carbide control rods on core degradation and fission product chemistry, and the release of fission products from high-burnup fuel and MOX fuel.

- **ARTIST**

The ARTIST test program is an international collaborative effort undertaken in Switzerland to ascertain the amount of decontamination that can occur in the secondary side of steam generators in PWR accidents initiated by steam generator tube ruptures or initiated by other means but involving steam generator tube ruptures. Such accidents have been found to be risk dominant for some PWRs. During last year, the scoping test program has been completed. Results of the tests show that decontamination is modestly larger than what had been anticipated in accident analyses. Plans are being formulated now to conduct integral system tests and additional tests to support modeling of secondary side decontamination.

- **MASCA**

The MASCA test program and its predecessor the RASPLAV program were undertaken to understand the technical feasibility of retaining core debris within reactor pressure vessels, especially with water flooding the outside of the vessel. These programs were conducted in Russia and involved the development of technology to produce large scale melts of prototypic core debris involving UO_2, ZrO_2, and Zr. The major tests in the program have now been completed. Efforts are under way to identify and maintain the experimental capabilities that have been developed for the MASCA program since these capabilities may be essential for the investigation of severe accidents in reactors that do not use light water technology.

OECD-MCCI

This is an international collaborative experimental study being conducted at the Argonne National Laboratory to investigate the viability of using an overlying layer of water to cool core debris interacting with structural concrete. This program is nearing completion.

Planned modifications of the MELCOR code to address the ACR-700 have been curtailed since the application for certification of this reactor has not been submitted. There still may a need to upgrade the modeling of iodine chemistry in reactor containments to respond to recent findings concerning the effects of tri-sodium phosphate buffer in reactor sumps on sump pump screen blockage.

The ACRS is very supportive of the strategy NRC has developed to maintain and update its capabilities for severe accident analyses.

The leveraging of resources through international collaborative experimental research is especially important. The planned extensions and continuations of current collaborations are well worth the investment. This type of collaboration in experimental research could be emulated in other NRC research areas such as fire safety research and thermal-hydraulics research.

Table 11. Severe Accident Research Activities

Job Code	Title	Comment
Y6321	***Benchmark, MOX Fuel Release, Source Term Experiments***	International Collaborative follow-on to the PHEBUS-FP experiments.
Y6328	***Assessment and Analysis of PHEBUS-ST***	In-kind support for the follow on to the PHEBUS-FP experiments. This work is providing data on fission product behavior during reactor accidents for use in MELCOR development.
Y6628	***Consequence Models and Uncertainty Assessment***	Uncertainty analysis of the MACCS code for computing reactor accident consequences.
Y6313	***OECD-MCCI Program***	International collaborative research on the interactions of core debris with concrete. This program should be completed next year
Y6690	***Analysis Support for OECD-MCCI Program***	In-kind and financial support for the international collaborative research on ex-vessel core debris interactions with concrete.
Y6312	***MASCA Program***	International collaborative research on the behavior of molten core debris in the lower plenum of a reactor vessel. This program has resolved safety issues with respect to invessel retention of core debris. The program has developed the capability to produce and test large-scale melts of uranium dioxide that may be of use in advanced reactor safety model development and validation.
Y6802	***MELCOR Severe Accident Code Development and Assessment***	Computer model for the analysis of severe reactor accident and repository for severe accident research results. This is the agency tool for Level 2 PRA including source term characterization; MELCOR is the repository for severe accident research results obtained by the agency.
Y6721	***AGT W/IBRAE-RAS on Nuclear Safety Analysis Codes***	Support for Russian investigators in the development of a FORTRAN-95 version of MELCOR. This program is modernizing the coding in MELCOR by cost-effective use of expertise in Russia.

Table 11. Severe Accident Research Activities
(Continued)

Job Code	Title	Comment
Y6848	***High Burnup Fission Product Release Data***	Refine release models in MELCOR for the effects of high fuel burnup; code analyses will be used to create a licensing source term applicable to high-burnup fuel and reflecting improved modeling of severe accidents.
Y6517	***High Burnup Source Term for Storage***	Establish the technical basis for the extension of regulatory guide on spent fuel heat generation in a spent fuel storage facility to include high-burnup fuel
Y6504	***Steam Generator Fission Product Retention***	International collaborative research on the retention of aerosols on the secondary sides of steam generators in containment bypass accidents (ARTIST program). This program provides an experimental resolution of a long-standing issue of source terms from accidents that bypass containments.
Y6607	***Support ARTIST Tests***	In-kind support for the ARTIST program - see Y6504 above.
Y6486	***Severe Accident Initiated Steam Generator Tube Rupture Sequences***	Investigation of the potential for induced steam generator tube failure during severe accidents leading to containment bypass. This is an important part of the Steam Generator Action plan and the analysis of plant behavior under accident conditions.
Research Programs to Maintain the MACCS Code for Consequence Analysis		
Y6785	***Plume Model Adequacy Evaluation***	Test the assumption that simple plume treatments in MACCS code are adequate by comparing with the state-of-the-art dispersion model. This activity is important to show MACCS is adequate for regulatory needs.
Y6628	***MACCS Uncertainty Assessment for Consequence Models***	Support for emergency planning.
Y6469	***Evaluation of Radionuclide Pathways and Uptakes***	Upgrade information on uptake pathways. This project upgrades the code to take advantage of more recent information.

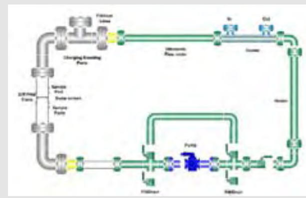
14 THERMAL-HYDRAULICS RESEARCH

Thermal hydraulics, especially the dynamics of two-phase flow, have always been essential elements of the regulatory evaluation of design basis accidents. NRC confirmatory evaluation of licensees' submittals in the area of thermal hydraulics has long been a major element of many licensing actions. Thermal-hydraulic analyses have grown ever more sophisticated. This trend is likely to continue for existing plants as licensees seek power uprates and take advantage of NRC's willingness to allow best-estimate analyses (with scrupulous attention to uncertainties) in the place of deliberately bounding, conservative analyses. To evaluate the adequacy of the licensees' analyses, NRC must have state-of-the-art thermal-hydraulic computational tools and equally sophisticated understanding of both thermal-hydraulic phenomena and the limitations of computer codes. NRC attempts to maintain its competence in the thermal-hydraulic field through its research program.

Major elements of the current NRC thermal-hydraulics research program can be grouped into three general areas:

- PWR sump screen blockage issues
- TRACE computer code development
- Experimental studies of thermal-hydraulic phenomena

These major features of the current thermal-hydraulics research program are discussed below.




Chemical Effects/Head-Loss Tests in a Simulated PWR Sump Pool Environment

GSI-191 addresses the potential for debris accumulation on PWR sump screens to affect emergency core cooling system (ECCS) pump net positive suction head margin. In response to a concern expressed by the ACRS, RES has initiated a program to investigate the potential for chemical reactions that can occur in the containment pool to produce chemical products that can increase the head losses over those due to the physical debris alone.

NRC and the nuclear utility industry jointly developed an Integrated Chemical Effects Tests (ICET) program to determine if chemical reaction products can form in representative PWR post-LOCA containment sump environment. These tests were conducted by Los Alamos National Laboratory (LANL) at the University of New Mexico (UNM). Chemical products were observed in all five test series.

A head–loss loop was set up at Argonne National Laboratory (ANL) to investigate the potential head loss associated with the chemical products observed in the ICET tests.

These recent research results indicate that a simulated pool environment containing phosphate and dissolved calcium can rapidly produce a calcium phosphate precipitate that , if transported to a fiber bed covered screen, produces significant head loss.

PWR Sump Screen Blockage

The sump screen blockage issue for PWRs is the analog of a previous issue identified for BWRs. Debris from coatings and insulation can be generated during the high-pressure blowdown of the reactor coolant system following a major pipe break. This debris can clog the screens protecting the intake pumps for the emergency cooling system and prevent adequate coolant flow. Blockage issues have been exasperated by the discovery of mechanical and chemical effects that magnify the blocking effects of debris trapped on the sump screens. As a result, it is difficult to design screens that are of sufficient size to ensure emergency core cooling. The industry is looking to the NRC for guidance on acceptable methods for sizing screens to protect the sump intakes of the cooling pumps.

The NRC is still in the exploratory phase of research on sump screen blockage. It is still identifying phenomena that affect blockage. It is far from developing tools and methods that can be used with confidence for making predictions. NRC staff is now analyzing the licensees' responses to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors." These responses should reveal the licensees' views of current predictive techniques and their applicability, as well as indicate what methods they expect to use to assess the adequacy of their current and modified screen systems. The NRC staff needs to have sufficient technical knowledge to evaluate these methods. Current NRC research is focused on significant gaps in knowledge, establishing what phenomena play significant roles, and on developing general awareness of what analytical steps are needed to describe the phenomena adequately. The ACRS would expect that many of the details of predictive methods, such as the coefficients in correlations, computational schemes, and methods for developing suitable conservatism to account for uncertainty, could be left to the licensees or to industry-sponsored organizations such as EPRI. This is possible, however, only when the phenomena are well understood and a technical basis has been established for their prediction. When this is not the case, the NRC may need to develop sufficient predictive ability of its own to achieve authoritative competence to evaluate licensees' submittals.

For example, the NRC-sponsored research has revealed the "thin bed effect". This appears to involve a dense agglomeration of fine particles that fill the pores in a layer of debris, such as fiberglass, but the mechanism by which it occurs and how it influences the pressure drop are not understood. Previous NRC acceptance of pertinent Nuclear Energy Institute (NEI) guidance now appears premature in light of confirmatory research that has revealed much larger influence of the bed structure (e.g. up to a factor of about 100 on pressure drop for the same mix of fibers and particles) than had previously been thought to be possible. Research in this area should be continued and expanded as needed in order to reduce the very large uncertainties surrounding these effects and to determine if a predictive capability is feasible.

Other important phenomena, such as chemical and downstream (of the screen) effects are now being investigated by RES. These are essentially exploratory studies that have uncovered some significant effects, but have yet to reveal their scope and magnitude. Predictive capability remains to be demonstrated. The NRC needs to evaluate the results of these studies and determine how much it can rely on the nuclear industry to develop reliable predictive tools and how much independent predictive capability it requires. Development of a predictive capability may require investment of substantial resources and time.

TRACE Computer Code Development

Several years ago, the NRC recognized that it could not sustain the continued maintenance of several thermal-hydraulic codes for each general type of nuclear power plant. It elected to consolidate its existing codes for the confirmatory analysis of licensee submittals on design basis thermal-hydraulic issues into a single code now called TRACE. The consolidation is now largely completed. The TRACE computer code is viewed by the NRC research staff as “as good as anything else that is out there.” The long-term validation and improvement phase of code development is at hand. Current research is devoted to improving features of the TRACE code, making it easier to use and validating it against available data. Some of the data already exist and other data are being generated. In addition, the integration of the TRACE code, coupled with the CONTAIN code to model containment response and the PARKS code for neutronic analyses into the regulatory processes of the agency has begun.

The TRACE code is reputed to now be able to serve as the “workhorse” thermal-hydraulic analysis code for the agency. In the course of its work to consolidate thermal-hydraulics codes into TRACE, the research staff has found many ways to improve the code. Such improvements should be done. Now, however, it is far more important that the integration of TRACE into the regulatory process be completed in an expeditious manner. The research staff working on the development needs to have input from users of the code on needed features and capability of the code. Inevitably, the introduction of a new computational tool will slow and detract the regulatory process for some transient period. There is no way to counter this difficulty associated with the introduction of a new computer code. It must be endured and the sooner this is done, the sooner the challenges associated with the use of a new code in the regulatory process can be overcome. Once TRACE is integrated into the regulatory process, the developers will receive valuable advice on how their efforts to improve the code should be directed to enhance the regulatory process.

Highest priority should be given to the integration of TRACE code into the regulatory process. As this integration progresses, the research staff can continue its efforts to improve and further develop TRACE on a “time available” basis. The ACRS is concerned that efforts to improve TRACE lack prioritization and defensible organization. Placing the TRACE code in the hands of users will also identify a host of needed improvements. Prioritization of technical improvements might be aided substantially by commissioning a detailed peer review of TRACE. To do this, the staff will have to have available code documentation of outstanding scope and quality. Such high quality code documentation will also be needed if the code is to become part of the regulatory process. Code documentation, then, is a task that ought to take precedence in the thermal hydraulic research effort.

Experimental Studies of Thermal-Hydraulic Phenomena

Thermal-hydraulic phenomena involving the flow of two-phase mixtures of steam and water are very complicated especially those involving blowdown from high pressure systems. Thermal-hydraulic phenomena that arise in advanced light water reactor designs that emphasize passive response to accidents are driven by subtle forces that require sophisticated understanding to ensure plant safety. As a consequence, NRC has long felt that it cannot rely solely on computer code projections of thermal-hydraulic phenomena to ensure adequate protection of the public health and safety. Experimental confirmation is also required. As the computer models used to analyze thermal-hydraulic phenomena have become more sophisticated, the experiments needed to validate model predictions have become progressively more integral in nature.

Experimental facilities have become larger and more complex. RES has an interest in maintaining these facilities for use in addressing future as well as current regulatory issues. Maintenance of large, complex experimental facilities has become a significant expense in this research area. The major experimental facilities used by NRC in the U.S. are the APEX and PUMA facilities as well as RBHT facility at Penn State University. Abroad, NRC is conducting tests at the PKL facility, the SETH tests and tests at the ROSA facility. Additional experimental needs may arise in connection with the design certification of the ESBWR.

APEX is a medium-size, scaled, integral test facility that proved useful for the certification of the AP600 and AP1000 reactor designs. It has been modified to provide data crucial to the analysis of thermal shock to reactor vessels. It is proposed now that the APEX facility be used for confirmatory analyses for AP1000 and for some “thermal hydraulic integral experiments.” These proposed applications would benefit from review to assess their focus and applicability.

PUMA is a medium size, scaled facility especially suited for evaluating passive emergency core cooling systems. It is being modified to be applicable to testing the emergency core cooling systems for the ESBWR.

The RBHT test program has been under way for a number of years with the purpose of improving core reflood models that are a key part of evaluating the adequacy of pressurized water reactor emergency core cooling systems. The reflood models may become critical if applications are submitted for large power uprates in PWRs. The proposed research program at the RBHT facility needs evaluation to see if the quality, scope and detail of the data are properly matched to the proposed uses of these data.

NRC has wisely not sought to duplicate large test facilities available overseas. Use of these facilities is possible through international programs. The SETH program was useful for resolving Generic Safety Issue (GSI) 185 and assessing the emergency heat removal systems in the ESBWR. Future work under this program at the ROSA and the PKL facilities in support of the TRACE code needs to be more clearly focused.

It is essential for NRC to maintain an ability to assess thermal-hydraulic phenomena that occur both in existing reactors and in future reactors. It is evident that the development of computer codes to predict thermal hydraulic phenomena and the experimental validation of these predictions will grow more burdensome with time. Major development efforts can be anticipated if very innovative designs using coolants other than water are brought forward for certification. It is not likely that the nuclear institutions of any one country will be able to develop adequate codes and conduct sufficient validation of these codes alone. International cooperative development of codes and conduct of experiments appear essential as NRC research moves beyond TRACE with its current capabilities and especially if analyses are needed for coolants other than water. NRC already takes substantial advantage of international experimental capabilities. Extending this international flavor in thermal-hydraulics research to include the development of computer codes will contribute to current ideas of multi-national design approval process. It may slow code development. It also may ensure that sufficient resources for code development are available so that it is feasible to meet the more exacting standards that are likely to be demanded in the future.

Table 12. Thermal-Hydraulics Research Activities

Job Code	Title	Comment
N6106	***Confirmatory Head Loss Testing***	Experiments to measure head loss across sump pump strainers in PWRs.
Y6871	***PWR Sump Screen Penetration and Throttle Valve Testing***	Experiments to determine the type and quantity of debris that can pass through typical PWR sump screens.
N6100	***Head Loss Testing***	Assess the susceptibility of recirculation screens to debris blockage during design basis accidents.
Y6999	***Integrated Chemical Effects Tests***	Five tests to determine representative chemical and material environments in PWRs that can contribute to sump blockage.
N6121	***GSI-191 Chemical Effects Simulations***	Experiments to determine chemical effects that can contribute to sump screen blockage.
N6198	***Transportability of Coatings***	Parametric study to ascertain if coatings can be transported to sumps under accident conditions.
N6083	***BWR ECCS Suction Concerns***	Technical assessment of Generic Issue 193 “BWR Suction Concerns.”
Y6769	***PUMA Test Facility***	Facility for the conduct of thermal hydraulics tests. This facility can produce data for natural circulation systems for use in ESBWR design certification.
Y6852	***PWR Thermal-Hydraulics Integral Experiments***	Tests at the APEX facility at Oregon State University.
N6042	***OECD/ROSA Program***	International collaborative tests of reactor accident thermal hydraulic phenomena.
Y6945	***Rod Bundle Heat Transfer Test Program - Phase 3***	Experiments at Penn State University in support of TRACE code analyses of small and large break loss of coolant accidents. To date, there is little evidence that data from this facility can be of value for TRACE code development. Further work in this facility should be scrutinized carefully to assure that it meets agency needs.

Table 12. Thermal-Hydraulics Research Activities
(Continued)

Job Code	Title	Comment
Y6589	***Thermal-Hydraulic Research***	Perform analytical and small-scale experimental work in support of the TRACE code. Neutronic work in this program in nearly complete. Long-range thermal hydraulic work needs to be shown necessary for agency needs.
N6043	***Thermal-Hydraulic Sub-channel International Standard***	Analysis for international standard problem for a BWR subchannel benchmark.
Y6571	***SETH Program - Test Facilities***	Thermal-hydraulics tests in two international efforts: PKL on boron dilution and PANDA in support of ESBWR certification.
Y6974	***OECD-PKL Program and Test Facility***	International collaborative research on boron dilution accidents including mid-loop operation.
N6213	***TRACE Verification and Validation***	Verification and validation of the TRACE thermal-hydraulics analysis code. This work is viewed as vital to the verification and validation of TRACE.
Y6673	***TRAC-M Development and Assessment - Small LOCA Processes (In the past, the TRACE code was called TRAC-M)***	Simulate separate effects tests with the TRACE code and show acceptable agreement with predecessor codes. Good progress has been made in this important work.
Y6666	***Advanced Numerical Methods in TRAC-M (In the past, the TRACE code was called TRAC-M)***	Advanced numerical methods for the TRACE code. This work is not essential for the current range of efforts to make TRACE useful to the agency.
N6147	***TRACE Development and Assessment Against Specified Tests***	Use TRACE code to evaluate level swell tests done at several facilities. This is a small part of the TRACE validation and verification effort.
N6201	***Gravity Reflood and SBLOCA TRACE Assessment***	Use the TRACE code to assess PUMA facility tests. This work necessary to lend credibility to TRACE for ESBWR analysis.

Table 12. Thermal-Hydraulics Research Activities
(Continued)

Job Code	Title	Comment
Y6525	***TRAC-M Code Maintenance (In the past, the TRACE code was called TRAC-M)***	Maintenance of the TRACE code. This is an essential activity.
N6040	***Data Acquisition***	Recover old input decks for the TRAC-PWR model.
N6072	***Implementation of ACR-700 (Misleading title, Project deals with PUMA input deck)***	This work is no longer necessary.
Y6198	***Continuation of Support for System Code Analysis***	Support for the SCDAP/RELAP5 computer code and the analysis of steam generator tube rupture accidents.
Y6392	***Maintenance, Application, Assessment and Development of NRC Computer Codes***	Consolidation of RELAP5 capabilities into TRACE. This work appears to overlap most of the TRACE development tasks. Incorporation of RELAP capabilities into TRACE has proven difficult because of code philosophy differences.
Y6667	***SNAP Implementation***	Graphical user interface for TRACE and other NRC computer codes. This work is important because of poor direct input methods inherited in TRACE from the underlying TRAC models.
Y6662	***AP1000 Confirmatory Thermalhydraulics Analysis***	Confirmatory thermal hydraulic analyses of a wide range of design basis accidents hypothesized to occur in AP1000. This work is complete.
Y6526	***Administer CAMP Meeting***	Meeting of users of NRC thermal-hydraulics codes. This program will assist in the international acceptance of TRACE.
N6030	***Flow-induced Vibrations and Effects on BWR components***	Analysis of component vibration that can lead to fatigue failure in BWRs.

15 REFERENCES

1. U.S. Nuclear Regulatory Commission, “Review and Evaluation of the Nuclear regulatory Commission Safety Research Program,” Advisory Committee on Reactor Safeguards (ACRS), NUREG-1635, Vol. 6, March 2004.

2. Code of Federal Regulations, Title 10, Part 50 (10 CFR 50), “Domestic Licensing of Production and Utilization Facilities,” U.S. Government Printing Office, Washington D.C., 2005.

3. 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors,” U.S. Government Printing Office, Washington D.C., 2005.

4. 10 CFR 50.48, “Fire Protection,” U.S. Government Printing Office, Washington D.C., 2005.

5. 10 CFR 50.54, “Conditions of Licenses,” U.S. Government Printing Office, Washington D.C., 2005.

6. 10 CFR 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events,” U.S. Government Printing Office, Washington D.C., 2005.

7. Appendix G to 10 CFR Part 50, “Fracture Toughness Requirements,” U.S. Government Printing Office, Washington D.C., 2005.

8. Appendix H to 10 CFR Part 50, “Reactor Vessel Material Surveillance Program Requirements,” U.S. Government Printing Office, Washington D.C., 2005.

9. 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants,” U.S. Government Printing Office, Washington D.C., 2005.

10. U.S. Nuclear Regulatory Commission, Supplement 4 to Generic Letter (GL) 88-20, “Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f),” June 28, 1991.

11. U.S. Nuclear Regulatory Commission, “Procedure and Submittals Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities,” NUREG-1407, June 1991.

12. U.S. Nuclear Regulatory Commission, “Staff Requirements Memorandum (SRM), SECY-4-0111, Recommended Staff Actions Regarding Agency Guidance in the Areas of Safety Conscious Work Environment and Safety Culture,” August 30, 2004.

13. Pouchet, A., “The European Benchmark Exercise on Human reliability Analysis, ”Proceedings of American nuclear society International Topical Meeting on probability, reliability, and safety Assessment (PSA ‘89), pp. 103-110, Pittsburgh, PA, April 2-7, 1989.

14. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.99, “Radiation Embrittlement of Reactor Vessel Materials,” Rev. 2, May 1988.

15. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," February 2004.

16. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," March 1997.

17. Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance."

18. Generic Safety Issue 185, "Control of Recriticality Following Small-break Locas in PWR."

Susquehanna Steam Electric Station

Advisory Committee on Reactor Safeguards

December 7, 2007

Non-Proprietary Version

Neutronic Methods Uncertainty Assessment

Ralph Grummer
Manager, Nuclear Technology

Overview

> Discuss Use of Pin and Bundle Gamma Scan Data in Generating Pin and Bundle Power Distribution Uncertainties

> Provide Sensitivity Studies of MCPR Safety Limit to Changes in Pin and Bundle Power Distribution Uncertainties

> Determine Increased Uncertainties to be used for CPPU MCPR Safety Limit Analysis

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Neutronic Methods
Use of Gamma Scan Data

> **Pin gamma scan data for ATRIUM fuel confirmed the pin power distribution uncertainty for the ATRIUM-10 design**
  - Quad Cities data at 7 axial levels → [ ]% uncertainty
    - 7x7 UO2, 8x8 UO2
  - KWU data at 4 axial levels → [ ]%
    - 9x9 UO2, 9x9 MOX, ATRIUM-10 UO2

> **Bundle power distribution uncertainty is determined from TIP measurements and a correlation coefficient based on Quad Cities bundle gamma scan data**
  - Correlation between adjacent bundle powers is reflected by a correlation coefficient

Neutronic Methods
Pin and Bundle Power Distribution Uncertainties

- [

-]

Neutronic Methods Conclusions

> In lieu of more gamma scan data
  - The pin power distribution uncertainty will be increased by [ ] to [ ] for future Susquehanna CPPU MCPR Safety Limit Analyses
  - The bundle power distribution uncertainty will be based on a [ ] reduced correlation coefficient of [ ] for future Susquehanna CPPU MCPR Safety Limit Analyses

Void Fraction Correlation Qualification

Doug Pruitt
Manager, Codes and Methods

Overview

- Licensing Void Fraction Correlation is appropriate for CPPU
  - Void correlation has been qualified against ATRIUM-10 void measurements
  - Void fraction uncertainty is already included in the MCPR Safety Limit Calculation through the bundle power distribution uncertainty
  - Higher quality reduces void fraction uncertainties
  - Sensitivity studies demonstrate that the MCPR Operating Limit is not sensitive to changes in void correlation

Void Fraction Correlation Uncertainties

- **A sensitivity analysis was performed to assess [**
- **]**

Void Fraction Correlation Uncertainties

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Void/Quality Sensitivity Results

- The use of an [ ] influences the core power distribution and void and scram reactivities
- [ ]
- Delta-CPR Impact
  - The most limiting transient ΔCPR increased by [ ], due to a slightly more top peaked power distribution and higher void reactivity
- MCPR Safety Limit Impact
  - The MCPR Safety Limit decreased by [ ], due to slightly higher radial peaking
- Net impact of void bias on the MCPR Operating Limit (MCPR Safety Limit + ΔCPR) is [ ]

Bypass Voids: Impact on Oscillation Power Range Monitor

Mr. Chester Lehmann

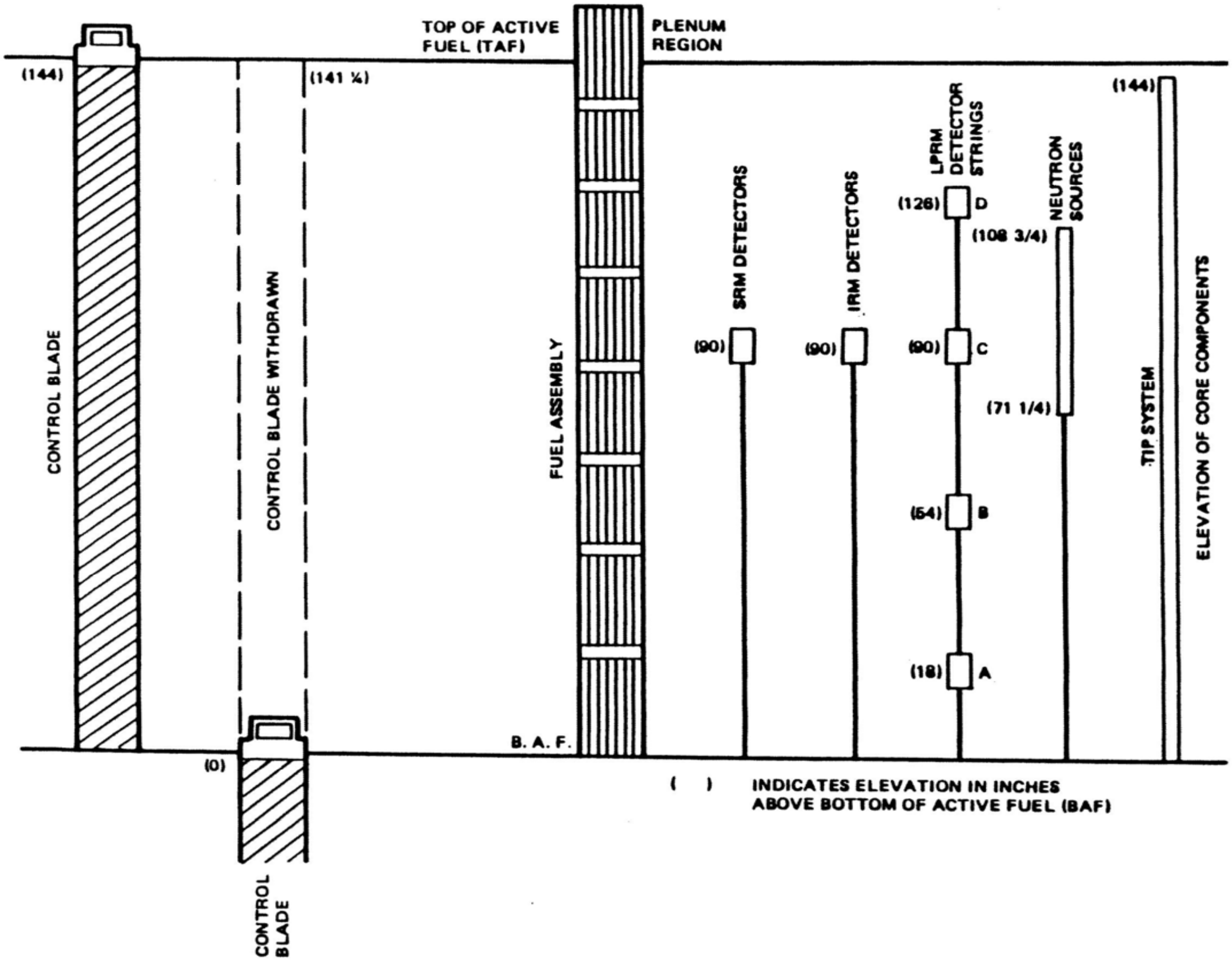
Supervisor – Plant Analysis

Bypass Voids: Impact on OPRM

OPRM Description

- **4 OPRM Trip Channels (GE Power Range Neutron Monitor System)**
- **Each OPRM Channel consists of 30 OPRM Cells (which cover the entire core)**
- **Uses 2 out of 4 trip logic**
- **The OPRM Cell signal is the sum of the 4 LPRM signals in that cell**
- **One OPRM Cell Trip causes its OPRM Channel to trip**
- **OPRM Cell Trip Occurs When Normalized (Peak/Average) Cell Signal $\geq$ Setpoint <u>and</u> Number of "Confirmation Counts" $\geq$ Setpoint**

Bypass Voids: Impact on OPRM
LPRM Axial Locations



Bypass Voids: Impact on OPRM
Phenomenon

- **Flow Decrease Increases Voiding in Upper Portion of Bypass Region**
- **Increased Voiding Decreases the Number of Thermal Neutrons at Upper LPRM Detector Locations (C & D Levels)**
- **Upper Level LPRM Signals will be Decreased**

Bypass Voids: Impact on OPRM

Setpoint Penalty Methodology

- **Analyze at Natural Circulation on Highest Rod Line - Highest Bypass Voiding**
- **Calculate Amount of Bypass Voiding at LPRM Locations**
  - **MICROBURN-B2**
  - **Multiple Bypass Channels / No cross flow credited**
  - **Analyze CPPU Core at BOC and EOC conditions**
  - **Use Maximum Calculated Bypass Voiding at C and D Level LPRMs in Lattice Physics Calculations**
- **Perform CASMO-4 Lattice Physics Calculations of Voiding Induced LPRM Signal Reduction**

Bypass Voids: Impact on OPRM
Setpoint Penalty Methodology

- **Assume only the Oscillatory Portion (i.e., peak) of the OPRM Signal is Affected**
- **Use Calculated Maximum C and D Level LPRM Signal Reduction**
- **Assume Most Limiting OPRM Cell Configuration**
  - **Two C Level and Two D Level LPRMs**
- **Calculate Average of C and D Level Signal Reductions to Represent Most Limiting OPRM Cell Signal Reduction**

Bypass Voids: Impact on OPRM
Setpoint Penalty Methodology

Example Setpoint Penalty Calculation :

- **OPRM Cell Signal Reduction = 5%**
- **OPRM calculated setpoint = 1.15**
- **OPRM setpoint penalty**

  (1.15 – 1.0) * 0.05 = 0.0075

Bypass Voids: Impact on OPRM
Method Conservatisms

- **Performing Analysis at Natural Circulation on Highest Rod Line**
  – **Susquehanna Technical Specifications require immediate manual reactor scram**
- **Applying LPRM Signal Reduction only to the Oscillatory Part of the LPRM Signal**
  – **OPRM trips on normalized amplitude (peak signal/average signal)**
- **Using Highest Calculated Bypass Voiding surrounding an LPRM**

Bypass Voids: Impact on OPRM

Method Conservatisms

- **Not crediting cross flow between bypass regions (maximizes bypass voiding)**
- **Penalty based on limiting OPRM cell configuration**
  - **95% of OPRM cells consist of less limiting LPRM configurations (less signal reduction)**
  - **These cells would also detect the oscillation**

Bypass Voids: Impact on OPRM
Conclusions

- **A Conservative OPRM Cell Signal Reduction will be Generated and Applied to the OPRM Setpoint for Susquehanna CPPU**

- **A Setpoint Penalty will be calculated from the OPRM Cell Signal Reduction and the Cycle Specific Calculated Setpoint**

Thermal Mechanical Methods

Michael Garrett
Manager, BWR Safety Analysis

Thermal Mechanical Methods

- Fuel rod Linear Heat Generation Rate (LHGR) limits are established using NRC-approved thermal mechanical methods
  - The Fuel Design Limit (FDL) LHGR ensures that fuel thermal mechanical design criteria (e.g., rod internal pressure) are not exceeded during steady state operation
  - The Protection Against Power Transients (PAPT) LHGR limit ensures fuel SAFDLs (<1% cladding strain and no fuel centerline melting) are not exceeded during Anticipated Operational Occurrences (AOOs)
- FDL and PAPT limits are unchanged from pre-CPPU operation
  - No failure of ATRIUM-10 fuel in Susquehanna units

LHGR Limits for ATRIUM-10 Fuel

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Pellet Clad Interaction (PCI)

> **REMACCX maneuvering restrictions (power ramp rate restrictions) provide protection from PCI failure during normal operation**

- **REMACCX restrictions implemented via the core monitoring system**
- **REMACCX restrictions unchanged with CPPU operation**
- **No PCI failure of ATRIUM-10 fuel in Susquehanna units**

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Pellet Clad Interaction (continued)

> Operating limit LHGR ensures SAFDLs are not exceeded during AOOs
- AOOs analyzed at CPPU conditions for Susquehanna
- Loss of feedwater heating (LFWH) was limiting event

> Significant fuel rod PCI failures are not expected during an AOO from CPPU conditions
- Fast core wide AOOs (e.g. turbine trip) - clad stress is low and PCI failures will not occur
- Slow core wide AOOs (e.g. LFWH) - operator action expected prior to PCI failures

Loss of Feedwater Heating (LFWH)

- Following LFWH, immediate operator action based on procedural requirements
  - Disturbance in feedwater system initiates alarms; operators trained to recognize and respond to LFWH
  - Loss of feedwater heater extraction steam occurred at Unit 2 in April 2007; operators initiated action to reduce power in < 3 minutes
- Without operator action, final (maximum) power is reached ~10 minutes after initiation of LFWH
- Operator action is expected to reduce core power in time to prevent PCI failure

LFWH Analyses with XEDOR

> **XEDOR is a tool for power maneuvering guidance**
  - **Reduced order stress model based on AREVA's fuel performance code RODEX4**
  - **Incorporated in MICROBURN-B2 with pin power reconstruction**
  - **Applied to every node of every rod in the core**
  - **Calculates clad hoop stresses with time variations of power and fast neutron flux**
  - **Under evaluation by EPRI as part of the Zero Failures by 2010 Initiative (with Anatech code FALCON)**

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LFWH Analyses with XEDOR

> LFWH analyzed for both CPPU cycle and current cycle designs to assess risk of PCI failure
  - Limiting cases follow startup or sequence exchanges assuming 100^{o}F feedwater temperature reduction and no operator action
  - [
  -
  -
  -

]

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PCI Conclusions

- Timely operator action for slow core wide AOOs will prevent sustained high stresses and provide PCI protection
- [

]

ATWS Instability

Mr. Chester Lehmann

Supervisor - Plant Analysis

(PPL)

Mr. Douglas Pruitt

Manager - Codes and Methods

(AREVA)

ATWS Instability
NEDO 32047-A: Purpose

- **Determine if Changes to ATWS Rule are Warranted for ATWS with Oscillations**
- **Demonstrate Core Coolability is Maintained**
- **Provide Justification for BWROG Proposed Mitigating Actions**
  - **Rapid SLCS Injection**
  - **Rapid Vessel Level Reduction to Decrease Core Inlet Subcooling**
  - **PPL Adopted these Proposed Actions in Emergency Operating Procedures**

ATWS Instability
NEDO 32047-A: Analysis

- **ATWS Instability Bounding Event – Turbine Trip with Flow Runback to Natural Circulation from MELLLA Line**
  - GE TRACG Analyses Performed from MELLLA point (currently licensed rod line for Susquehanna; not increased for CPPU)
- **Conservative Fuel Analysis**
  - Peak fuel pin used to model all pins in the bundle (accounts for differences in fuel types)
  - Cladding failures predicted at tops of leading pins

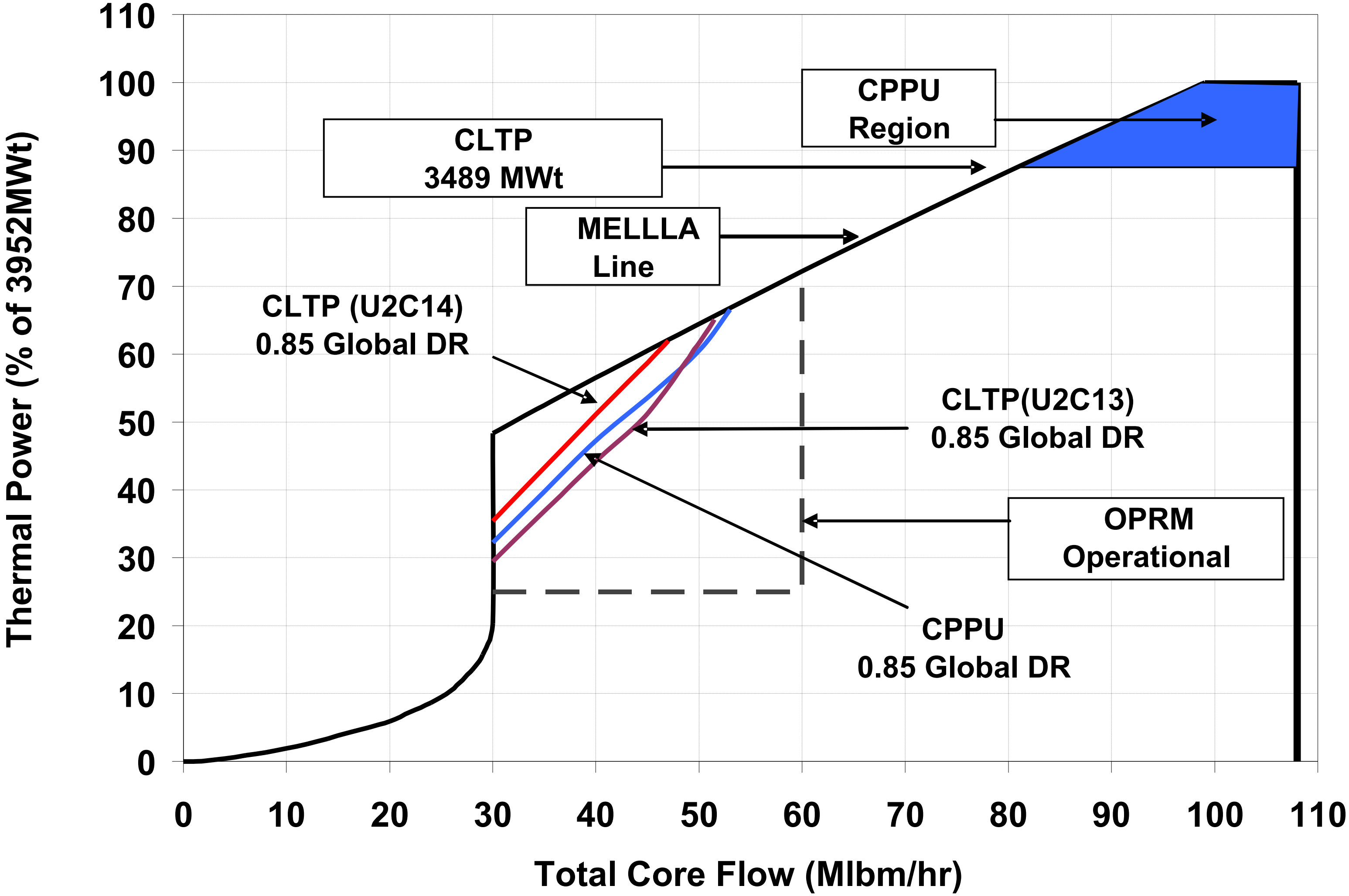
ATWS Instability
Topical Report SER Conclusions

- **NEDO-32047-A**
  - Core Coolability can be maintained
  - EPG revisions are sufficient for mitigating ATWS with oscillations

- **NEDC-33048-A (GE CPPU LTR)**
  - ATWS Instability Analysis Not Needed for CPPU (No Increase in Rod Line)

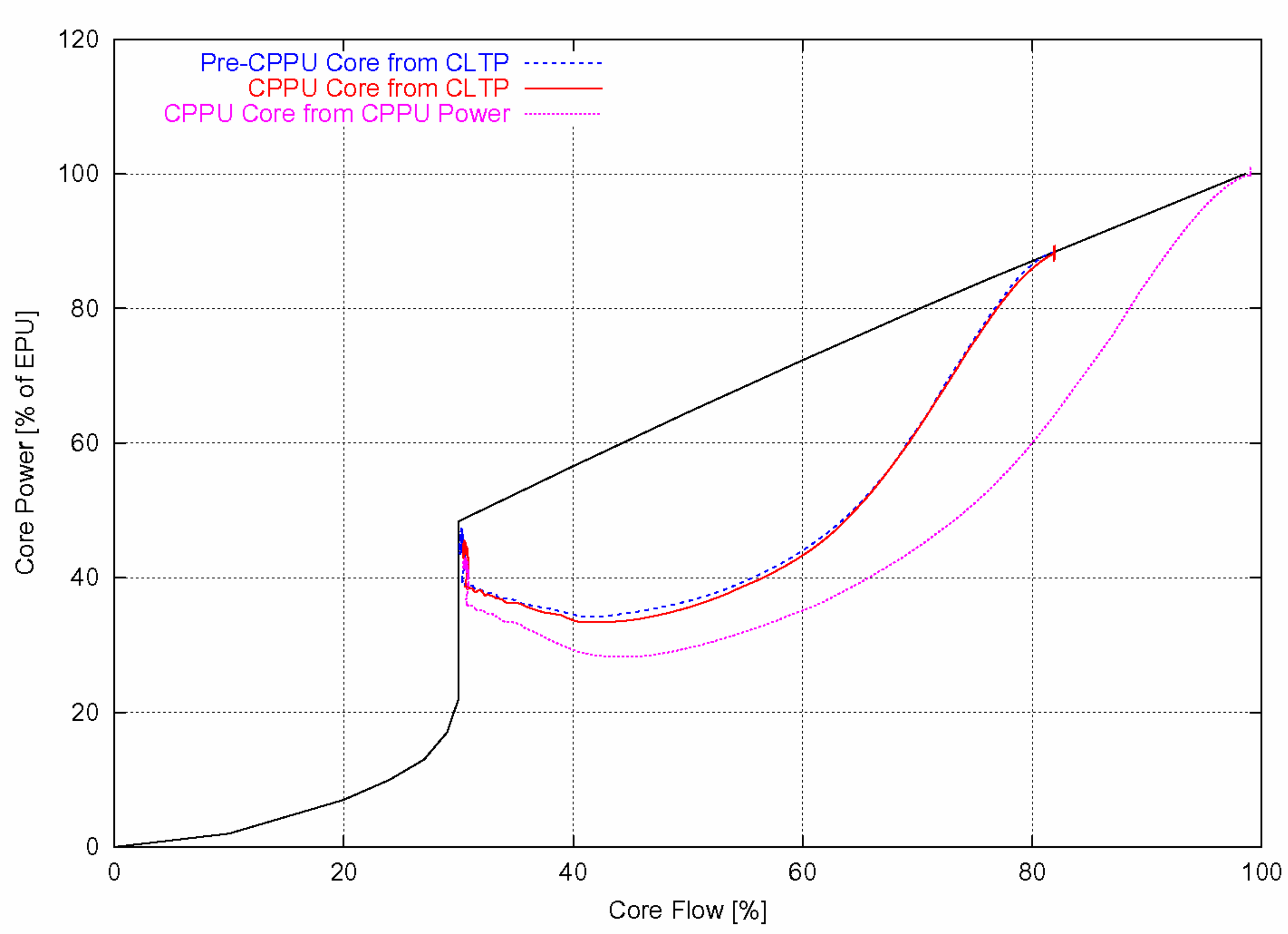
ATWS Instability

Comparison of CPPU & CLTP Global Decay ratios



ATWS Instability

Power/Flow Trajectories: CLTP & CPPU



ATWS Instability

CPPU / ATRIUM-10 Beneficial Changes

- **CPPU produces a flatter radial power distribution**

- **ATRIUM-10 fuel characteristics compared to 8x8 fuel analyzed in NEDO-32047-A**

ATWS Instability

Considerations from First Principles

- **Large unstable oscillation growth is terminated by two nonlinear feedback mechanisms**
  - Neutron kinetics nonlinear damping
  - [ ]

ATWS Instability
Neutron kinetic nonlinear damping

- **Global mode limit cycle amplitude depends on [ ]**

ATWS Instability

Conclusion: Neutron Kinetics Nonlinear Damping

- [ ]
  - All fuel designs introduced since NEDO 32047-A complied with NRC requirement for equal or better stability characteristics than previously approved fuel designs
  - Global decay ratios are calculated on a cycle to cycle basis
- **Variation in the maximum oscillation amplitude depends principally on the [ ]**

ATWS Instability
[ ]

- [ ]

ATWS Instability

[ ]

- [ ]

ATWS Instability

Conclusion

- [ ]

- NEDO-32047-A is applicable to Susquehanna CPPU

Susquehanna Power Uprate
Fuel System Design Review

Paul Clifford
Division of Safety Systems
Office of Nuclear Reactor Regulation

Staff Review

- Susquehanna's license amendment request was reviewed in accordance with established regulatory guidance.
  - NUREG-0800, Standard Review Plan Section 4.2
  - RS-001, Power Uprate Review Standard
- License amendment request satisfies all of Susquehanna's <u>current</u> licensing basis requirements.
  - Approved models and methods used to demonstrate compliance.
  - Approved fuel assembly design, ATRIUM-10, with proven fuel reliability.
  - Fuel design limits unchanged from pre-CPPU operation.

Current PCI/PCMI Regulatory Criteria

- Standard Review Plan Section 4.2 provides the following two criteria related to PCI/PCMI:
  - Transient-induced cladding strain below 1.0% (elastic + plastic).
  - No fuel centerline melting.
- Conservative aspects of Susquehanna’s application of the SRP criteria include:
  - Deterministic combination of manufacturing tolerances and modeling uncertainties.
  - Strain capability of ATRIUM-10 fuel rod design exceeds 1% uniform (elastic+plastic), especially at lower corrosion levels.
  - Unrealistic operator response and deterministic application of instrument uncertainties and system response times.

PCI/SCC Concern

- PCI/SCC is only a concern during an AOO power excursion for <u>actual</u> fuel rod failures which may occur below <u>predicted</u> failure based on conservative estimates of:
  - MCPR thermal design limits
  - 1% (elastic+plastic) strain
  - Fuel centerline melt
- Explicit PCI/SCC modeling unlikely to significantly increase the number of fuel failures beyond current predictions (to the extent where calculated offsite doses would be substantially increased).

Barrier Cladding Fuel Design

Barrier ≠ More Safety Margin

- While barrier fuel designs have been shown to be less susceptible to PCI/SCC, this design feature is more susceptible to secondary failure (e.g., long axial splits).
- Plants with barrier fuel will tend to push fuel harder with more aggressive power maneuvering.

Variations in Barrier Fuel Design

Barrier ≠ No PCI/SCC Concerns

- Level of PCI/SCC resistance varies with design.
  - Concerns with secondary failure have prompted vendors to develop low-alloy cladding liners.
  - Liner thickness varies with fuel rod design.
- Staff unable to address PCI/SCC susceptibility without a robust methodology capable of differentiating PCI/SCC resistance for various fuel rod designs.

PCI/SCC Regulatory Criteria

- Establishing <u>generic</u> PCI/SCC regulatory criteria will be a long, complex process.
  - Develop a detailed mechanistic fuel rod design model capable of calculating local stress concentrations and tuned to capture the chemical attack of PCI/SCC.
  - Develop a PCI/SCC Specified Acceptable Fuel Design Limit (SAFDL).
  - Develop regulatory guidance and testing requirements.
  - Elicit public and industry comment.
  - Revise NUREG-0800, Standard Review Plan Section 4.2.
  - Complete backfit determination pursuit with 10 CFR 50.109.
    - <u>If</u> the proposed change in regulatory staff position qualifies as either an exception (e.g. compliance, adequate protection) or cost-justified substantial increase in safety under the provisions of 10 CFR 50.109, <u>then</u> develop an implementation schedule.

December 7th, 2007

MLC Summary of ESBWR Subcommittee Meeting (11-15-2007)

The subcommittee meeting reviewed Chapters 9, 10, 13, and 16 of the DCD and the draft SER. The following points summarize comments from the subcommittee and consultants.

- The staff has done a competent job with their reviews of these chapters of the DCD.
- The ESBWR Regulatory Treatment of Non-Safety Systems seems to comply with the regulations, but we need to better understand how such systems are treated in practical terms and RTNSS treatment in the PRA. T. Kress has also suggested that the ACRS review how Importance Measures are used in determining SSC's. For example, if the diesel generators are in this RTNSS category, how are they to be started and loaded before and/or after the 72 hour post-accident period.
- HVAC questions still remain during the 72 hours post-accident period. Under the condition that no AC powered systems are available, there is a need to address the issues particularly with respect to the effects on equipment on adverse temperatures and control room habitability due to the air quality of intake and exhaust air. GEH has noted our concern and committed to more detailed analyses relative to both issues.
- The Standby-Liquid Control System (SLCS) is designed to operate with nitrogen cover gas pressure to insure injection under all conditions. GEH indicated that the potential exists for the nitrogen gas injection to continue with gas injection into the RCS. The subcommittee members would like more information as to the effects of nitrogen on long-term cooling.
- The ESBWR design of the spent fuel transfer system prompted some questions. GEH indicated that Chapter 15 would include various off-normal spent fuel cooling events; e.g., assessment of the case of an isolated inclined transfer tube containing two fuel bundles to assure that the system does not overheat and fail fuel. Other spent fuel pool issues can be reviewed at that time.
- Passive safety systems (IC, PCCS, GDCS, VB, DIC) surveillance testing frequencies need to be revisited when the final technical specifications document is provided to us in DCD-Rev#5. This issue involves the testing frequencies and demonstration of operability for passive systems. It may need to consider risk as well as engineering judgment assessments.
- Non-Safety systems may have indirect effects on safety; e.g., instrument air quality. The impacts of moisture and contamination in IA systems create some of the most challenging situations for plant operators. Unusual situations can lead operators to take what appear to be surprising actions. This may become a safety issue. We need to understand how staff has considered the likelihood and potential impacts of moisture and contamination in these systems.
- Materials issues related to the ESBWR water chemistry will be a continuing topic of discussion; e.g., Hydrogen water chemistry as an option versus a design requirement.

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ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
548th	11	11.1

Title **RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS**

Authors

SAM DURAISWAMY

List of Documents Attached

See attached list

11

Instructions to Preparer
1. Paginate Attachments
2. Punch holes
3. Place Copy in file box

From Staff Person
SAM DURAISWAMY

SUBJECT	ANALYSIS	EDO LTR.	ACRS LTR.
Development of a Technology-Neutral Regulatory Framework (MC/DCF)	11/15/07 (pp. 1-2)	11/01/07 (pp. 3–4)	09/26/07 (pp. 5-12)
NRC's Staff's Assessment of the Industry Study Related to Dissimilar Metal Weld Issues in Pressurizer Nozzles (WJS/CGH)	11/29/07 (p. 13)	11/23/07 (p. 14-15)	10/19/07 (pp. 16-19)
Digital Instrumentation and Control Systems Project Plan and Interim Staff Guidance (GEA/GSS)	12/03/07 (p. 20)	11/21/07 (pp. 21–22)	10/16/07 (pp. 23-25)

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

November 15, 2007

MEMORANDUM TO: Michael Corradini, Chair
Future Plant Designs Subcommittee

FROM: David C. Fischer, Senior Staff Engineer /RA/

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER ON THE DEVELOPMENT OF A TECHNOLOGY-NEUTRAL REGULATORY FRAMEWORK

Attached is a copy of the EDO's November 6, 2007, letter of response to the ACRS September 26, 2007, report on the development of a technology-neutral regulatory framework. In its September 26, 2007, report to the Chairman (also attached), the Committee made five conclusions or recommendations:

1. We concur with the staff that the safety objective of the framework should be to ensure that advanced reactors, as a minimum, provide at least the same degree of protection of the public and the environment that is required for current-generation light water reactors (LWRs), and that advanced reactor designs comply with the Commission's safety goal quantitative health objectives (QHOs).

2. We concur with the staff that a set of licensing-basis events (LBEs) is needed as part of the licensing basis to structure the interactions between the staff and the applicant and to focus the conduct of mechanistic analyses. Identifying the LBEs by using the probabilistic risk assessment (PRA) reduces the risk that licensing-basis requirements will divert attention from events of real safety significance.

3. The use of a frequency-consequence (F-C) curve is an appropriate way to establish a range of regulatory requirements to limit radiation exposure to the public. However, a sequence-specific F-C curve, such as that developed in NUREG-1860 may not be a sufficient licensing criterion. A complementary cumulative distribution function (CCDF) F-C curve ("risk curve") that sums the contributions to risk from the entire spectrum of accident sequences establishes limits on risk better than the LBE F-C curve.

4. We are concerned that extension of the F-C curves to very low dose levels may unduly increase requirements for the scope and level of detail in the PRA performed to demonstrate compliance with the F-C curve. It may also detract attention from accidents which could have a more significant impact on public health and safety.

5. The framework should recognize accident prevention as a fundamental regulatory goal and should specify a quantitative limit on the frequency of an accident. In technology-neutral terms, an accident can be defined as the release of radionuclides within the plant significantly in excess of normal operating limits.

STAFF RESPONSE (IN PART):

The staff stated that it agreed with many of the Committee's conclusions and recommendations and indicated that it plans to address them within NUREG-1860. Specifically, the staff plans on publishing NUREG-1860 in December 2007 (consistent with a September 10, 2007 staff requirements memorandum on SECY-07-0101). Appendix C to NUREG-1860 will provide a list and discussion of the programmatic, policy, and technical issues (including those raised by the ACRS) that need to be addressed when and if the Agency implements the approach described in NUREG-1860. Appendix L to NUREG-1860 will summarize each issue raised by the ACRS, as well as the comments submitted by various stakeholders, and provide a staff response to each. The staff said that it would treat the additional comment appended to the Committee's September 26, 2007 letter as stakeholder comments in Appendix L to NUREG-1860.

ANALYSIS:

In general, the EDO response is satisfactory in that the staff agreed with many of the Committee's conclusions and recommendations. However, the staff's responses to the specific issues raised by the ACRS (including the additional comments provided byy individual members) will be provided in Appendix L to NUREG-1860, scheduled to be published in December 2007. The Committee should consider the staff's specific responses to the Committee's conclusions and recommendations after NUREG-1860 has been published.

Attachments: As stated

cc: ACRS Members
F. Gillespie
S. Duraiswamy
C. Santos

November 1, 2007

Dr. William J. Shack, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: DEVELOPMENT OF A TECHNOLOGY-NEUTRAL REGULATORY FRAMEWORK

Dear Dr. Shack:

Thank you for your letter, dated September 26, 2007, in which you conveyed the views of the Advisory Committee on Reactor Safeguards (ACRS) regarding the development of a technology-neutral regulatory framework for the U.S. Nuclear Regulatory Commission (NRC). In so doing, you addressed the "issues, critical to the development of a framework, that were still being debated within the ACRS," as noted in the Committee's previous letter on this subject, dated May 16, 2007. We agree with many of the conclusions and recommendations in your letter, and plan to address all of them within NUREG 1860.

Consistent with previous staff correspondence on this topic to the ACRS, the staff maintains that (1) the framework's primary objective of demonstrating the feasibility of a possible risk-informed and performance-based approach that would serve as the technical basis for licensing a reactor employing any technology has been achieved, (2) publication of NUREG-1860 does not represent closure of the issues within the framework, but it documents a significant piece of research that may be used in the future, and (3) the issues should be addressed as part of any potential future development of regulatory guidance or rulemaking that would be needed to implement an approach akin to the framework. Moreover, in SECY-07-0101, "Staff Recommendations Regarding a Risk-Informed and Performance-Based Revision to 10 CFR Part 50," dated June 14, 2007, the staff committed to publish the framework after considering the stakeholders' comments and any additional comments from the ACRS. The Commission approved that commitment in its staff requirements memorandum, dated September 10, 2007. In fulfilling that commitment, the staff is scheduled to publish NUREG-1860 in December 2007.

In preparing to publish NUREG-1860, the NRC staff has reviewed the ACRS comments, as well as those received as a result of the public review. In particular, the staff has added two appendices (Appendices C and L). Appendix C provides a list and discussion of the programmatic, policy, and technical issues that need to be addressed when and if the agency implements the approach described in NUREG-1860. The issues raised by the ACRS are addressed in Appendix C. In addition, Appendix L summarizes each issue raised by the ACRS, as well as the comments submitted by various stakeholders, and provides a staff response to each.

The staff recognizes that the additional comments attached to your letter, dated September 26, 2007, represent the views of individual members, rather than official Committee positions. Nonetheless, the staff considers these to be additional stakeholder comments, which warrant inclusion and staff responses in Appendix L to NUREG-1860.

The staff appreciates the Committee's continued interest and collaboration in developing the agency's technology-neutral regulatory f ramework. We look forward to ongoing interaction on this important topic.

Sincerely,

/RA/

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Klein
Commissioner Jaczko
Commissioner Lyons
SECY

September 26, 2007

The Honorable Dale E. Klein
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: DEVELOPMENT OF A TECHNOLOGY-NEUTRAL REGULATORY FRAMEWORK

Dear Chairman Klein:

During the 545th meeting of the Advisory Committee on Reactor Safeguards (ACRS), September 6-8, 2007, we completed our review of draft NUREG-1860, "Framework for Development of a Risk-Informed, Performance-Based Alternative to 10 CFR Part 50." We met with the NRC staff and discussed this matter during our 540th meeting, March 8-9, 2007. In addition, our Subcommittee on Future Plant Designs reviewed this document on March 7, 2007. We had the benefit of the documents referenced. In our May 16, 2007, report we stated that there were issues, critical to the development of a framework, that were still being debated within the ACRS. This report provides our view on some issues important to the development of a conceptual framework.

CONCLUSIONS AND RECOMMENDATIONS

1. We concur with the staff that the safety objective of the framework should be to ensure that advanced reactors, as a minimum, provide at least the same degree of protection of the public and the environment that is required for current-generation light water reactors (LWRs), and that advanced reactor designs comply with the Commission's safety goal quantitative health objectives (QHOs).

2. We concur with the staff that a set of licensing-basis events (LBEs) is needed as part of the licensing basis to structure the interactions between the staff and the applicant and to focus the conduct of mechanistic analyses. Identifying the LBEs by using the probabilistic risk assessment (PRA) reduces the risk that licensing-basis requirements will divert attention from events of real safety significance.

3. The use of a frequency-consequence (F-C) curve is an appropriate way to establish a range of regulatory requirements to limit radiation exposure to the public. However, a sequence-specific F-C curve, such as that developed in NUREG-1860, may not be a sufficient licensing criterion. A complementary cumulative distribution function (CCDF) F-C curve ("risk curve") that sums the contributions to risk from the entire spectrum of accident sequences establishes limits on risk better than the LBE F-C curve.

4. We are concerned that extension of the F-C curves to very low dose levels may unduly increase requirements for the scope and level of detail in the PRA performed to demonstrate compliance with the F-C curve. It may also detract attention from accidents which could have a more significant impact on public health and safety.

5. The framework should recognize accident prevention as a fundamental regulatory goal and should specify a quantitative limit on the frequency of an accident. In technology-neutral terms, an accident can be defined as the release of radionuclides within the plant significantly in excess of normal operating limits.

DISCUSSION

The framework proposed by the staff is intended to provide the conceptual basis for the development of a technology-neutral regulatory system. This system is intended to achieve a level of safety considering all modes of operation that is consistent with the expectations stated in the Commission's policy statement entitled, "Regulation of Advanced Nuclear Power Plants." The policy statement states that the Commission expects that advanced reactors will provide enhanced margins of safety and that advanced reactor designs will comply with the Commission's Safety Goal Policy Statement (Safety Goals). The Commission also expects that advanced reactors should provide, as a minimum, at least the same degree of protection of the public and the environment that is required for current-generation LWRs.

Reactors must be designed and operated in a manner that ensures "adequate protection" to the public. Lower levels of risk are desirable and can be considered in terms of cost benefit. The Safety Goals define a level of risk below which no additional risk reductions need to be considered. The objective of the framework is to help develop requirements for future reactors that will ensure there is only a small chance that the risk will exceed that defined by the Safety Goals. Since all risks may not be recognized or fully evaluated in the design certification and licensing process, the regulatory system should also address situations or conditions at operating reactors in which the risks could exceed the Safety Goals.

For application to a site, the framework states that the integrated risk from all new reactors at the site must meet the QHOs. We concur with the staff that this is an appropriate level of expectation for safety in new reactors and an appropriate treatment of risk at a site.

Compliance with the QHOs depends, in part, upon site characteristics. For design certification, it is convenient to have criteria that minimize dependence on site characteristics. The approach taken in the framework to define criteria that can be used by the designer is to provide limits on dose at certain distances from the plant boundary. The staff assumes that these dose limits are sufficiently conservative to provide a high degree of confidence that a design which meets these limits will meet the QHOs and other regulatory limits at most sites. Although such dose limits will in fact be technology dependent, we concur with this approach for developing guidance for designers.

In addition to the QHOs, the current regulations have other limits on the release of radioactive material and on radiation doses to the public during normal operation and hypothetical accidents. In the staff's current approach to a framework, these requirements have been used to develop an F-C curve where the frequency is frequency of an individual PRA sequence and the consequence is the dose associated with that sequence, calculated at prescribed distances that vary with the frequency. Such an approach can also be viewed as a defense-in-depth measure that sets high-level requirements for reliability and inspection. Limits on the frequencies of smaller releases on this F-C curve control the allowable degradation of "barriers" that prevent the inadvertent release of radioactive material to the environment.

In the development of such an F-C curve, the goal is to provide consistency with 10 CFR Part 50, 10 CFR Part 20, Environmental Protection Agency requirements, and the QHOs.

NUREG-1860 presents a candidate F-C curve. Some judgment was required to assign frequencies to the various dose ranges. An alternative candidate F-C curve is discussed in EPRI TR-1013582, "Technical Elements of a Risk-Informed, Technology-Neutral Design and Licensing Framework for New Nuclear Plants." It is premature to determine whether either of these curves is the most appropriate expression of the current requirements.

In addition to the requirement that each PRA sequence meet the F-C curve limits, the PRA results must demonstrate that the total integrated risk satisfies the QHOs. There are also some additional cumulative dose limits for sequences with frequencies greater than 10^{-3}/year. However, the current framework as described in NUREG-1860 does not contain a complete definition of risk in terms of a CCDF F-C curve, which describes the frequency of exceeding a given dose summed over all PRA sequences ("risk curve"). A CCDF F-C curve establishes better limits on risk. A candidate CCDF F-C curve that attempts to provide consistency with the QHOs, other regulatory requirements, and reflects experience with current operating reactors is discussed in EPRI TR-1013582. However, development of an appropriate CCDF F-C curve will require additional effort.

The F-C curve, the QHOs, and a CCDF F-C curve prescribe limits on the release of radioactive material from the plant. The framework currently does not have a quantitative requirement for accident prevention corresponding to the core damage frequency (CDF) value currently used for LWRs. The discussion of surrogate measures such as CDF in NUREG-1860, notes their usefulness in balancing accident prevention and mitigation, but continues to focus on their usefulness as simplified representations of the QHOs. Accident prevention should be considered a fundamental goal of regulation. In technology-neutral terms, an accident can be defined as the release of radionuclides within the plant significantly in excess of normal operating limits. Although it may not be possible to relate a limit on the frequency of such accidents to the QHOs, it would be a reasonable extension of current regulatory practice to establish a quantitative limit on the frequency of such accidents. Such a requirement should be included in the framework.

The framework described in NUREG-1860 envisions a far more central role for PRA in the regulatory system and design process than in the current licensing process. Even though there will be significant issues of modeling uncertainty and completeness associated with new designs, PRA methods provide the best tool to identify vulnerabilities that challenge design assumptions. For LWRs, the scope of PRAs has been focused on estimates of the frequency of beyond-design-basis accidents. For future reactor designs, the PRA approach described in the framework provides estimates of the frequencies of sequences with consequences that range from small releases to severe accidents with large releases of radioactive materials. We are concerned that extension of the F-C curve to very low dose levels may unduly increase requirements for the scope and level of detail in the PRA performed to demonstrate compliance with the F-C curve. It may also detract attention from accidents which could have a more significant impact on public health and safety.

In NUREG-1860, it is proposed that the PRA be used to identify a set of LBEs that encompass a whole spectrum of off-normal events (including frequent, infrequent, and rare initiating events and event sequences) and include a spectrum of radioactive material releases from minor to major. An additional defense-in-depth LBE is imposed to ensure that a postulated release of radioactive material from the fuel and the reactor will not exceed the 10 CFR Part 100 limits.

The LBEs play a role akin to design-basis accidents (DBAs) in 10 CFR Part 50. However, there are important differences. LBEs are based on sequence frequencies, not initiating event frequencies. They are not artificially limited to considerations of single failure, and thus allow considerations of vulnerabilities associated with a relatively frequent initiating event cascading through a series of failures to an event with significant consequences.

The LBEs will be proposed by the applicant and reviewed and approved by the staff. The LBEs are chosen from the PRA by grouping similar accident sequences into an event class. An LBE is selected as the event from the class with the bounding consequence. It is assigned a frequency equal to the most frequent event in the class. There is no unique definition of a "sequence" in a PRA. This is recognized in NUREG–1860 and some guidance is provided to ensure that the sequences to be compared to the F-C curve and used to identify the LBEs are defined in a meaningful way. However, additional guidance would have to be provided in order to implement the framework. Since the PRA is envisioned as a living PRA, LBEs can be changed during the life of the plant based on operational experience or other additional knowledge.

Although the PRA provides the best characterization of the risk of the plant, we support the concept of the LBEs. The LBEs provide a useful check on the quality of the mechanistic analyses used in the PRA, provide additional margin, establish a well-defined commitment for the licensee, and set limits on regulatory attention and control. Since LBEs are based on the PRA, there is less chance that they will divert attention from events of real safety significance, as can be the case with the current DBA approach. The li censing basis may also include other metrics from the PRA such as importance measures.

The use of LBEs as the fundamental licensing basis reduces the dependence on the "bottom line" numbers of the PRA. The PRA is used primarily to identify the most significant sequences and not to provide "risk numbers" as part of the licensing basis.

Additional comments by ACRS members Thomas S. Kress, Dana A. Powers, and Graham B. Wallis are provided below.

Sincerely,

/RA/

William J. Shack
Chairman

Additional Comments by ACRS Member Thomas S. Kress

The Committee' report does not embrace two long-standing ACRS positions: (1) the criteria for design safety of new reactors should be consistent with a core damage frequency (CDF) of $10^{ÿ5}$/year and a large release frequency (LRF) of 10^{-6}/year; and (2) design and siting should be separated as much as practical in the regulatory process.

The report correctly considers that a frequency-consequence complementary cumulative distribution function (CCDF) design acceptance criterion would properly summate the risk. It should be a mandatory part of the framework. It is entirely possible to construct such a CCDF acceptance criterion that would make it consistent with any chosen values of CDF and LRF. The ACRS report should call for the inclusion of such a criterion for CDF of 10^{-5}/year and LRF of 10^{-6}/year for the following reasons:

- These values would ensure that several such plants placed on a greenfield site would meet the current QHOs with margin.
- The recommended values for CDF and LRF are consistent with current U.S. and international positions.
- Having a CCDF criterion that embodies both a CDF and a LRF provides for a balance between prevention and mitigation.
- Such a criterion would provide a consistent way to compare the safety status of new plants with that of the current plants.
- It would also provide a way to relate new regulatory requirements to the existing ones that utilize CDF and large early release frequency (LERF) (for example, the Backfit Rule).
- Societal risk is not addressed by either the framework or the ACRS report. The above recommended value for LRF is highly likely to be a good surrogate for acceptable societal risk.

In accepting the framework's LBE frequency-consequence figure-of-merit design curve which uses dose as the consequence measure, the Committee has compromised the principle of separation of design and siting. This curve requires the designer to utilize a surrogate characteristic site along with a level-3 PRA to determine the dose targets. This is an unnecessary burden to place on the design at the stage for which the actual site is not known especially since the use of equivalent curies released from containment will better serve the purpose and be simultaneously compatible with the CCDF LRF criterion that necessarily uses curies released as the consequence measure. This compatibility greatly enhances the design process that is envisioned to require iteration on the figure-of-merit acceptance curve to meet the real risk criteria embodied in the CCDF curve.

Additional Comments by ACRS Member Dana A. Powers

A well crafted, technology-neutral regulatory framework could facilitate the development of higher efficiency nuclear power technologies and innovative application of nuclear technologies to address the economic and security issues confronting our Nation. The overly complicated regulatory framework developed by the staff is not a useful first step in the needed evolution of the current regulatory system to become technology neutral.

The proposed framework is not well founded. My colleague, Professor Wallis, has gone to great lengths to point to questionable elements of the framework foundation. I note that staff did not take advantage of the current General Design Criteria (Appendix A, 10 CFR Part 50) many of which were defined before there was broad acceptance of LWR technology and are, consequently, technology neutral. Regulatory experience of the last 35 years might well suggest a few additional criteria or some amendment of current criteria. Together, these criteria would have provided a sound foundation for a technology-neutral regulatory framework useful to nuclear power plant designers, builders, and regulators.

Instead, staff has chosen to base its proposed regulatory framework on risk assessment. The proposed framework demands PRAs well beyond the current state of the art. It is plausible that future risk assessments, unlike those done today, could address all accident initiators under all modes of operation in some integrated way. It might be possible in some future time to do the comprehensive uncertainty analysis addressing both parameter uncertainty and model uncertainty to obtain mean values for comparison to the quantitative health objectives that the staff envisages in its proposed framework. But, the staff has gone well beyond even this plausible future to expand the scope of PRAs mandated for regulation to extremes not even imaginable today. The introduction of the frequency-consequence (F-C) curves extending down to very low dose levels will necessitate these vast expansions in the scope of risk assessments. Risk assessments will need to include events associated with drains in the plant chemistry laboratory to meet staff expectations communicated through the F-C curves. This expansion in scope will impose burdens on both licensees and regulators heretofore never imagined. It will detract from a focus on safety issues that really do pose significant threats to the public health and safety.

The complexity of the proposed framework may have arisen as the authors attempted to satisfy many skeptics. Preservation of the design basis accident (DBA) concept under the guise of "licensing basis events" (LBEs) is remarkable. The deficiencies of DBAs as a feature of the regulatory system have become apparent to us all since the Three-Mile Island accident in 1979. Staff proposes that all pretense of realistic regulation be abandoned for the LBEs. Like the current DBAs, these LBEs will be analyzed using very conservative methods. Staff hopes that such LBEs will be defined anew for each reactor technology and each change to each technology, and discounts the likelihood that LBEs will ossify much as have DBAs into a legalistic analysis framework disconnected from physical reality. The technology-neutral regulatory framework proposed by the staff is destined to descend into a concentration on a few stylized accidents driven as a result of the focus on very low probabilities and the consequent neglect of more probable events that actually pose risks to the public. Preservation of the DBA concept is all the more remarkable in the proposed framework since it appears to turn its back on the breadth of attention sought in the drive over the last few years to develop a risk-informed regulatory system.

All of the motivations for the preservation of the current regulatory approach via the LBEs are not entirely certain. It may be that the staff sought a mechanism to develop sufficient understanding to optimize application of inspection and monitoring resources to new types of nuclear power plants. It is remarkable that the authors elected not to use importance metrics derived from the risk assessments such as "risk achievement worth" and "risk reduction worth." The importance metrics are among the most powerful results that can be derived from risk assessments even when these risk assessments do not meet the extreme standards of scope envisaged in the proposed framework. The authors mandate construction of risk assessments of unbelievable scope and depth but make no use of the results beyond a rather effete comparison to "bottom line" risk results widely considered to be the most uncertain aspects of risk assessments. The importance metrics derived from the risk assessments provide a comprehensive examination of systems, structures, and components important to plant safety and an identification of those critical aspects of plants that merit close inspection and monitoring. The metrics do this without singling out particular types of accidents. The metrics are most reliable when they are derived from realistic analyses. The importance metrics of risk assessments have already been demonstrated to be a far more useful mechanism for the optimal allocation of safety resources by both the licensee and the regulator. Use of both risk reduction worth and risk achievement worth could be developed into a rational mechanism for introduction of defense-in-depth into safety regulation. Yet, importance metrics make no appearance in the proposed regulatory framework.

Some have suggested that the proposed framework be tested on a new reactor technology such as a gas-cooled nuclear power plant. I think this is not a good idea. Aside from the deficiencies of the proposed framework identified above and elsewhere, there is not a good phenomenological basis for assessing gas-cooled reactor safety. Even such a routine analysis as assessing the radionuclide release associated with expected depressurization events at gas-cooled reactors cannot be confidently done today as has been demonstrated in a Phenomena Identification and Ranking exercise recently undertaken by the NRC staff. This will assuredly handicap any application of a proposed regulatory framework focused as this one is on F-C curves and bottom-line risk results.

Additional Comments by ACRS Member Graham B. Wallis

My colleagues have made considerable progress on this important issue. However, there are still many features of some of the recommendations by them and by the staff for which the justification and implications have not been adequately evaluated.

The framework proposed by the staff in draft NUREG-1860 requires substantial revision to demonstrate that it responds to the needs of the Agency and that appropriate choices have been made. I have provided reasons and suggestions in a set of detailed comments which has been shared with the staff (Reference 6).

At an early point in the revised document, the following questions should be answered by providing convincing analysis and rationale:

- Are the QHOs the only top level regulatory criteria? Should they be supplemented by additional metrics such as those describing societal and environmental risk? (There are disparate assessments of the health effects of the Chernobyl event, but little doubt of the consequences for the livelihood of farmers and herders in the northern U.K. and Scandinavia).

- The QHOs are probabilistic criteria. They depend on the PRA results. With what confidence should they be met? Is the level of technical representation of accident scenarios in the PRA adequate to form the sole major basis for regulation?

- Are the QHOs realistic requirements? They have not been used in previous regulation. The example in Appendix E appears to show that a current LWR would fail to meet them. The latent cancer QHO is equivalent to 4 mrem/Ry; this can all be consumed by a few frequent events of insignificant consequence which would not now be modeled in the PRA. Multiple-reactor sites might fail to meet this requirement under present normal operating conditions.

- Do the QHOs need to be supplemented by requirements for additional " deterministic" analyses of the type represented by 10 CFR 50.46? How will these differ from the mechanistic analyses in the PRA? To what degree should this requirement be specified? How is it decided how much is enough?

- Should defense-in-depth requirements, such as for a containment/confinement, multiple barriers and a coolable geometry, be imposed? If so, shouldn't they be described as part of the basis of the framework and not hidden in a few lines of text?

- Should anything be specified about the allowable frequency-consequence spectrum of accidents? What functions will this serve? What is the best choice of format?

- What will be used to describe the overall safety status of the plant? CDF and LERF now play this role. Can the QHO metrics be used instead?

- Will the PRA technical analysis incorporate tools such as the thermal-hydraulic codes used in the current analysis of DBAs, as appears to be indicated in Appendix F? Does this require major technical and computational research and development?

- Besides requiring that a licensee meet the QHOs by summing up the latent cancer and fatality metrics from the PRA outputs, and perhaps also meet some defense-in-depth criteria, are any secondary requirements desirable? Some, such as for PRA quality, may be justified in the interest of effectiveness and efficiency in meeting the primary ones. Do some other requirements in Part 50 and other existing regulations influence the design of the framework? Are these compatible with the QHOs?

- What regulatory functions does the framework satisfy at the design certification stage, the combined license (COL) stage, and later stages of regulation such as approval of changes to the plant? Are there needs for inspection and enforcement which require that additional or modified features be specified in some parts of the framework? Will there be additional site- and technology-specific requirements?

- What will the "licensing basis" of a plant look like under the new framework? Will the Safety Analysis Report (SAR) be significantly better focused, more economical for the Agency and the licensee, and clearer to the informed public than under the present system?

References:

1. Memorandum dated April 3, 2007, from Farouk Eltawila, Director, Division of Risk Assessment and Special Projects, RES, to Frank P. Gillespie, Executive Director, ACRS, Subject: Transmittal of Proposed "Technology Neutral Framework" for Advisory Committee on Reactor Safeguards Review.
2. Report dated April 20, 2007, from William J. Shack, Chairman, ACRS to Dale E. Klein, Chairman, NRC, Subject: Technology-Neutral Framework for Future Plant Licensing.
3. Report dated May 16, 2007, from William J. Shack, Chairman, ACRS to Dale E. Klein, Chairman, NRC, Subject: Draft Commission Paper on Staff Plan Regarding a Risk-Informed and Performance-Based Revision to 10 CFR Part 50.
4. U.S. Nuclear Regulatory Commission Policy Statement on "Regulation of Advanced Nuclear Power Plants," published July 12, 1994, 59 FR 35461.
5. EPRI-1013582, "Technical Elements of a Risk-Informed, Technology-Neutral Design and Licensing Framework for New Nuclear Plants," December 2006.
6. Letter dated July 24, 2007, from Graham B. Wallis, ACRS to Luis A. Reyes, Executive Director for Operations, NRC, Subject: Comments on Draft NUREG-1860, "Framework for Development of a Risk-Informed, Performance-Based Alternative to 10 CFR Part 50."

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

November 29, 2007

MEMORANDUM TO: William J. Shack, Acting Chair
Materials, Metallurgy, and Reactor Fuels Subcommittee

FROM: Charles G. Hammer, Senior Staff Engineer **/RA/**

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER ON NRC STAFF'S SAFETY ASSESSMENT OF THE INDUSTRY STUDY RELATED TO DISSIMILAR METAL WELD ISSUES IN PRESSURIZER NOZZLES

Attached is a copy of the EDO's November 23, 2007 letter of response to the ACRS' October 19, 2007 letter on the NRC staff's assessment of the industry's advanced finite element analysis of primary water stress corrosion cracking (PWSCC) in pressurizer nozzles. A copy of the Committee's letter is also attached.

Committee Letter

In its March 22, 2007 letter the ACRS concluded that studies undertaken by the industry and staff have been timely and significantly improved the technical basis for assessing flaws in dissimilar metal welds in pressurizer nozzles. The ACRS letter also supported the efforts of the staff to pursue further work on residual stresses including additional efforts to obtain more experimental confirmation of the welding residual stresses.

EDO Response

The EDO response stated that the staff's safety assessment discussed a number of conservatisms and uncertainties related to the advanced finite element analyses and noted that the sensitivity study shows that results are highly dependent upon the weld residual stress profile. The response stated that weld residual stress models are not fully validated by full-scale experiments and that RES is undertaking a robust program to validate the weld residual stress models. **The response also stated that this program is scheduled to be completed by the end of 2008, and that the staff will keep ACRS informed of these efforts.**

Analysis

The EDO's response is satisfactory.

Attachments: As stated

cc: ACRS Members F. Gillespie S. Duraiswamy C. Santos

November 23, 2007

RECEIVED
NOV 26 2007

Dr. William J. Shack, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: RESPONSE TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS LETTER, DATED OCTOBER 19, 2007, CONCERNING NRC STAFF SAFETY ASSESSMENT OF THE INDUSTRY STUDY RELATED TO DISSIMILAR METAL WELD ISSUES IN PRESSURIZER NOZZLES

Dear Dr. Shack:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to your letter, dated October 19, 2007, concerning the NRC staff's safety assessment of the industry's study related to dissimilar metal weld issues in pressurizer nozzles. We note that the Advisory Committee on Reactor Safeguards (ACRS) agrees that the advanced finite element analyses conducted by industry and NRC staff to address potential primary water stress corrosion cracking in pressurizer nozzles led to important advances in fracture mechanics capability and understanding of weld residual stresses.

Your letter indicates that the studies by the industry and the staff have been timely and have also significantly improved the technical basis for the assessment of circumferential flaws in dissimilar metal welds in pressurizer nozzles. Your letter also supports efforts by the staff to pursue further work on residual stresses including additional efforts to obtain more experimental confirmation of the welding residual stresses.

In its safety assessment the staff discussed a number of conservatisms and uncertainties related to the advanced finite element analyses. The staff noted that the sensitivity study shows that results are highly dependent upon the weld residual stress profile. However, weld residual stress models are not fully validated by full-scale experiments. The NRC Office of Nuclear Regulatory Research is undertaking a robust program to validate the weld residual stress models. This program is scheduled to be completed by the end of 2008. The NRC staff will keep ACRS informed of these efforts.

In closing, we value the Advisory Committee's review of the advanced finite element analyses of the pressurizer nozzle dissimilar weld issue. We appreciate your conclusions and recommendations on the issue.

Sincerely,

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Klein
Commissioner Jaczko
Commissioner Lyons
SECY

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

ACRSR-2272

October 19, 2007

Mr. Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: NRC STAFF'S SAFETY ASSESSMENT OF THE INDUSTRY STUDY RELATED TO DISSIMILAR METAL WELD ISSUES IN PRESSURIZER NOZZLES

Dear Mr. Reyes:

During the 546th meeting of the Advisory Committee on Reactor Safeguards, October 4-5, 2007, we discussed the NRC staff's safety assessment of the industry's advanced finite element analysis of primary water stress corrosion cracking (PWSCC) in pressurizer nozzle dissimilar metal butt welds, and other related activities. Preliminary results of this analysis were presented to us at our meeting on July 11-13, 2007. During these meetings, we had the benefit of discussions with representatives of the NRC staff and the industry, and of the documents referenced. We previously provided a report on the dissimilar metal weld issues on March 22, 2007.

CONCLUSION AND RECOMMENDATION

1. The studies undertaken by the industry and the staff have been timely and improved significantly the technical basis for the assessment of circumferential flaws in dissimilar metal welds in pressurizer nozzles.

2. We support the efforts of the staff to pursue further work on residual stresses including additional efforts to obtain more experimental confirmation of the welding residual stresses.

BACKGROUND AND DISCUSSION

In October 2006, ultrasonic examination revealed five indications in three nickel-based dissimilar metal welds joining the ferritic steel nozzles to the austenitic stainless steel coolant piping at the Wolf Creek Generating Plant. These indications were interpreted as large circumferential cracks.

The nickel-based alloys (Alloy 82 and 182) used for these welds are known to be susceptible to PWSCC in the primary coolant environment of pressurized water reactors (PWRs). Only about 15 percent of the dissimilar metal pressurizer nozzle welds in PWRs have been inspected. Since the adjoining base metals are resistant to PWSCC, axial cracks will be limited to a length no greater than the width of the weld. Such cracks may lead to leakage, but are unlikely to lead to rupture or significant loss of coolant. However, circumferential cracks can potentially grow to sizes that could lead to rupture.

Prior to the Wolf Creek findings, the staff and industry had recognized the potential for cracking in such dissimilar metal welds, and the industry had instituted a program to inspect these welds or apply weld overlays similar to those used at Wolf Creek. Most licensees of PWRs with susceptible nozzle welds will complete inspections or apply weld overlays during 2007. However, nine plants plan to perform these activities during outages in the spring of 2008. The plants that have not yet completed inspections or mitigation activities have committed to enhanced leakage detection as a compensatory action until these activities are completed.

After the Wolf Creek findings, the staff took the position that mitigation activities should be completed by the end of 2007 rather than the spring of 2008. The industry presented analyses that suggested the likelihood that nozzles can rupture without prior warning is low, and that the increased risk associated with a schedule for completing inspection and mitigation activities in the spring of 2008 is acceptably small. The industry undertook a program to develop more rigorous analyses. The affected licensees also committed to accelerate the schedule for inspection and mitigation and complete the work by the end of 2007, if the results of the analysis did not demonstrate adequate margin for leak-before-break.

Prior to the initiation of this effort, most fracture mechanics analyses have assumed that the crack shape is either elliptical or constant depth. The improved analysis developed by the industry considers crack growth at each point along the crack front and allows the crack to change shape as dictated by the stress distribution and appropriate crack growth correlations. Even with this increased capability to model the growth of cracks, there are large uncertainties in important variables, such as the welding residual stresses and the applied loads on the welds, that affect the results. In conjunction with the development of the advanced fracture mechanics analysis, the industry also undertook a large study to characterize the range of weld residual stresses and applied loads associated with dissimilar metal welds on pressurizer nozzles. Extensive sensitivity analyses were performed to assess the effects of weld residual stress profile, nozzle dimensions and geometry, initial crack shape, initial crack dimensions, multiple cracks, operational loads, PWSCC crack growth rates, and plastic redistribution of loads.

As part of its review of the industry results, the NRC staff established a substantial, essentially real time, independent confirmatory study to review, benchmark, and verify the results of the industry's advanced fracture mechanics analysis, estimates of the weld residual stresses, and choice of failure criteria. In general, there was excellent agreement between the results of the industry study and the staff's confirmatory study.

The studies performed by the staff and industry show that the assumption that the crack shape remains elliptical or constant depth is conservative compared to the more realistic analysis in which the shape of the crack is not artificially constrained. The sensitivity studies showed that the behavior of the cracks is highly dependent on the weld residual stress profiles. In particular, inner diameter weld repairs create high local tensile stresses, which would cause a postulated crack to grow faster radially than circumferentially and eventually grow throughwall at the location of the repair. As a result, leakage will occur before rupture. Welds without repairs can promote the development of cracks that can grow around a large fraction of the circumference before growing throughwall.

The work performed by the staff and industry has provided a significant increase in the capability to realistically model the growth of flaws in reactor components and will be useful in a variety of applications. Although it has been recognized for many years that the most significant stresses associated with stress corrosion cracks are welding residual stresses, the work done here is the most comprehensive study of welding residual stresses performed to date. The work has helped to characterize the effects of weld repairs and highlight the importance of correctly accounting for interactions between the weld region and the remainder of the piping system.

Despite this improved understanding, there are still significant uncertainties in the state of knowledge regarding cracks that could exist in uninspected welds. The staff and industry have addressed these uncertainties through the selection of initial crack profiles for the analyses. The staff has judged that a 360°/10% throughwall crack and a 21:1 aspect ratio/ 26% throughwall crack are conservative choices. We agree with these choices in light of experience with intergranular stress corrosion cracking, the nature of the indications at Wolf Creek, and the likelihood that, if cracking were extensive, leaks would have appeared at high-stress locations associated with weld repairs.

With these postulated flaws, the analyses show large margins against rupture prior to the occurrence of detectable leakage for the spray, safety, and relief nozzle geometries. The margins for the surge nozzles are significantly smaller, but still adequate unless additional conservatisms are added to the analyses such as an increase in the crack growth rate by a factor of 10. Therefore, the staff has concluded that the advanced finite element analysis provides reasonable assurance that the nine affected plants can operate safely until scheduled outages in spring 2008.

The staff and industry have accomplished a large amount of work under a very demanding schedule to address this problem. The work has led to important advances in fracture mechanics capability and understanding of weld residual stresses. We support the efforts of the staff to pursue further work on residual stresses including experimental confirmation.

Sincerely,

/RA/

William J. Shack
Chairman

References:

1. Memorandum from Michele G. Evans, Director, Division of Component Integrity, Office of Nuclear Reactor Regulation, to Catherine Haney, Director, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation, "Safety Assessment on the Advanced Finite Element Analysis Related to Growth of Postulated Primary Water Stress Corrosion Cracking in Pressurizer Nozzle Dissimilar Metal Butt Welds," dated September 7, 2007 (ML072430091 and ML072400199).

2. Engineering Mechanics Corporation of Columbus, "Final Report on Implication of Wolf Creek Indications," August 2007 (ML072470394).

3. Electric Power Research Institute, "Materials Reliability Program: Advanced FEA Evaluation of Growth of Postulated Circumferential PWSCC Flaws in Pressurizer Nozzle Dissimilar Metal Welds (MRP-216, Revision 1): Evaluation Specific to Nine Subject Plants," August 2007 (ML072410235).

4. Letter from William J. Shack, Chairman, Advisory Committee on Reactor Safeguards, to Luis A. Reyes, Executive Director for Operations, U.S. Nuclear Regulatory Commission, dated March 22, 2007, "Proposed NRC Staff and Industry Activities for Addressing Dissimilar Metal Weld Issues Resulting From the Wolf Creek Pressurizer Nozzle Weld Inspection Results" (ML070810710).

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

December 3, 2007

MEMORANDUM TO: George E. Apostolakis, Chair
Digital I&C Systems Subcommittee

FROM: Girija S. Shukla, Senior Program Manager /RA/
Reactor Safety Branch, ACRS

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER ON DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS PROJECT PLAN AND INTERIM STAFF GUIDANCE

Attached is a copy of the November 21, 2007 EDO letter of response to the October 16, 2007 ACRS letter on the NRC staff's Digital Instrumentation and Control (I&C) Systems Project Plan, and Interim Staff Guidance (ISG). A copy of the Committee's letter is also attached.

Committee Letter

In its October 16, 2007 letter the ACRS concluded that the staff's three interim guidance reports on diversity and defense-in-depth, communications, and human factors will help with the review of anticipated near-term licensing actions related to digital I&C. The Committee recommended that in the longer term, an alternative process to the 30-minute criterion should be developed to determine the conditions under which operator manual actions can be credited as a diverse protective function. Additionally, the issue of spurious actuations needs to be examined further.

EDO Response

The EDO response stated that the staff's agrees with the Committee's recommendation that as part of developing longer term regulatory guidance for D3 assessments for digital systems an alternative process for determining the conditions under which operator manual actions can be credited as a diverse protection function should be explored. As part of the ongoing work of the D3 and human factors TWGs, the staff will investigate the possibility of developing alternative guidance. The staff also agrees with the Committee's recommendation to further examine the issue of spurious actuations as it relates to D3 guidance. Also, as identified by the Committee, the staff will evaluate spurious actuations in developing the longer term regulatory guidance for D3 assessments. In particular, this will include the areas of automatically reconfigurable systems and unintended functions actuated during the progression of a plant transient or accident. Additional assessment of these areas will be included in the long term activities of the Digital I&C Project Plan for Diversity and Defense-In-Depth.

Analysis

The EDO's response is satisfactory.

Attachments: As stated

cc: ACRS Members F. Gillespie S. Duraiswamy C. Santos

November 21, 2007

Dr. William J. Shack, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS PROJECT PLAN AND INTERIM STAFF GUIDANCE

Dear Dr. Shack:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to your letter to Chairman Klein, dated October 16, 2007, which summarized the views of the Advisory Committee on Reactor Safeguards (ACRS or the Committee) regarding the staff's activities related to the Digital Instrumentation and Controls (I&C) Project Plan and Interim Staff Guidance (ISG). The staff and I appreciate your support of the ISGs on diversity and defense-in-depth (D3), highly integrated control room communications, and highly integrated control room human factors.

Since the beginning of the year the staff has been working to develop a number of ISGs through the work of seven task working groups (TWGs) under the direction of the NRC Digital I&C Steering Committee to provide additional guidance in areas of Digital I&C and human factors.

The staff agrees with the Committee's recommendation that as part of developing longer term regulatory guidance for D3 assessments for digital systems an alternative process for determining the conditions under which operator manual actions can be credited as a diverse protection function should be explored. As part of the ongoing work of the D3 and human factors TWGs, the staff will investigate the possibility of developing alternative guidance. The staff also agrees with the Committee's recommendation to further examine the issue of spurious actuations as it relates to D3 guidance. Also, as identified by the Committee, the staff will evaluate spurious actuations in developing the longer term regulatory guidance for D3 assessments. In particular, this will include the areas of automatically reconfigurable systems and unintended functions actuated during the progression of a plant transient or accident.

Additional assessment of these areas will be included in the long term activities of the Digital I&C Project Plan for Diversity and Defense-In-Depth.

The staff and I appreciate the comments and recommendations provided by the ACRS and look forward to continuing to work with the ACRS as the staff completes the remaining ISGs and develop final regulatory guidance.

Sincerely,

/RA Martin J. Virgilio for/

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Klein
Commissioner Jaczko
Commissioner Lyons
SECY

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

ACRSR-2269

October 16, 2007

The Honorable Dale E. Klein
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS PROJECT PLAN AND INTERIM STAFF GUIDANCE

Dear Chairman Klein:

During the 546th meeting of the Advisory Committee on Reactor Safeguards, October 4-5, 2007, we reviewed the Digital Instrumentation and Control (I&C) Systems Project Plan, and Interim Staff Guidance (ISG) prepared by the NRC staff. Our Digital I&C Systems Subcommittee reviewed this matter during a meeting on September 13, 2007. During our review, we had the benefit of discussions with representatives of the NRC staff and Nuclear Energy Institute. We also had the benefit of the documents referenced.

CONCLUSION AND RECOMMENDATIONS

1. The staff's three interim guidance reports on diversity and defense-in-depth, communications, and human factors will help with the review of anticipated near-term licensing actions related to digital I&C.
2. In the longer term, an alternative process to the 30-minute criterion should be developed to determine the conditions under which operator manual actions can be credited as a diverse protective function.
3. The issue of spurious actuations needs to be examined further.

DISCUSSION

The Digital I&C Project Plan includes a process for developing interim guidance to support the review of anticipated near-term licensing actions. The long-term objective is to develop recommendations that will be used to update the Standard Review Plan and other relevant regulatory documents.

Three ISGs were issued recently. They address the issues of diversity and defense-in-depth, communications, and human factors. The guidance contained in these documents is appropriate to support the review of near-term licensing actions related to digital I&C.

One critical issue addressed by the ISG on diversity and defense-in-depth is the acceptability of manual actions to address the need for diversity. The ISG states that when protective action is required within 30 minutes, it is difficult to demonstrate the feasibility and reliability of manual actions. Therefore, in situations where the protective action is required in less than 30 minutes, the ISG identifies the installation of an independent and diverse automated backup system as an acceptable approach. When protective action is not required for at least 30 minutes, the ISG identifies manual actions as acceptable.

The industry argues that each case where manual actions are to be credited should be evaluated on its own merits. A process is needed to determine, on a case-by-case basis, whether an automated backup system should be installed or manual actions could be credited. We believe that a similar process defined in NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire," could be used for the digital I&C systems.

Although in principle we agree with the industry's viewpoint, we recognize the value of the staff's 30-minute criterion. In response to our suggestion, the staff added the following statements to the ISG: "The methods described in this interim staff guidance are not the only methods that the staff may find acceptable. The staff may also find other methods acceptable, but other methods may warrant more in-depth staff review."

The ISG on diversity and defense-in-depth states that potential spurious trips and actuations are of a lesser safety concern than failures to trip or actuate. This assertion may not be justified for spurious signals that automatically reconfigure systems or initiate unintended functions during the progression of a plant transient or accident. Although these actuations should be annunciated in the main control room, they may cause unanticipated conditions that require operator intervention to restore the required safety functions. Further attention is needed to evaluate potential spurious signals that may alter the normal progression of automatic plant response.

We commend the staff for developing ISGs that will help with the review of anticipated near-term licensing actions related to digital I&C. We are also encouraged by the progress and the degree of collaboration between the staff and the industry in addressing the many challenging issues that need to be resolved before updating the Standard Review Plan and other relevant regulatory documents.

Sincerely,

/RA/

William J. Shack
Chairman

REFERENCES

1. Memorandum dated October 3, 2007, from Patricia Silva, Director, Task Working Groups Digital Instrumentation and Controls Project, Office of Nuclear Reactor Regulation, to Cayetano Santos, Chief, Reactor Safety Branch, Advisory Committee on Reactor Safeguards, transmitting:

    - Digital I & C Project Plan, July 12, 2007 (ML071900253).
    - Digital I & C - ISG-02, Diversity and Defense-in-depth, September 26, 2007 (ML072540118).
    - Digital I & C - ISG-04, Highly-Integrated Control Room Digital Communications Systems, September 28, 2007 (ML072540138).
    - Digital I & C - ISG-05, Highly-Integrated Control Room Human Factors, September 28, 2007 (ML072540140).
    - Draft Digital I & C - ISG-XX, Cyber Security Associated with Digital Instrumentation and Controls (ML072260584).

2. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.152 Rev. 2, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants," January 2006.

3. U.S. Nuclear Regulatory Commission, NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire," September 2006.