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

August 24, 2010

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador **/RA/**
Technical Secretary, ACRS

SUBJECT: CERTIFICATION OF THE MEETING MINUTES FROM THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS 563rd FULL COMMITTEE MEETING HELD ON JUNE 3-5, 2009 IN ROCKVILLE, MARYLAND

The minutes of the subject meeting were certified on June 24, 2009 as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment:
As stated

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

June 24, 2009

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

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

SUBJECT: MINUTES OF THE 563rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), JUNE 3-5, 2009

I certify that based on my review of the minutes from the 563rd ACRS Full Committee meeting, and to the best of my knowledge and belief, I have observed no substantive errors or omissions in the record of this proceeding subject to the comments noted below.

OFFICE	ACRS	ACRS:RSB
NAME	SMeador	CSantos/sam
DATE	06/ 24 /09	06/24/09

OFFICIAL RECORD COPY

CERTIFIED Date Certified: 06/24/09

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MINUTES OF THE 563rd ACRS MEETING

June 3-5, 2009

Appendices

During its 563rd meeting, June 3-4, 2009, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following report, letters, and memorandum:

<u>REPORT</u>

Report to Gregory B. Jaczko, Chairman, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Report on the Safety Aspects of the License Renewal Application for the National Bureau of Standards Test Reactor, dated June 16, 2009

<u>LETTERS</u>

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

- Draft Final Revision 2 to Regulatory Guides 1.21, “Measuring, Evaluating and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste,” and 4.1, “Radiological Environmental Monitoring for Nuclear Power Plants,” dated June 17, 2009

- Safety Evaluation for the Mitsubishi Heavy Industries Topical Report MUAP-07006-P, Revision 2, “Defense-In-Depth and Diversity,” Related to the US-APWR Design, dated June 25, 2009

<u>MEMORANDUM</u>

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

- Proposed Revisions to Regulatory Guides 1.174, 1.177, 1.40, 1.68.2, 1.159, DG-3037, and 1.183, dated June 9, 2009

MINUTES OF THE 563rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

ROCKVILLE, MARYLAND

The 563rd meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on June 3-5, 2009. Notice of this meeting was published in the *Federal Register* on May 18, 2009 (72 FR 23222-23224). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting agenda. The meeting was open to public attendance.

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

ATTENDEES

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

I. Chairman's Report (Open)

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

Dr. Mario Bonaca, Committee Chairman, convened the meeting at 8:30 a.m. In his opening remarks he announced that the meeting was being conducted in accordance with the provisions of the Federal Advisory Committee Act. He reviewed the agenda items for discussion and noted that no written comments or requests for time to make oral statements from members of the public had been received. Dr. Bonaca also noted that a transcript of the open portions of the meeting was being kept and speakers were requested to identify themselves and speak with clarity and volume.

II. License Renewal Application and the Revised Final Safety Evaluation Report for the National Institute of Standards and Technology (NIST) Reactor

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

The Committee met with representatives of the National Institute of Standards and Technology (NIST), the applicant, and the NRC staff to discuss the license renewal application for the National Bureau of Standards Reactor (NBSR) and the associated NRC staff's revised final Safety Evaluation Report (SER). Specifically, the discussion was focused on the resolution of one open item related to flow coast-down data used in the loss-of-offsite power accident analysis.

While responding to a question raised at an earlier ACRS Subcommittee meeting, the applicant discovered that the pump coast-down curve for the RELAP analysis was compared to the data measured under different conditions. The applicant promptly reported the error to the NRC on March 30, 2009, and it was briefly discussed during the previous ACRS meeting on April 2, 2009. This was treated as a license renewal open item.

Since then, the applicant has completed its re-analysis. During the meeting, the applicant showed a comparison of the revised flow coast-down data and the flow used in the prior analysis. NIST representatives described the results of the re-analysis and stated that the results do not alter the conclusions presented in the SER.

The NRC staff described its independent review and calculation to assess the safety significance of the error. The staff also discussed the review of the applicant's re-analysis. Based on its review, the staff concluded that there is reasonable assurance that a loss-of-offsite power will not result in fuel damage and that the consequences of the accident are bounded by the Maximum Hypothetical Accident.

The Committee issued a report to the NRC Chairman on this matter, dated June 16, 2009, recommending that the NIST application for renewal of the NBSR operating license be approved.

III. Draft Final Regulatory Guides 1.21 and 4.1

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

The Committee met with representatives of the NRC staff and the Nuclear Energy Institute to discuss Draft Final Regulatory Guide 1.21 (DG-1186), "Measuring, Evaluating, and Reporting Radioactive Materials in Liquid and Gaseous Effluents and Solid Waste," and Draft Final Regulatory Guide 4.1 (DG-4013), "Radiological Environmental Monitoring for Nuclear Power Plants." Both Regulatory Guides were issued for public comment in November 2008. The staff described how the public comments were addressed. The representative from NEI noted a need for an integrated approach implementing guidance documents in this area. The current guides are more than 30 years old, and the revisions are intended to update the NRC staff's guidance and incorporate insights and recommendations from NRC's Liquid Radioactive Release Lessons Learned Task Force.

The Committee issued a letter to the Executive Director for Operations on this matter, dated June 17, 2009, recommending that Revision 2 of Regulatory Guide 1.21 and of Regulatory Guide 4.1 be issued.

IV. Pellet-Clad Interaction Failures under Extended Power Uprate Conditions

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

The Committee met with representatives of the NRC staff and industry to discuss the potential need for developing regulatory criteria to protect against pellet-clad interaction (PCI) failures. On December 20, 2007, the ACRS issued a report to the Commission on the Susquehanna Steam Electric Station Extended Power Uprate. The added comments in that report expressed concern about the use of non-barrier fuel, which is not specifically designed to protect against PCI failures. During the June 3-4, 2009, ACRS meeting, data were presented to the Committee showing that PCI failures may occur in less than five minutes during certain transients, thereby precluding operator actions. NRC staff stated that PCI failures are of low safety significance. Further, developing rules for PCI failures constitutes a change in regulatory position and requires consideration of backfitting. Industry representatives from AREVA, Global Nuclear Fuel, and the Electric Power Research Institute presented various views opposing new regulatory criteria on PCI failures. The Committee decided that this is not an immediate safety concern and a letter is not needed.

V. Diversity and Defense-in-Depth Topical Report Associated with the US-APWR Design

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

The Committee met with representatives of the NRC staff to discuss the Safety Evaluation for Revision 2 of Mitsubishi Heavy Industries (MHI) Topical Report MUAP-07006-P, "Defense-in-Depth and Diversity," for the U.S. Advanced Pressurized Water Reactor (US-APWR). The Topical Report describes MHI's generic methodology to address defense-in-depth and diversity in digital I&C systems. The staff discussed the scope of the topical report and points of discussion identified during the May 21, 2009, Subcommittee meeting to review this matter. The staff also described its findings and conclusions. The staff concluded that the approach documented in the Topical Report and responses to the requests for additional information conform to regulatory requirements. This conclusion is subject to the satisfactory completion of 11 design certification application specific action items documented in the Safety Evaluation.

The Committee issued a letter to the Executive Director for Operations on this matter, dated June 25, 2009, recommending that the staff's Safety Evaluation be issued.

VI. Subcommittee Report

The Chairman of the Reliability and PRA Subcommittee provided a report to the Committee summarizing the results of the June 1-2, 2009, meeting with the NRC staff on the (i) proposed Revision 1 to Regulatory Guide (RG) 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," and proposed Standard Review Plan (SRP) Section 9.5.1.2, "Risk-Informed, Performance-Based Fire Protection," (ii) development of guidelines for performing human reliability analysis in fire probabilistic risk assessments, and (iii) risk metrics for new light-water reactor risk-informed applications.

During the Subcommittee meeting, the staff explained proposed changes to RG 1.205 (DG-1218). This Guide endorses Revision 2 to NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)," and includes integrated lessons learned from observation visits, fire PRA reviews, and plant License Amendment Request (LAR) reviews. The guidance in the proposed new SRP Section 9.5.1.2 is consistent with the proposed changes to RG 1.205.

The EPRI representative stated that RG 1.205 requires the use of conservative methods from NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," and deviations from these methods will require prior NRC approvals. He further stated that outdated prescriptive and conservative methods should not be imposed on the licensees and, instead, the guidance provided in RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," should be considered appropriate.

The staff also presented the issues and options for the implementation of risk metrics for new light water reactor risk-informed applications. The staff discussed the risk-informed initiatives and high level goals and objectives for new reactors, the current risk-informed framework, the new reactor implementation issues, the revised options based on stakeholder feedback, and the preliminary evaluation of options. The staff stated that in the near term, risk-informed applications for new reactors have been proposed as risk-managed technical specifications.

VII. Quality Assessment of Selected Research Projects

The Committee discussed the results of the ACRS Panels' review of the quality assessment of the NRC research projects on the following topics: NUREG/CR-6964, "Crack Growth Rates and Metallographic Examinations of Alloy 600 and Alloy 82/182 from Field and Laboratory Materials Testing in PWR Environments," and Draft NUREG/CR-XXXX, "Diversity Strategies for Nuclear Power Plant Instrumentation and Control Systems." The Committee plans to discuss its draft report on the assessment of the quality of the above projects during its meeting on July 8-10, 2009.

VIII. Executive Session

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

A. Reconciliation of ACRS Comments and Recommendations/EDO Commitments

- The Committee considered the EDO's response of May 8, 2009, to comments and recommendations included in the April 9, 2009, ACRS letter on the Draft Final Revision 2 to Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of May 18, 2009, to comments and recommendations included in the April 21, 2009, ACRS letter on Draft Final Regulatory Guide 1.211, "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants." The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of April 28, 2009, to comments and recommendations included in the March 19, 2009 ACRS letter on Draft Final Regulatory Guide 5.73, "Fatigue Management of Nuclear Power Plant Personnel." The Committee decided that it was satisfied with the EDO's response.

B. Report of the Planning and Procedures Subcommittee Meeting

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

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

Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through September 2009 were discussed and the objectives were to:

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

REVISED ACRS SUBCOMMITTEE STRUCTURE

The revised ACRS Subcommittee Structure was discussed. The revision involves core member assignments for each Subcommittee; elimination of completed tasks and addition of new tasks; reinstatement of the Naval Reactors Subcommittee; Chairmanship assignments; and the creation of a new “Siting” Subcommittee that replaces the Early Site Permits Subcommittee. A brief description of the above change is as follows:

Core Member Assignments

During its January 2009 retreat, the Committee asked the ACRS staff to assign core members to each ACRS Subcommittee. Accordingly, such assignments have been made and discussed with individual ACRS members (pp.). The members are reminded that core member assignments will not prohibit them from attending Subcommittee meetings of interest to them even though they are not members of those Subcommittees. However, it is the responsibility of the members to make sure that they do not exceed the 130-day limit.

Subcommittee Tasks

Completed tasks have been eliminated, some task have been revised, and new tasks have been added.

Reinstatement of the Naval Reactors Subcommittee

Naval Reactors Subcommittee has been reinstated to review the technical aspects of the reactor for the new aircraft carrier.

Creation of a New Subcommittee

A new Subcommittee named “Siting” has been created to replace the existing Early Site Permits Subcommittee. There are no new early site permit applications expected in the future. Site permit review is expected to be conducted in parallel with the design certification and COL review. The Siting Subcommittee will review the site-related issues such as seismic issues, implementation of the lessons learned form the review of the early site permit applications, and other site-related issues.

Chairmanship Assignments

-Mr. Stetkar will Chair the Subcommittee on Naval Reactors

-Dr. Powers, current Chairman of the Early Site Permits Subcommittee, will Chair the Subcommittee on Siting.

-Dr. Bley will Chair the Subcommittee on Future Plant Designs.

-Dr. Corradini, the current Chairman, will remain as a member and will Chair this Subcommittee when reviewing the Advanced Reactor Research Plan.

The revised Subcommittee Structure was sent to the members on May 26, 2009 for comment. This revision reflects incorporation of comments received. This revised Subcommittee Structure will become effective on June 8, 2009. The ACRS Subcommittee structure will be revised as needed to balance the workload among members and ACRS staff.

WEBSTREAMING OF THE ACRS MEETINGS

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

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

During the May meeting, the Committee decided to establish a Panel to discuss the pros and cons of participating in the Webstreaming program and provide recommendations for use in making its decision. The panel consists of:

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

The Panel should provide its recommendations by September 15, 2009.

ACRS REVIEW OF SAFEGUARS AND SECURITY MATTERS (EMH)

The ACRS has been reviewing regulatory matters in the areas of Safeguards and Security consistent with the Commission guidance in the October 31, 2003 Staff Requirements Memorandum (SRM). In that SRM, the Commission stated the following:

In the security arena, the ACRS should continue to focus attention and expertise on technical issues associated with the progression and potential consequences of postulated terrorist actions, and the assessment of the effectiveness mitigation strategies. The ACRS should not involve itself in issues associated with threat assessment (i.e. assessment of the likelihood of various types of events), physical security, or force-on-force assessments since these are outside the Committees area of expertise, and involves intelligence information not available to the Committee.

As a result of his recent conversations with some Commissioners, Dr. Bley raises the issue of whether the ACRS should become more involved in safeguards/security issues than it has been.

Please note that unless the Commission issues another SRM to supersede the October 2003 SRM, the Committee has no choice but to comply with the directions in the October 2003 SRM. The Committee should discuss whether it really wants to get involved in reviewing issues associated with the physical security and force-on-force assessments.

SCHEDULING OF SUBCOMMITTE MEETINGS

The Committee has a long-standing policy not to hold Subcommittee meetings in August so that the members and staff can take vacation. The new members may not be aware of this policy and some of the experienced members have forgotten about the existence of this policy. Consequently, meetings have been scheduled to be held in August. Such a practice precludes some members and staff from taking vacation. Some members propose an alternative that Subcommittee meetings not be held between mid-July and mid- August.

REGULATORY GUIDES

a) Draft Final Regulatory Guides

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

Draft Final Revision 3 Regulatory Guide 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants"

Revision 3 to Regulatory Guide 1.100 was issued for public comment as draft guide (DG) 1175 in May 2008. Regulatory Guide 1.100 endorses, with exceptions, IEEE Standard 344-2004, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," and ASME QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants." Several important changes have been made in this revision. First, Regulatory Guide 1.148, "Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants," is being subsumed into this revision so that all guidance for seismic qualification of equipment will be contained in one document. (Regulatory Guide 1.148 will be withdrawn after issuance of Revision 3 of Regulatory Guide 1.100.) Second, the guidance for use of earthquake and test experience data has been greatly expanded for seismic qualification of both electrical and active mechanical equipment. Finally, guidance has been added for plants with high-frequency ground motion, i.e., greater than 33 Hz, in their required response spectra.

Draft Final Regulatory Guide 1.215, "Guidance for ITAAC Closure Under 10 CFR Part 52"

Regulatory Guide 1.215 (DG-1204) was issued for public comments on March 13, 2009. The comment period closed on May 13, 2009. This Guide describes a method that the staff considers acceptable for use in satisfying the requirements for documenting the completion of inspections, tests, analyses, and acceptance criteria (ITAAC). In particular, this guide endorses the methodologies described in the industry guidance document Nuclear Energy Institute (NEI) 08-01, "Industry Guideline for the ITAAC Closure Process Under 10 CFR Part 52," Revision 3, issued January 2009, for the implementation of 10 CFR 52.99, "Inspection During Construction."

A December 5, 2008 SRM states that "*[t]he staff should provide the Commission an opportunity to review the guidance before reaching a decision to endorse the 'Industry Guideline for ITAAC Closure Process under 10 CFR Part 52,' NEI 08-01."* The staff's due date to the Commission is 8/27/09. The staff requests that if the ACRS review this guide, it do so during the July 2009 meeting.

b) Proposed Regulatory Guides

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

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

The staff issued Revision 1 to RG 1.174 in November 2002. This Guide provides guidance for using risk information in support of licensee-initiated licensing basis (LB) changes to a nuclear power plant that require such review and approval. The guidance provided does not preclude other approaches for requesting LB changes. Rather, this Guide is intended to improve consistency in regulatory decisions in areas in which the results of risk analyses are used to help justify regulatory action. As such, this Guide, the use of which is voluntary, provides general guidance concerning one approach that the NRC has determined to be acceptable for analyzing issues associated with proposed changes to a plant's LB and for assessing the impact of such proposed changes on the risk associated with plant design and operation. This guidance does not address the specific analyses needed for each nuclear power plant activity or design characteristic that may be amenable to risk-informed regulation.

Based on his review of this Proposed Regulatory Guide, Dr. Apostolakis recommends that the Committee review the draft final revision to Regulatory Guide 1.174 after reconciliation of public comments.

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

The staff issued RG 1.177 in August 1998. The respective revised rule 10 CFR 50.86, is in the process of being issued for public comment. This Guide provides the staff's guidance for using risk information to evaluate changes to nuclear power plant technical specifications allowed outage times (AOTs) and (surveillance time intervals (STIs) in order to assess the impact of such changes on the risk associated with plant operation. Other types of TS changes that follow the principles outlined in this Guide may be proposed and will be considered on their own merit. The guidance provided here does not preclude other approaches for requesting TS changes. Rather, this Guide is intended to improve consistency in regulatory decisions related to TS changes in which the results of risk analyses are used to help justify the change. As such, this Guide, the use of which is voluntary, provides guidance concerning an approach that the NRC has determined to be acceptable for analyzing issues associated with proposed changes to a plant's TS and for assessing the impact of such changes on the risk associated with plant design and operation.

Based on his review of this Proposed Regulatory Guide, Dr. Bley recommends that the Committee review the draft final revision to Regulatory Guide 1.177 after reconciliation of public comments.

Proposed Revision 1 to Regulatory Guide 1.40 (DG 1150), "Qualification of Continuous Duty Safety-Related Motors for Nuclear Power Plants"

Revision 1 to this Guide endorses the updated IEEE Standard. The Working Group on Qualification of Motors of the Nuclear Power Engineering Committee of the IEEE; developed IEEE Std. 334-2006, "Qualifying Continuous Duty Class 1E Motors for Nuclear Power Generating Stations." The IEEE Standards Board approved IEEE Std. 334-2006 on September 15, 2006, and it was published on January 31, 2007. This standard establishes criteria for qualifying continuous duty Class 1E motors used in mild and harsh environments in nuclear power plants to demonstrate their ability to perform their intended safety functions. The standard also provides guidance for the qualification of refurbished motors and insulation systems for motor rewinds. The standard is the updated version of IEEE Std. 334-1971, which the NRC endorsed in Regulatory Guide 1.40, "Qualification Tests of Continuous-Duty Motors Installed inside the Containment of Water-Cooled Nuclear Power Plants," issued March 1973.

Based on his review of this Proposed Regulatory Guide, Mr. Stetkar recommends that the Committee review the draft final revision to Regulatory Guide 1.40 after reconciliation of public comments.

Proposed Revision 2 to Regulatory Guide 1.68.2 (DG 1236), "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water Cooled Nuclear Power Plants"

Revision 1 of Regulatory Guide 1.68.2 was issued in July 1978. The objective of this revision is to provide clear and up-to-date guidance for developing and conducting a test program to demonstrate remote shutdown capability. Staff experience and interaction with applicants since that time have identified deficiencies in the guide that should be corrected. For example, some applicants did not understand that GDC 19 requires the licensee to demonstrate the ability to trip the reactor from outside the control room as well as maintain it in a safe condition during hot shutdown. Additionally, questions and comments from licensees identified the need for clarification on the role of additional personal in the control room during the testing. These individuals may be performing non-safety-related activities that would not be required during an actual emergency. Finally, many of the initial startup test programs submitted for review did not fully address the second requirement in GDC 19, namely, the ability to take a reactor from hot shutdown to cold shutdown from outside the control room. This last provision is of considerable importance since demonstration of this capability lends the added assurance that, in the event a fire or other event causes the control room to become unusable for an indeterminate length of time, no danger to the health and safety of the public from potential loss of controlled residual heat removal capability would result.

Based on his review of this Proposed Regulatory Guide, Mr. Sieber recommends that the Committee review the final revision to Regulatory Guide 1.68.2 after reconciliation of public comments.

Proposed Revision 2 to Regulatory Guide 1.159 (DG 1229), “Assuring the Availability of Funds for Decommissioning Nuclear Reactors”

The staff published Revision 1 to Regulatory Guide 1.159 in October 2003 to reflect changes in the regulations and to include guidance on the amendments to 10 CFR 50.75. Revision 2 of Regulatory Guide 1.159 provides clarification of certain concepts. The most substantive changes are found in Section C, “Regulatory Position,” and involve: (1) paragraph 3 of Subsection 1.3, “Decommissioning Cost Estimates”; (2) Subsection 2.1.5 of Section 2.1, “Guidance Applicable to All Methods of Financial Assurance”; and (3) Subsection 2.2.8 of Section 2.2, “Prepayment and External Sinking Fund.” The changes in (1) are primarily word changes in paragraph 3 for clarification. The changes in (2) relate to a change in the timing for making adjustments to the licensee’s financial assurance amount(s) and mechanism(s). The changes to (3) specify when a greater than 2 percent real rate of return will be allowed and reflect any withdrawals made during the safe-store period when taking the allowed credit through the projected decommissioning period.

Based on his review of this Proposed Regulatory Guide, Dr. Ryan recommends that the Committee review the final revision to Regulatory Guide 1.159 after reconciliation of public comments.

Proposed New Draft Regulatory Guide

The staff plans to issue the following new Draft Regulatory Guide for public comment and would like to know whether the Committee wants to review this Guide prior to being issued for public comment.

Proposed New Draft Regulatory Guide (DG 3037), “Guidance for Fuel Cycle Facility Change Process”

DG-3037 is a proposed new Regulatory Guide. This proposed guidance is to assist licensees in providing more consistent evaluations and reports with the fact that fuel cycle facilities have different purposes, designs, and safety programs. The requirements in 10 CFR 70.72, “Facility Changes and Change Process,” require certain processes to be implemented to control facility configuration. Based on a threshold, some facility changes require NRC approval. Other facility changes, which do not require NRC approval, are required to be summarized and reported annually.

Based on his review of this Proposed New Draft Regulatory Guide, Dr. Ryan recommends that the Committee review the final revision to this new Guide after reconciliation of public comments.

THIRD QUADRIPARTITE WORKING GROUP MEETING

Japan’s Nuclear Safety Commission (NSC) will host the third Quadripartite Working Group (WG) Meeting in Tokyo scheduled for October 13-15, 2009 on the main topic of Digital I&C. Japan proposed an additional day to discuss seismic issues. France and Germany have indicated their preference for 1.5 days on Digital I&C and an afternoon on seismic issues. France has confirmed they will not present on the seismic issues. The third day, October 15th, is the site tour.

REAPPOINTMENT OF ACRS MEMBERS (EMH)

The Commission has reappointed Dr. Powers for a fifth term, Drs. Armijo and Banerjee for a second term. In the SRM related to the reappointment of Dr. Powers, the Commission states the following:

> The staff should continue to recruit new members to the Committee. In safety significant matters, it is important to continually evaluate staff positions from diverse point of view. With an overarching priority of maintaining the highest level of technical competency, the Committee should maintain a mix of new members and more senior members to ensure diversity while still maintaining some continuity of knowledge during the review of safety issues.

FEDERAL REGISTER NOTICE AND PRESS RELEASE TO SOLICIT CANDIDATES FOR MEMBERSHIP ON THE ACRS

Draft federal register notice and press release to solicit candidates for membership on the ACRS were sent to the Commission for approval. In the SRM approving issuance of these documents for publication, the Commission states the following:

> With an overarching priority of maintaining the highest level technical competency, the ACRS should work to ensure a diverse group of individuals is considered during the interview process.

The meeting was adjourned at 12:00 noon on June 5, 2009.

as integrating offsite response methodologies with onsite EP programs, is also provided in the ISG. Once the EP final rule is published and NRC staff completes the ISG, the staff will issue it for use. The NRC staff will incorporate the updated guidance information in NSIR/DPR–ISG–01 into future revisions of NUREG–0654/FEMA–REP–1, ''Criteria for the Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,'' and other EP guidance documents.

Some NRC EP regulatory requirements are being revised that warrant guidance outside the scope of the proposed ISG. The NRC staff plans to provide additional guidance for addressing proposed changes to Section 50.54(q) concerning emergency plan changes in the form of a new Regulatory Guide (RG), proposed changes to Appendix E to 10 CFR Part 50 regarding emergency action levels for hostile action events in a revision to RG 1.101, ''Emergency Planning for Nuclear Power Plants,'' and proposed changes to Section 50.47(b)(10) and Appendix E to 10 CFR Part 50 regarding updates to evacuation time estimates in the form of a new NUREG/CR document. The Federal Emergency Management Agency (FEMA) is addressing offsite EP guidance changes in support of the proposed EP rule and other offsite EP program issues in a new supplement to NUREG–0654/FEMA–REP–1, as well as other FEMA documents. These documents will be issued separately for public comment prior to publishing the EP final rule.

DATES: Comments must be filed no later than 75 days from the date of publication of this notice in the **Federal Register.** Comments received after this date will be considered, if it is practical to do so, but the NRC staff is able to ensure consideration only for comments received on or before this date.

ADDRESSES: NSIR/DPR–ISG–01, ''Emergency Planning for Nuclear Power Plants,'' is available for inspection and copying for a fee at the NRC Public Document Room, Public File Area O1–F21, 11555 Rockville Pike, Rockville, Maryland. Publicly available documents created or received at the NRC after November 1, 1999, are available electronically at the NRC's Electronic Reading Room at *http://www.nrc.gov/reading-rm.html.* From this site, the public can gain entry into the NRC's Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. The ADAMS Accession number for this ISG is: ML083540070. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737, or by e-mail at *pdr.resource@nrc.gov.*

Comments will be made available to the public in their entirety; personal information, such as your name, address, telephone number, e-mail address, etc., will not be removed from your submission. You may submit comments by any one of the following methods: *Federal e-Rulemaking Portal:* Go to *http://www.regulations.gov*; search on Docket ID: NRC–2008–0122. Direct questions about NRC dockets to Carol Gallagher by telephone at 301–492–3668 or e-mail at *carol.gallagher@nrc.gov. Mail comments to:* Michael T. Lesar, Chief, Rulemaking and Directives Branch, Office of Administration, Mail Stop: TWB–05–B01M, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Cite the publication date and page number of this **Federal Register** notice.

FOR FURTHER INFORMATION CONTACT: Mr. Donald R. Tailleart, Division of Preparedness and Response, Office of Nuclear Security and Incident Response, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001; telephone 301–415–2966 or e-mail at *don.tailleart@nrc.gov.*

SUPPLEMENTARY INFORMATION: The NRC posts its issued staff guidance on the NRC external web page at *http://www.nrc.gov/reading-rm/doc-collections/isg/.*

The NRC staff is issuing this notice to solicit public comments on the proposed NSIR/DPR–ISG–01. After the NRC staff considers any public comments, it will make a determination regarding the proposed NSIR/DPR–ISG–01.

Dated at Rockville, Maryland, this 5th day of May 2009.

For the Nuclear Regulatory Commission.

Melvyn N. Leach,

Director, Division of Preparedness and Response, Office of Nuclear Security and Incident Response.

[FR Doc. E9–11036 Filed 5–15–09; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards

In accordance with the purposes of Sections 29 and 182b of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on June 3–5, 2009, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Monday, October 6, 2008, (73 FR 58268–58269).

Wednesday, June 3, 2009, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

8:35 a.m.–9:45 a.m.: License Renewal Application and the Revised Final Safety Evaluation Report for the National Institute of Standards and Technology (NIST) Reactor (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and NIST regarding the License Renewal Application for the NIST Reactor, the associated NRC staff's revised final SER, and related matters.

10 a.m.–12 p.m.: Draft Final Regulatory Guides 1.21 and 4.1 (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding Draft Final Regulatory Guide 1.21 (DG–1186), ''Measuring, Evaluating, and Reporting Radioactive Materials in Liquid and Gaseous Effluents and Solid Wastes,'' and Draft Final Regulatory Guide 4.1 (DG–4013), ''Radiological Environmental Monitoring for Nuclear Power Plants,'' and related matters.

1 p.m.–3 p.m.: Pellet-Clad Interaction Failures under Extended Power Uprate Conditions (Open/Closed)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and nuclear industry regarding pellet-clad interaction failures under extended power uprate conditions, and related matters. [**Note:** A portion of this Session may be closed pursuant to 5 U.S.C. 552b(c)(4), to discuss project information that is proprietary to Global Nuclear Fuel (GNF) and/or Westinghouse, or their contractors.]

3:15 p.m.–4:45 p.m. Diversity and Defense-in-Depth Topical Report Associated with the US–APWR Design (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Mitsubishi Heavy Industries, Ltd., regarding the Diversity and Defense-in-Depth Topical Report and the associated NRC staff's Safety Evaluation Report associated with the US–Advanced Pressurized Water Reactor (US–APWR) Design, and related matters.

5 p.m.–5:15 p.m.: Subcommittee Report (Open)—The Committee will hear a report by and hold discussions with the Chairmen of the Reliability and PRA Subcommittee regarding (i) proposed Rev. 1 to Regulatory Guide 1.205, ''Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants,'' and proposed Standard Review Plan Section 9.5.1.2, ''Risk-Informed, Performance-Based Fire Protection, (ii) development of guidelines for performing human reliability analysis in fire probabilistic risk assessments, and (iii) risk metrics for new light-water reactor risk-informed applications, that were discussed during the meeting on June 1–2, 2009.

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

Thursday, June 4, 2009, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

8:35 a.m.–9:30 a.m.: Quality Assessment of Selected Research Projects (Open)—The Committee will hear reports by and hold discussions with the members of the ACRS Panels regarding the quality assessment of the NRC research projects on: NUREG/CR–6964, ''Crack Growth Rates and Metallographic Examinations of Alloy 600 and Alloy 82/182 from Field and Laboratory Materials Testing in PWR Environments,'' and Draft NUREG/CR–XXXX, ''Diversity Strategies for Nuclear Power Plant Instrumentation and Control Systems.''

9:45 a.m.–10:45 a.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings and other matters related to the conduct of ACRS business. [**Note:** A portion of this Session may be closed pursuant to 5 U.S.C. 552b(c)(2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.]

10:45 a.m.–11 a.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

11:15 a.m.–12:15 p.m.: Discussion of Topics for Meeting with the Commission (Open)—The Committee will discuss the following topics scheduled for the meeting with the Commission on June 4, 2009: Crediting Containment Accident Pressure in the NPSH Calculations, Pressurized Thermal Shock Rule, Digital Instrumentation and Control Matters, Options to Revise NRC Regulations Based on the International Commission on Radiation Protection (ICRP) Recommendations, and Progress on Recommendations of the Independent External Review Panel on the Materials Licensing Program.

1:30 p.m.–3:30 p.m.: Meeting with the Commission (Open)—The Committee will meet with the Commission at the Commissioners' Conference Room, One White Flint North, to discuss the topics noted above.

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

Friday, June 5, 2009, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

12:30 p.m.–1 p.m.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of Committee activities and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 6, 2008, (73 FR 58268–58269). In accordance with those procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting. Persons desiring to make oral statements should notify the Cognizant ACRS staff named below five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Cognizant ACRS staff prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Cognizant ACRS staff if such rescheduling would result in major inconvenience.

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

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, as well as the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Girija Shukla, Cognizant ACRS staff (301–415–6855), between 7:15 a.m. and 5 p.m. (ET). ACRS meeting agenda, meeting transcripts, and letter reports are available through the NRC Public Document Room at *pdr.resource@nrc.gov,* or by calling the PDR at 1–800–397–4209, or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS) which is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* or *http://www.nrc.gov/reading-rm/doc-collections/ACRS/*.

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

Dated: *May 12, 2009.*

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E9–11531 Filed 5–15–09; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

[OMB Control No. 3206–0167; Forms RI 34–1, RI 34–3, RI 34–17, and RI 34–19]

Submission for OMB Review; Request for Clearance of a Revised Information Collection

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: In accordance with the Paperwork Reduction Act of 1995 (Pub. L. 104–13, May 22, 1995), this notice announces that the Office of Personnel Management (OPM) has submitted to the Office of Management and Budget (OMB) a request for clearance of a revised information collection. This information collection, ''Financial Resources Questionnaire'' (OMB Control No. 3206–0167; Forms RI 34–1 and RI 34–17), collects detailed financial information for use by OPM to determine whether to agree to a waiver, compromise, or adjustment of the collection of erroneous payments from the Civil Service Retirement and Disability Fund. ''Notice of Amount Due Because Of Annuity Overpayment'' (OMB Control No. 3206–0167; forms RI 34–3 and RI 34–19), informs the annuitant about the overpayment and collects information from the annuitant about how repayment will be made.

Approximately 450 RI 34–1 and 70 RI 34–17 forms are completed annually. Approximately 1,351 RI 34–3 and 210 RI 34–19 forms are completed annually. Each form takes approximately 60 minutes to complete. The annual estimated burden is 450 hours (RI 34–1), 70 hours (RI 34–17), 1,351 hours (RI 34–3) and 210 hours (RI 34–19) respectively. The total annual estimated burden is 2,081 hours.

For copies of this proposal, contact Cyrus S. Benson by telephone at (202) 606–0623, by FAX (202) 606–0910, or by e-mail at *Cyrus.Benson@opm.gov.* Please include a mailing address with your request.

DATES: Comments on this proposal should be received within 30 calendar days from the date of this publication.

ADDRESSES: Send or deliver comments to:

James K. Freiert, Deputy Assistant Director, Retirement Services Program, Center for Retirement and Insurance Services, U.S. Office of Personnel Management, 1900 E Street, NW., Room 3305, Washington, DC 20415–3500; and

Alexander Hunt, OPM Desk Officer, Office of Information & Regulatory Affairs, Office of Management and Budget, New Executive Office Building, 725 17th Street, NW., Room 10235, Washington, DC 20503.

For information regarding administrative coordination contact: Cyrus S. Benson, Team Leader, Publications Team, RIS Support Services/Support Group, U.S. Office of Personnel Management, 1900 E Street, NW., Room 4H28, Washington, DC 20415, (202) 606–0623.

U.S. Office of Personnel Management.

John Berry,

Director.

[FR Doc. E9–11506 Filed 5–15–09; 8:45 am]

BILLING CODE 6325–38–P

OFFICE OF PERSONNEL MANAGEMENT

Proposed Collection; Comment Request for Review of a Revised Information Collection: (OMB Control No. 3206–0034; RI 30–2)

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: In accordance with the Paperwork Reduction Act of 1995 (Pub. L. 104–13, May 22, 1995), this notice announces that the Office of Personnel Management (OPM) intends to submit to the Office of Management and Budget (OMB) a request for review of a revised information collection. This information collection, ''Annuitant's Report of Earned Income'' (OMB Control No. 3206–0034; RI 30–2), is used annually to determine if disability retirees under age 60 have earned income which will result in the termination of their annuity benefits.

Comments are particularly invited on: Whether this collection of information is necessary for the proper performance of functions of the Office of Personnel Management, and whether it will have practical utility; whether our estimate of the public burden of this collection of information is accurate, and based on valid assumptions and methodology; and ways in which we can minimize the burden of the collection of information on those who are to respond, through the use of appropriate technological collection techniques or other forms of information technology.

We estimate 21,000 RI 30–2 forms are completed annually. The RI 30–2 takes approximately 35 minutes to complete for an estimated annual burden of 12,250 hours. For copies of this proposal, contact Cyrus S. Benson on (202) 606–4808, FAX (202) 606–0910 or via E-mail to *Cyrus.Benson@opm.gov.* Please include a mailing address with your request.

DATES: Comments on this proposal should be received within 60 calendar days from the date of this publication.

ADDRESSES: Send or deliver comments to—James K. Freiert, Deputy Assistant Director, Retirement Services Program, Center for Retirement and Insurance Services, U.S. Office of Personnel Management, 1900 E Street, NW., Room 3305, Washington, DC 20415–3500.

For information regarding administrative coordination contact: Cyrus S. Benson, Team Leader, Publications Team, RIS Support Services/Support Group, U.S. Office of Personnel Management, 1900 E Street, NW., Room 4H28, Washington, DC 20503, (202) 606–0623.

U.S. Office of Personnel Management.

John Berry,

Director.

[FR Doc. E9–11525 Filed 5–15–09; 8:45 am]

BILLING CODE 6325–38–P

SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–59904; File No. SR–ISE–2009–22]

Self-Regulatory Organizations; Notice of Filing and Immediate Effectiveness of Proposed Rule Change by International Securities Exchange Relating to Far Away Market Maker Fees

May 12, 2009.

Pursuant to Section 19(b)(1) of the Securities Exchange Act of 1934 (the ''Act''),[1] and Rule 19b–4 thereunder,[2] notice is hereby given that on April 29, 2009, the International Securities Exchange, LLC (the ''Exchange'' or the ''ISE'') filed with the Securities and Exchange Commission (''Commission'') the proposed rule change as described in Items I, II, and III below, which items have been prepared by the self-regulatory organization. The ISE has filed the proposed rule change as one establishing or changing a due, fee, or other charge imposed by the ISE under Section 19(b)(3)(A)(ii) of the Act[3] and

[1] 15 U.S.C. 78s(b)(1).

[2] 17 CFR 240.19b–4.

[3] 15 U.S.C. 78s(b)(3)(A)(ii).

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

May 22, 2009

AGENDA
563rd ACRS MEETING
JUNE 3-5, 2009

WEDNESDAY, JUNE 3, 2009, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 – 9:45 A.M. License Renewal Application and the Revised Final Safety Evaluation Report for the National Institute of Standards and Technology (NIST) Reactor (Open) (JDS/PW)
2.1) Remarks by the Subcommittee Chairman
2.2) Briefing by and discussions with representatives of the NRC staff and NIST regarding the License Renewal Application for the NIST Reactor, the associated NRC staff's revised final SER, and related matters.

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

9:45 – 10:00 A.M. * BREAK *****

3) 10:00 – 12:00 P.M. Draft Final Regulatory Guides 1.21 and 4.1 (Open) (MTR/NMC)
3.1) Remarks by the Subcommittee Chairman
3.2) Briefing by and discussions with representatives of the NRC staff regarding Draft Final Regulatory Guide 1.21 (DG-1186), "Measuring, Evaluating, and Reporting Radioactive Materials in Liquid and Gaseous Effluents and Solid Wastes," and related matters.
3.3) Briefing by and discussions with representatives of the NRC staff regarding Draft Final Regulatory Guide 4.1 (DG-4013), "Radiological Environmental Monitoring for Nuclear Power Plants," and related matters.

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. Pellet-Clad Interaction Failures under Extended Power Uprate Conditions (Open/Closed) (JSA/MLB)

4.1) Remarks by the Subcommittee Chairman

4.2) Briefing by and discussions with representatives of the NRC staff and nuclear industry regarding pellet-clad interaction failures under extended power uprate conditions, and related matters.

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

[NOTE: A portion of this Session may be closed pursuant to 5 U.S.C. 552b (c)(4) to discuss information that is proprietary to Global Nuclear Fuel (GNF) and/or Westinghouse, or their contractors]

3:00 – 3:15 P.M. * BREAK *****

5) 3:15 – 4:45 P.M. Diversity and Defense-in-Depth Topical Report Associated with the US-APWR Design (Open) (OLM/NMC)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff and Mitsubishi Heavy Industries, Ltd. regarding the Diversity and Defense-in-Depth Topical Report and the associated NRC staff's Safety Evaluation report associated with the US-Advanced Pressurized Water Reactor (US-APWR) Design and related matters.

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

[NOTE: A portion of this Session may be closed pursuant to 5 U.S.C. 552b (c)(4) to discuss information that is proprietary to Mitsubishi Heavy Industries and/or its contractors]

4:45 – 5:00 P.M. * BREAK *****

6) 5:00 – 5:15 P.M. Subcommittee Report (Open) (GEA/DCB/GSS/YKS)

Report by and discussions with the Chairmen of the Reliability and PRA Subcommittee regarding (i) proposed Rev. 1 to Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," and proposed Standard Review Plan Section 9.5.1.2, "Risk-Informed, Performance-Based Fire Protection," (ii) development of guidelines for performing human reliability analysis in fire probabilistic risk assessments, and (iii) risk metrics for new light-water reactor
risk-informed applications, that were discussed during the meeting on June 1-2, 2009.

7) 5:15 – 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
7.1) License Renewal Application and the Revised Final Safety Evaluation Report for the National Institute of Standards and Technology (NIST) Reactor (JDS/PW)

7.2) Draft Final Regulatory Guides 1.21 and 4.1 (MTR/NMC)
7.3) Pellet-Clad Interaction Failures under Extended Power Uprate Conditions (JSA/MLB)
7.4) Diversity and Defense-in-Depth Topical Report Associated with the US-APWR Design (OLM/NMC)

THURSDAY, JUNE 4, 2009, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

9) 8:35 – 9:30 A.M. Quality Assessment of Selected Research Projects (Open) (DAP/HPN)
9.1) Remarks by the Subcommittee Chairman
9.2) Report by and discussions with members of the ACRS Panels which performed the quality assessment of the NRC research projects on: NUREG/CR-6964, "Crack Growth Rates and Metallographic Examinations of Alloy 600 and Alloy 82/182 from Field and Laboratory Materials Testing in PWR Environments," and Draft NUREG/CR-XXXX, "Diversity Strategies for Nuclear Power Plant Instrumentation and Control Systems."

9:30 – 9:45 A.M. * BREAK *****

10) 9:45 – 10:45 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed) (MVB/EMH)
10.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the Full Committee during future ACRS meetings.
10.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

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

11) 10:45 – 11:00 A.M. Reconciliation of ACRS Comments and Recommendations (Open) (MVB/CS/AFD)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

11:00- 11:15 A.M. ***BREAK***

12) 11:15 – 12:15 P.M. Discussion of Topics for Meeting with the Commission (Open) (MVB, et al./EMH, et al.)
Discussion of following topics for meeting with the Commission:

- Overview
- Crediting Containment Accident Pressure in the NPSH Calculations
- Pressurized Thermal Shock Rule
- Digital Instrumentation and Control Matters
- Options to Revise NRC Regulations Based on the International Commission on Radiation Protection (ICRP) Recommendations / Progress on Recommendations of the Independent External Review Panel on the Materials Licensing Program

12:15 – 1:30 P.M. * LUNCH *****

13) 1:30 – 3:30 P.M. Meeting with the Commission (Open) (MVB, et al. /EMH, et al.)
Meeting with the Commission, Commissioners' Conference Room, One White Flint North, to discuss topics listed under Item 12.

3:30 – 4:00 P.M. * BREAK *****

14) 4:00 – 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

14.1) License Renewal Application and the Revised Final Safety Evaluation Report for the National Institute of Standards and Technology (NIST) Reactor (JDS/PW)
14.2) Draft Final Regulatory Guides 1.21 and 4.1 (MTR/NMC)
14.3) Pellet-Clad Interaction Failures under Extended Power Uprate Conditions (JSA/MLB)
14.4) Diversity and Defense-in-Depth Topical Report Associated with the US-APWR Design (OLM/NMC)

<u>FRIDAY, JUNE 5, 2009, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND</u>

15) 8:30 – 12:30 P.M. **(10:30-10:45 A.M. BREAK)**

<u>Preparation of ACRS Reports</u> (Open)
Continue discussion of the proposed ACRS reports listed under Item 14.

16) 12:30 – 1:00 P.M.

<u>Miscellaneous</u> (Open) (MVB/EMH)
Discussion of matters related to the conduct of Committee activities and specific issues that were not completed during previous meetings, as time and availability of information permit.

<u>NOTES:</u>

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

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

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

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
563rd FULL COMMITTEE MEETING

June 3-5, 2009

PLEASE PRINT

TODAY'S DATE: June 3, 2009

	NAME	NRC ORGANIZATION
1	William B Kennedy	NRR/DPR/PRTA
2	Steve Garry	NRR/DIRS
3	Richard L. Conatser	NRR/DIRS
4	Thomas Blount	NRR/DPR
5	Jeremy Silver	NRR
6	Kathryn Brock	NRR/DPR
7	J C Shepherd	FSME/DWMEP
8	Edward O'Donnell	RES
9	James A. Isom	NRR/DIRS
10	JOHN VOGLEWEDE	RES/DSA
11	Paul Clifford	NRR/DSS
12	Bill Ruland	NRR/DSS
13	Joe Ashcraft	NRO/DE/ICE2
14	Paul Kallan	NRO/DNRL/NMIP
15	Royce Beacom	NRO/DE/ICE1
16	Tony Attard	NRR/DSS/SNPB
17	Jeff Ciocco	NRO
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
563rd FULL COMMITTEE MEETING

June 3-5, 2009

PLEASE PRINT

TODAY'S DATE: June 3, 2009

	NAME	AFFILIATION
1	Tom Myers	NIST
2	Rob Dimeo	NIST
3	DAVID BROWN	NIST
4	MIKE ROWE	NIST
5	Robert Williams	NIST
6	WADE Richards	NIST
7	PAUL BRAND	NIST
8	GEORGE OLIVER	NEI
9	Douglas Pruitt	AREVA
10	Chris Hoffman	PPL
11	Anthony Giancatarino	Exelon
12	JASON HARTZELL	PPL
13	RONNIE Gardner	AREVA
14	GORDON CLEFTON	NEI
15	Tom Eichenberg	TVA
16	DAVE BORTZ	Duke
17	Robert Montgomery	ANATECH/EPRI
18	Nayem Jahingir	GNF
19	Bert Dunn	AREVA
20	RUSS FAWCETT	GNF
21	William Slagle	Westinghouse
22	Thomas Lindqvist	Westinghouse
23	KENJI IMASHIO	MHI
24	Ryan Sprengel	MNES
25	MASANORI ONOZUKA	MNES
26		
27		
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
563rd FULL COMMITTEE MEETING

June 3-5, 2009

PLEASE PRINT

TODAY'S DATE: June 4, 2009

	NAME	NRC ORGANIZATION
1	Jocelyn Mitchell	RES
2	Steven A. Laur	NRR/DRA
3	Bill Ruland	NRR/DSS
4	Steve Smith	NRR/DSS
5	Sanaa Wasten	NSIR/DSP
6	Kimyata Morgan Butler	FSME/DILR
7	Mike Waterman	RES/DE/DICB
8	Terry Jackson	NRO/DE/ICE1
9	Royce Beacom	NRO/DE/ICE1
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
563rd FULL COMMITTEE MEETING

June 3-5, 2009

PLEASE PRINT

TODAY'S DATE: June 4, 2009

	NAME	AFFILIATION
1	J D Wolcott	TVA
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Revised June 18, 2009

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

June 18, 2009

AGENDA
564th ACRS MEETING
JULY 8-10, 2009

WEDNESDAY, JULY 8, 2009, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 – 10:00 A.M. License Renewal Application and the Final Safety Evaluation Report (SER) for the Beaver Valley Power Station (Open) (DCB/CLB)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and FirstEnergy Nuclear Operating Company regarding the License Renewal Application for the Beaver Valley Power Station, the associated NRC staff's final SER, and related matters.

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

10:00 – 10:15 A.M. * BREAK *****

3) 10:15 – 11:45 A.M. Draft Final Revision 3 to Regulatory Guide 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants" (Open) (JWS/MPL)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding Draft Final Revision 3 to Regulatory Guide 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," and related matters.

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

11:45 – 12:45 P.M. * LUNCH *****

4) 12:45 – 2:45 P.M. Applicability of TRACE Code to Evaluate New Light Water Reactor (LWR) Designs (Open/Closed) (SB/HPN/DEB)
4.1) Remarks by the Subcommittee Chairman
4.2) Briefing by and discussions with representatives of the NRC staff regarding applicability of the TRACE Code to evaluate new LWR designs, and related matters.

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

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

2:45 – 3:00 P.M. * BREAK *****

5) 3:00 – 4:00 P.M. Format and Content of the Biennial Research Report to the Commission on the NRC Safety Research Program (Open) (DAP/HPN)
5.1) Remarks by the Subcommittee Chairman
5.2) Discussion of the Format and Content of the ACRS Biennial Report to the Commission on its review and evaluation of the NRC Safety Research Program, and related matters.

4:00 – 4:15 P.M. * BREAK *****

6) 4:15 – 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
6.1) License Renewal Application and the Final Safety Evaluation Report for the Beaver Valley Power Station (DCB/CLB)
6.2) Draft Final Revision 3 to Regulatory Guide 1.100, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants” (JWS/MPL)
6.3) Applicability of TRACE Code to Evaluate New Light Water Reactor (LWR) Designs (SB/HPN/DEB)

THURSDAY, JULY 9, 2009, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

8) 8:35 – 10:30 A.M. Design Certification (DC)/Combined License (COL) Interim Staff Guidance (ISG) -006 and Nuclear Energy Institute (NEI) document NEI 08-08, Revision 1 (Open) (MTR/DAW)
8.1) Remarks by the Subcommittee Chairman
8.2) Briefing by and discussions with representatives of the NRC staff regarding DC/COL-ISG-006, “Interim Staff

Guidance on Evaluation and Acceptance Criteria for 10 CFR 20.1406 to Support Design Certification and Combined License Applications," and NEI 08-08, Revision 1, "Generic FSAR Template Guidance for Life Cycle Minimization of Contamination," Contamination," and related matters.

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

10:30 – 10:45 A.M. * BREAK *****

9) 10:45 – 12:15 P.M. Draft Final Regulatory Guide 1.215, "Guidance for ITAAC Closure under 10 CFR Part 52" (Open) (DCB/MLC/GSS)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussions with representatives of the NRC staff regarding Draft Final Regulatory Guide 1.215 that provides guidance for closure of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) under 10 CFR Part 52, and related matters.

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

12:15 – 1:15 P.M. * LUNCH *****

10) 1:15 – 1:45 P.M. Quality Assessment of Selected Research Projects (Open) (DAP/HPN)

10.1) Remarks by the Subcommittee Chairman

10.2) Discussion of the draft final report on ACRS assessment of the quality of the NRC research projects on: NUREG/CR-6964, "Crack Growth Rates and Metallographic Examinations of Alloy 600 and Alloy 82/182 from Field and Laboratory Materials Testing in PWR Environments," and Draft NUREG/CR-XXXX, "Diversity Strategies for Nuclear Power Plant Instrumentation and Control Systems."

11) 1:45 – 2:15 P.M. Subcommittee Reports (Open)

11.1) Report by and discussion with the Chairman of the ESBWR Subcommittee regarding review of the resolution of containment issues associated with the ESBWR design certification; and selected Chapters of the draft SER associated with the North Anna COL application referencing the ESBWR design, that were discussed on June 17-18, 2009. (MLC/CLB)

11.2) Report by and discussion with the Chairman of the Plant License Renewal Subcommittee regarding interim review of the Prairie Island License Renewal Application and the SER with Open Items, that were discussed on July 7, 2009. (HBR/CLB)

12) 2:15 – 3:00 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed) (MVB/EMH)

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

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

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

13) 3:00 – 3:15 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (MVB/CS/AFD)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

3:15 – 3:30 P.M. * BREAK *****

14) 3:30 – 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

14.1) License Renewal Application and the Final Safety Evaluation Report for the Beaver Valley Power Station (DCB/CLB)

14.2 Draft Final Revision 3 to Regulatory Guide 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants" (JWS/MPL)

14.3) Applicability of TRACE Code to Evaluate New Light Water Reactor (LWR) Designs (SB/HPN/DEB)

14.4) DC/COL-ISG-006 and NEI 08-08, Revision 1, for meeting requirements of 10 CFR 20.1406, "Minimization of Contamination" (MTR/DAW)

14.5) Draft Final Regulatory Guide 1.215, "Guidance for ITAAC Closure under 10 CFR Part 52" (DCB/MLC/GSS)

FRIDAY, JULY 10, 2009, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

15) 8:30 – 6:00 P.M. **(12:00-1:00 P.M. LUNCH)**
Preparation of ACRS Reports (Open)
Continue discussion of the proposed ACRS reports listed under Item 14.

16) 6:00 – 6:30 P.M.
Miscellaneous (Open) (MVB/EMH)
Discussion of matters related to the conduct of Committee activities and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTES:

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

Appendix V

LIST OF DOCUMENTS FROM THE
563RD ACRS MEETING JUNE 3-5, 2009

Agenda Item 2:
License Renewal Application and the Revised Final Safety Evaluation Report for the National Institute of Standards and Technology (NIST) Reactor

1. Proposed Schedule
2. Summary Report
3. Attachments:
   - Letter from Dr. Wade Richards (NIST) to NRC/Document Control Desk: Response to ACRS License Renewal Subcommittee Meeting Follow-up Items, "RAI (TAC No. MD3410)," dated 3/19/2009
   - Memo from Dr. Wade Richards (NIST) to William Kennedy (NRC): Updated Loss-of-Offsite-Power Accident Analysis – "Response to ACRS Question (April 2, 2009 Meeting)," dated 4/22/2009
   - NIST License Renewal Subcommittee Meeting Minutes (2/4/2009)
   - NIST License Renewal Application, dated April 9, 2004
   - Application Supplements (RAIs and Responses)
   - NRC staff's final SER, dated May, 2009
   - NIST Technical Specifications, dated May, 2009
   - Non-Power Reactor Standard Review Plan: NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," dated February 1996

Agenda Item 3:
Draft Final Regulatory Guides 1.21 and 4.1

4. Proposed Schedule
5. Status Report

Agenda Item 4:
Pellet-Clad Interaction Failures under Extended Power Uprate Conditions

6. Agenda
7. Status Report
8. Attachments
   - Letter from William J. Shack, "Susquehanna Steam Electric Station Units 1 and 2 Extended Power Uprate Application"
   - Added comments on Susquehanna letter
   - Memorandum from M. W. Libarkin, "Request for Information on Pellet-Clad Interaction"
   - Memorandum from Michael Tokar, transmitting "Report to ACRS Concerning NRR Efforts on Pellet/Cladding Interaction"
   - Slides from March 3, 2009 Materials, Metallurgy, and Reactor Fuels Subcommittee meeting on PCI fuel failures

Appendix IV

LIST OF DOCUMENTS FROM THE
563RD ACRS MEETING JUNE 3-5, 2009

Agenda Item 5:
Diversity and Defense-in-Depth Topical Report Associated with the US-APWR Design

9. Proposed Schedule
10. Summary Reports
11. Attachments
    - NRC Safety Evaluation Report - Topical Report on Defense-in-Depth & Diversity
    - US-APWR Design Certification Application-Specific Action Items

NIST Response to Open Item

ACRS Meeting

June 3, 2009

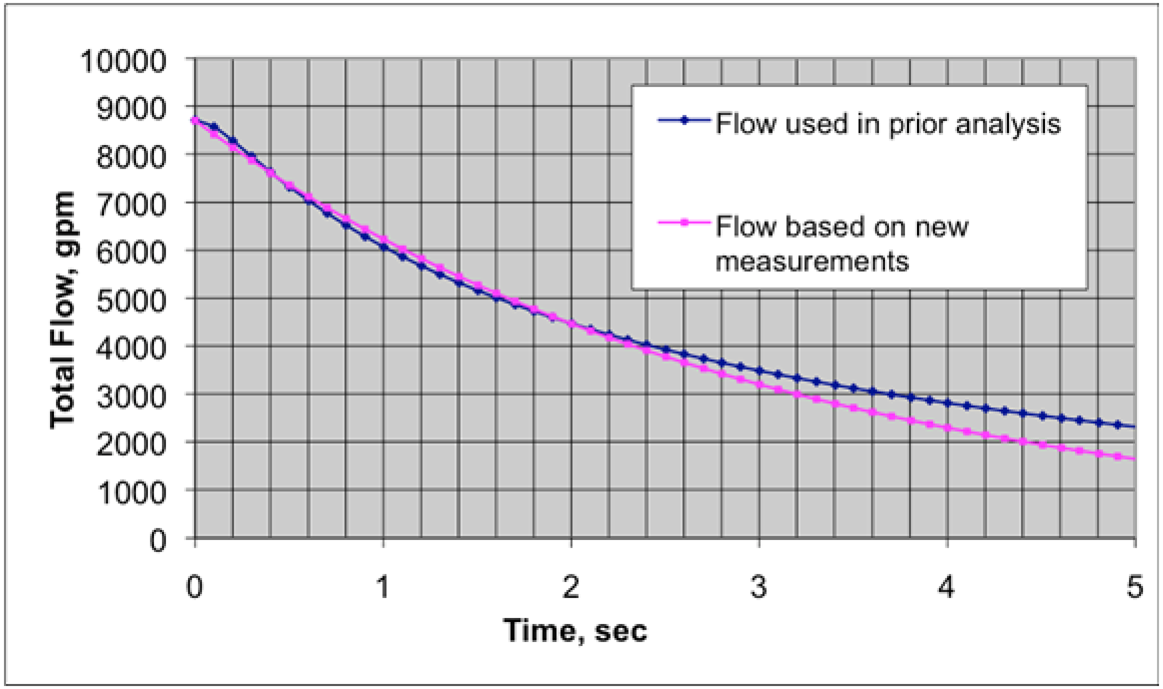
NIST PARTICIPANTS

- Dr. Robert Dimeo, Director NCNR
- Dr. Wade Richards, Chief ROE
- Dr. Robert Williams, Section Head Nuclear Analysis
- Dr. Mike Rowe, Special Advisor to NCNR Director
- Mr. Thomas Myers, Chief Reactor Ops.
- Mr. David Brown, Supervisor Health Physics

Open Item From Meeting of 4/3/2009

- While responding to a question raised at an earlier ACRS Subcommittee meeting, NIST identified an issue with pump coast-down
- It was noted that the pump coast-down curve used for the RELAP analysis was compared to the data measured under different conditions
- Although the curve used in the analysis was very conservative, a new curve was measured under appropriate conditions for comparison

Results



Comparison between prior flow model and new model, which was conservatively based upon new measurement

RELAP ANALYSES

- The minimum CHFR occurs at approximately 1.5 s, where the two curves coincide
- As a result, the MCHFR of 2.17 is unchanged within error from the earlier value of 2.19
- Detailed analyses out to 30 s show that the system progresses to a stable natural circulation state
- The fuel temperature remains below 137°C, substantially below the safety limit

Conclusion

- The limiting loss of flow accident (with failure of shutdown pumps to start) has been extensively re-analyzed
- The results show a substantial margin against DNB (MCHFR > 2)
- This accident poses no danger of fuel damage
- The SAR will be updated to reflect this revised analysis

U.S.NRC

UNITED STATES NUCLEAR REGULATORY COMMISSION

Protecting People and the Environment

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

National Institute of Standards and Technology National Bureau of Standards Test Reactor License Renewal

June 3, 2009

William B. Kennedy, Project Manager

Office of Nuclear Reactor Regulation

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

Open Item

- In addressing the concerns of the ACRS subcommittee members, the licensee identified an unrelated error in a measured flow coastdown data set
- The flow coastdown data set was used to benchmark the RELAP model used to analyze the loss-of-offsite power accident
- The licensee promptly reported the error on March 30, 2009, to the NRC project manager

Initial NRC Response

- The staff performed a preliminary independent review and calculation to assess the safety significance of the error
  - safety margin reduced, but still adequate
  - isolated error
  - staff's calculation in close agreement with licensee's initial assessment
- The staff discussed the significance of the error with the licensee and a plan to update the flow coastdown data set and the accident analysis

NRC Staff Review

- The licensee submitted a revised loss-of-offsite-power accident analysis on April 22, 2009
- The staff compared the updated flow coastdown data set against the erroneous data set and found them to be nearly identical
- The staff reviewed the assumptions used in the updated accident analysis and found them to be as conservative as those used in the original analysis

Updated Safety Evaluation

- The minimum critical heat flux ratio (safety margin) at the hot spot on the fuel cladding decreased from 2.19 to 2.17
- The maximum fuel temperature is 137 degrees Celsius (the safety limit is 450 degrees Celsius)
- The staff concludes that there is reasonable assurance that a loss of offsite power will not result in fuel damage and that the consequences of the accident are bounded by the MHA

Revision of RG 1.21 (Effluents) and RG 4.1 (Environmental Monitoring)

Presentation for:

ACRS Committee Meeting

June 3, 2009

Richard Conatser & Steve Garry
NRR Div. Inspection & Regional Support

Outline

- Introduction (People & Project)
- History (Drivers for Change)
- Documents
- Reg Guide Update Initiative
- Reasons for revising RGs
- Considerations: Backfit, Consistency, Delay Publication
- Closure and Questions

Introduction: Project & People

- Team formed in 2006
  - HQ: NRR, NRO, FSME, RES
  - Regions: I-IV
  - Some are here today
- Progress
  - FRN Oct and Nov 2008
  - Public Meeting in January
  - Office Concurrence and ACRS Sub.: May
  - OGC and ACRS

History (Drivers for Change)

- H-3 in Ground Water
  - Salem – 2003, SFP Leak
  - Braidwood – Mar 2005, H-3 in Well
  - Indian Point – Sep 2005, Crack in SFP
- Lessons Learned Task Force Report
  - Sep-2006, Total of 26 Recommendations
  - 10 Recommendations → RG 1.21
  - 4 Recommendations → RG 4.1

Documents

- RG 1.21 (Effluents)
  - Measuring, Evaluating, Reporting Effluents
  - Abnormal Releases, C-14, Sampling, Surveys, Principal Nuclides, LLD
- RG 4.1 (Environmental)
  - Monitoring Radioactivity in the Environs
  - Exposure Pathways, Routes of Exposure, Samples, Spills, Reports

RG Updates

- 476 Reg Guides to Revise
- NRC Chairman Memo, Jun-2006
- Phases 1 thru 3, ECD Dec-2009
- RG 1.21 and RG 4.1 are in Phase 3

Benefits of Revising RGs

- RG Update
- Lessons Learned Task Force Rec.
  - Ground Water Issues (Surveys, etc)
- Dated Guidance (RGs 35 years old)
- Incorporate OE & Lessons Learned
  - TEDE, Direct Rad, C-14, LLD, etc
- NEI, EPRI, ANI issued new guidance
- Updated NRC guidance is needed

Public Comment: Back-fit

- RGs are not regulations
- RGs describe acceptable methods
- Licensees may continue to use Rev. 1
- Licensees are not required to commit

Public Comment: “Inconsistencies”

- Discussed at ACRS Subcommittee
- NUREG-1301 and 10 CFR 50
  - Semi-annual vs Annual Reports
- 10 CFR 20 and 10 CFR 50
  - TEDE vs Whole Body Dose
- NUREG-1301 and RG 1.21
  - NUREG silent C-14, RG includes C-14
- NUREG-0543 and RG 1.21
  - Calculating EPAs 40 CFR 190 Dose

Public Comment: Delay RGs

- Discussed possibility at ACRS Sub.
- ICRP-103 dose methodology pending
  - SECY-08-197
  - Engage Stakeholders
  - May take many years to complete
- Plants not required to commit to RG 1.21

→ Staff Recommendation: Issue RGs consistent with RG Update Initiative

Questions

?

Regulatory Guides 1.21 & 4.1 Issues (DG 1186 & DG 4013)

George Oliver
June 2009
ACRS

NEI
NUCLEAR ENERGY INSTITUTE

DG-1186 & DG-4013 Issues
Industry & Staff Efforts

- Industry Contribution From 30+ Individuals
- Many Detailed Technical Comments
- Professional & Productive Relationship With Staff
  – January 15, 2009 Workshop Productive
- Emergence Of SECY 08-0197
  – 40 Guidance Documents Impacted
  - An Integrated Approach Is Needed

NEI

DG-1186 & DG-4013 Issues
Need For Integrated Approach

- **DG-1186 & DG-4013 Duplicate & Inconsistent With Other Guidance**
  - **Several Guidance Documents Related To Groundwater**
- **SECY 08-0197 Offers A Real Opportunity**
  - **Benefits Of Consolidated Guidance**

NEI

DG-1186 & DG-4013 Opportunities

- The Existing Guidance Should Remain Applicable
  - The Licensing Basis Is Not Impacted
- Clarification Of Solid Radioactive Waste Reporting
- Elimination of On Site Radiological Monitoring Programs From DG-4014
- Additional Flexibility
  - Calculate C-14 Effluents

NEI

Meeting Objective

- Assess the risk of PCI/SCC fuel failures during BWR Anticipated Operational Occurrences at EPU conditions.
- ACRS Letter, Dec. 20, 2007 Susquehanna Extended Power Uprate

  ”The staff should develop the capability and perform a thorough review and assessment of the risk of Pellet-Cladding Interaction (PCI) fuel failures with conventional fuel cladding during anticipated operational occurrences.”

Topics

- Background and Reasons for Concern
- PCI Basics
  - Power/burnup dependence
  - Appearance, Mechanism
- PCI failure powers, failure strains and times-to-failure.
- Conclusions and Recommendations

Background

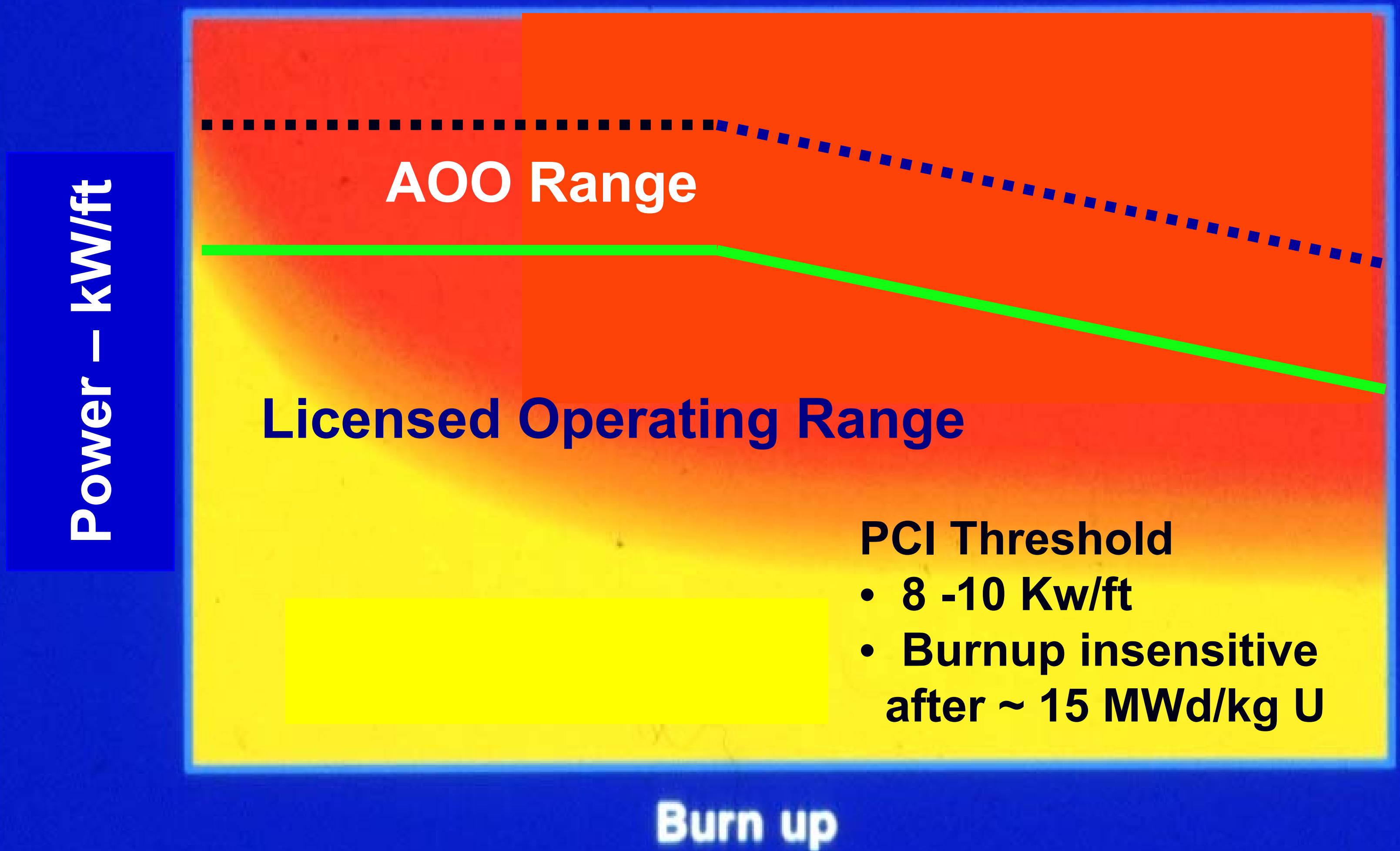
- NRC analyses in late 1970s early 80s
  – Notified vendors ...ready to introduce PCI fuel failure analyses into plant safety analyses.
- PCI mitigations introduced in late 70s, early 80s
  – Operating restrictions, 9x9 and 10x10 fuel bundles introduced
  – PCI resistant design licensed and in service.
- Technical assumptions:
  – existing thermal-mechanical licensing limits were sufficient to prevent PCI during AOOs
  – transients were over too quickly to cause PCI failures during AOOs

No incentive for PCI-specific regulatory changes

Reasons for Current Concern

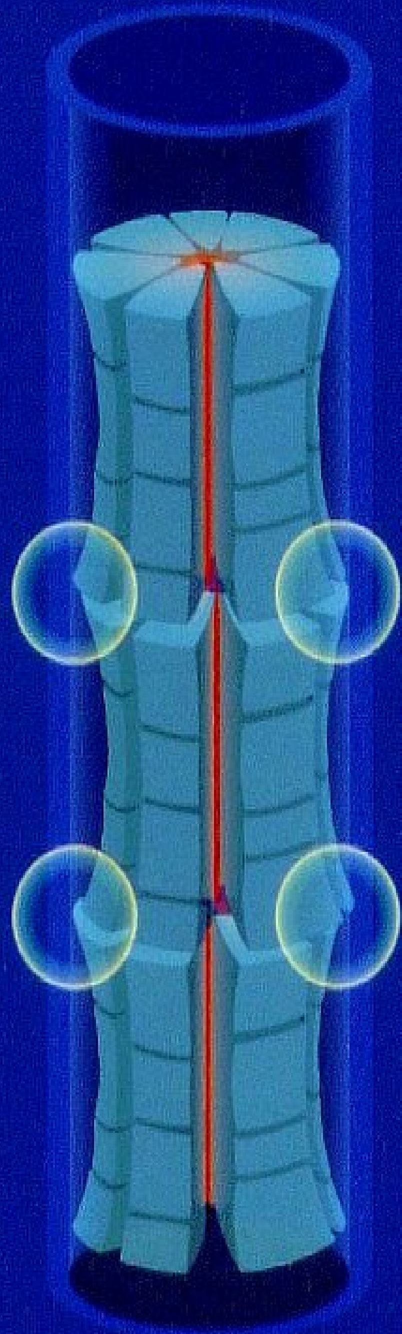
- Margins gained by design changes introduced in the1980s are disappearing.
  - Peak LHGRs of today's 10x10 fuel designs are the same as old 8x8 designs.
  - Number of fuel rods at risk during AOOs increasing in proportion to magnitude of EPU.
  - Use of non-PCI-resistant fuel increasing
- PCI failure strains are much lower than the <1% strain acceptance criteria.
- PCI failure times are very short at AOO power levels.

PCI-MAP



Power – kW/ft
AOO Range
Licensed Operating Range
PCI Threshold
• 8 -10 Kw/ft
• Burnup insensitive after ~ 15 MWd/kg U
Burn up

PCI FAILURE MECHANISM

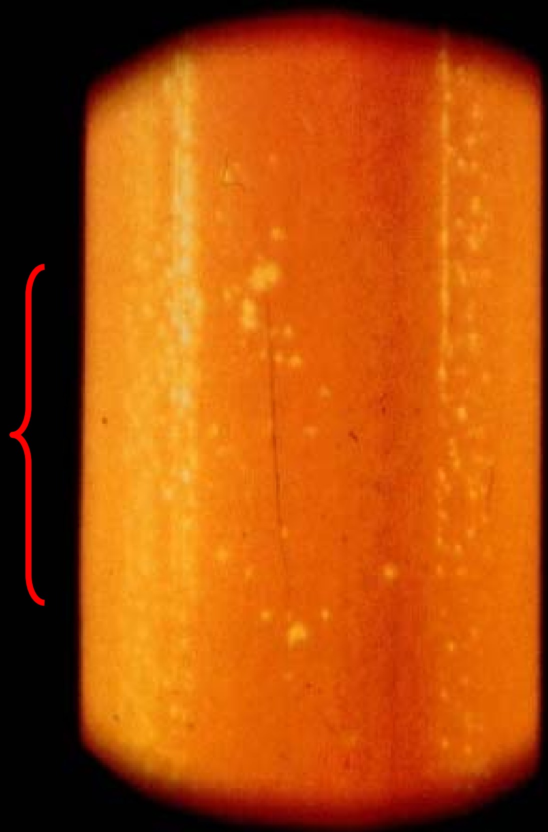


Maximum
- Biaxial Stresses
- Iodine, Cadmium

Axial Locking of Fuel Column
- Thermal Expansion
- Stochastic Stacking

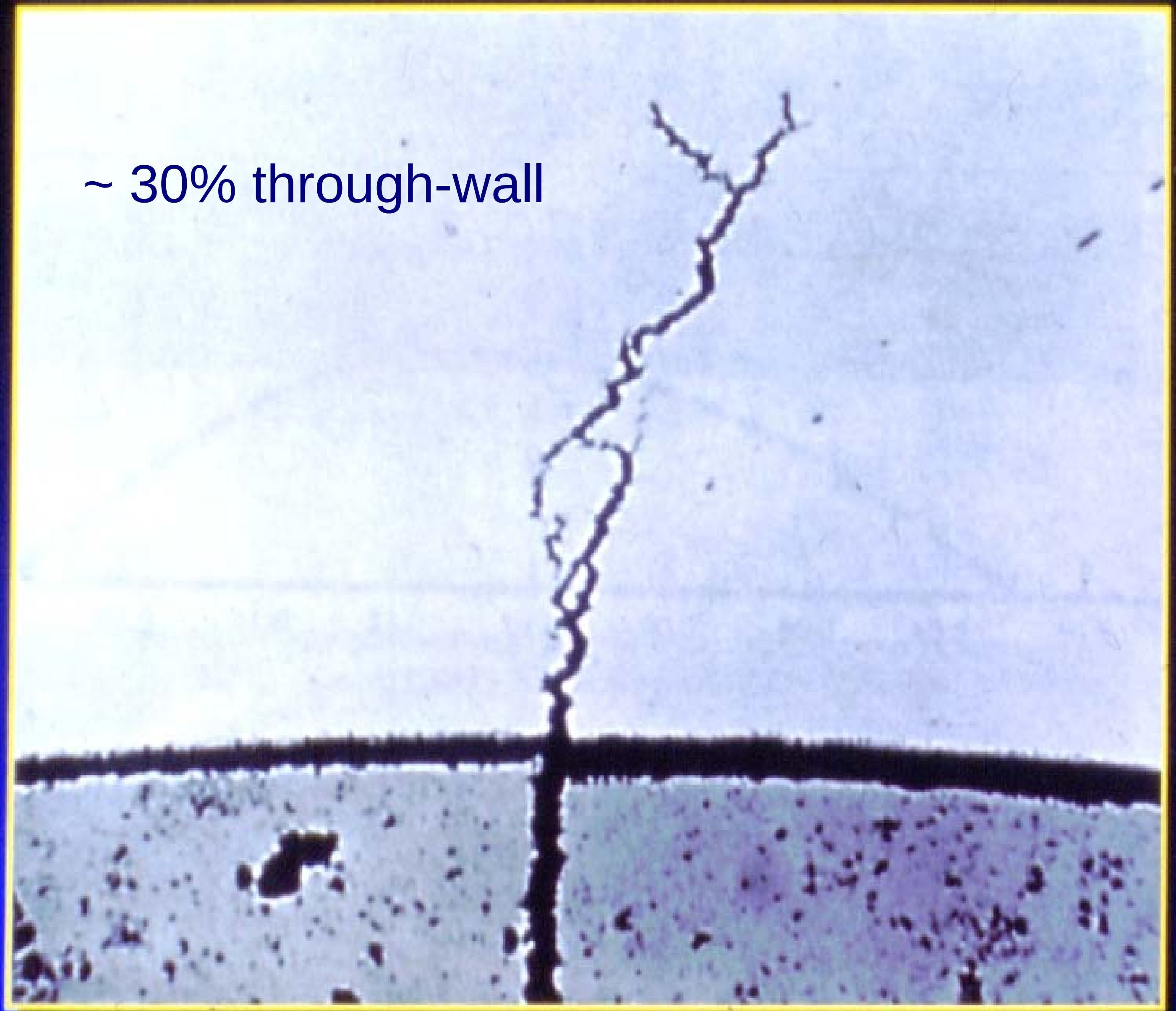
SB398.20

PELLET CLAD INTERACTION CRACK

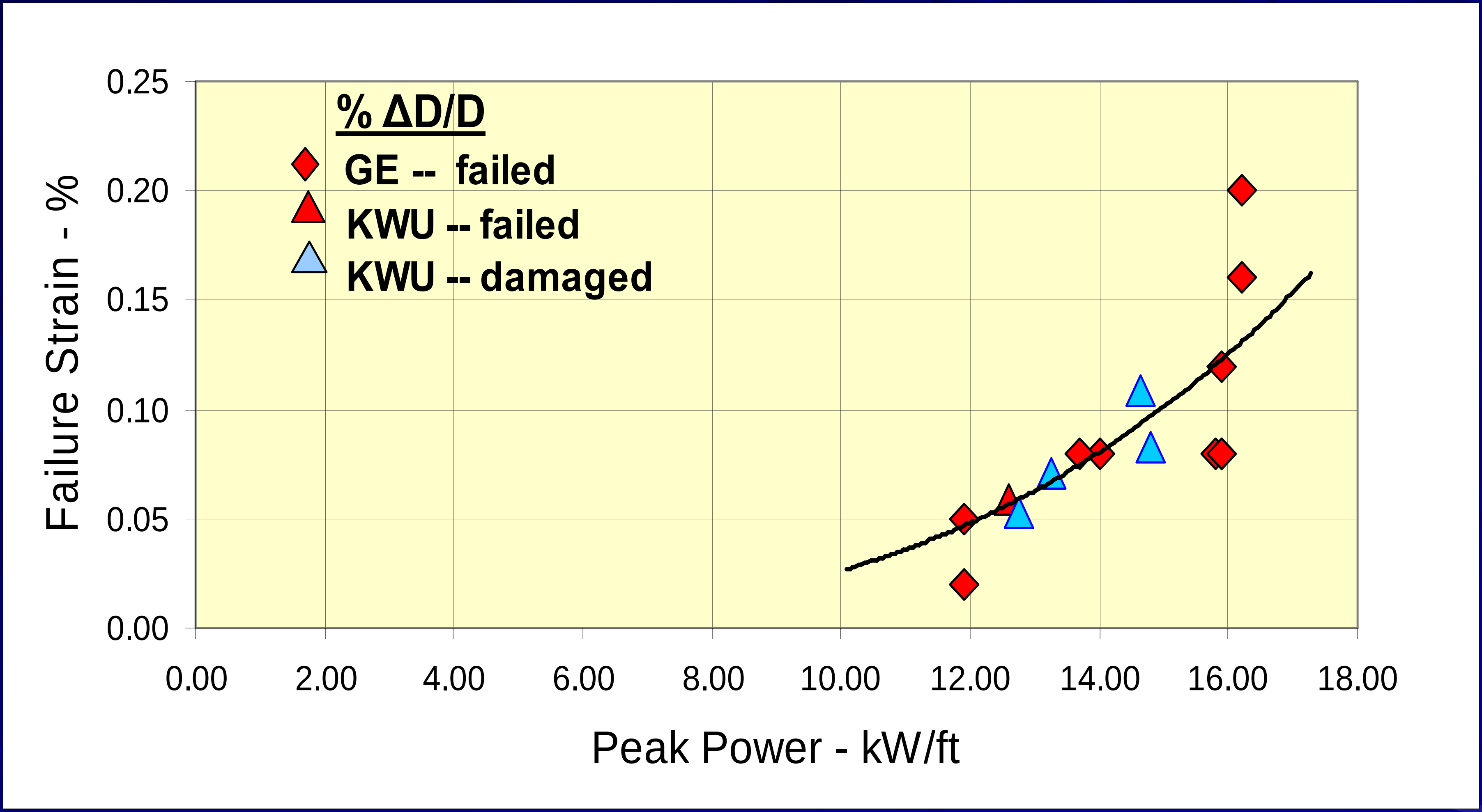


- BWR fuel rod
- Typical axial crack
- << 1% plastic strain

Branching in Incipient PCI Crack

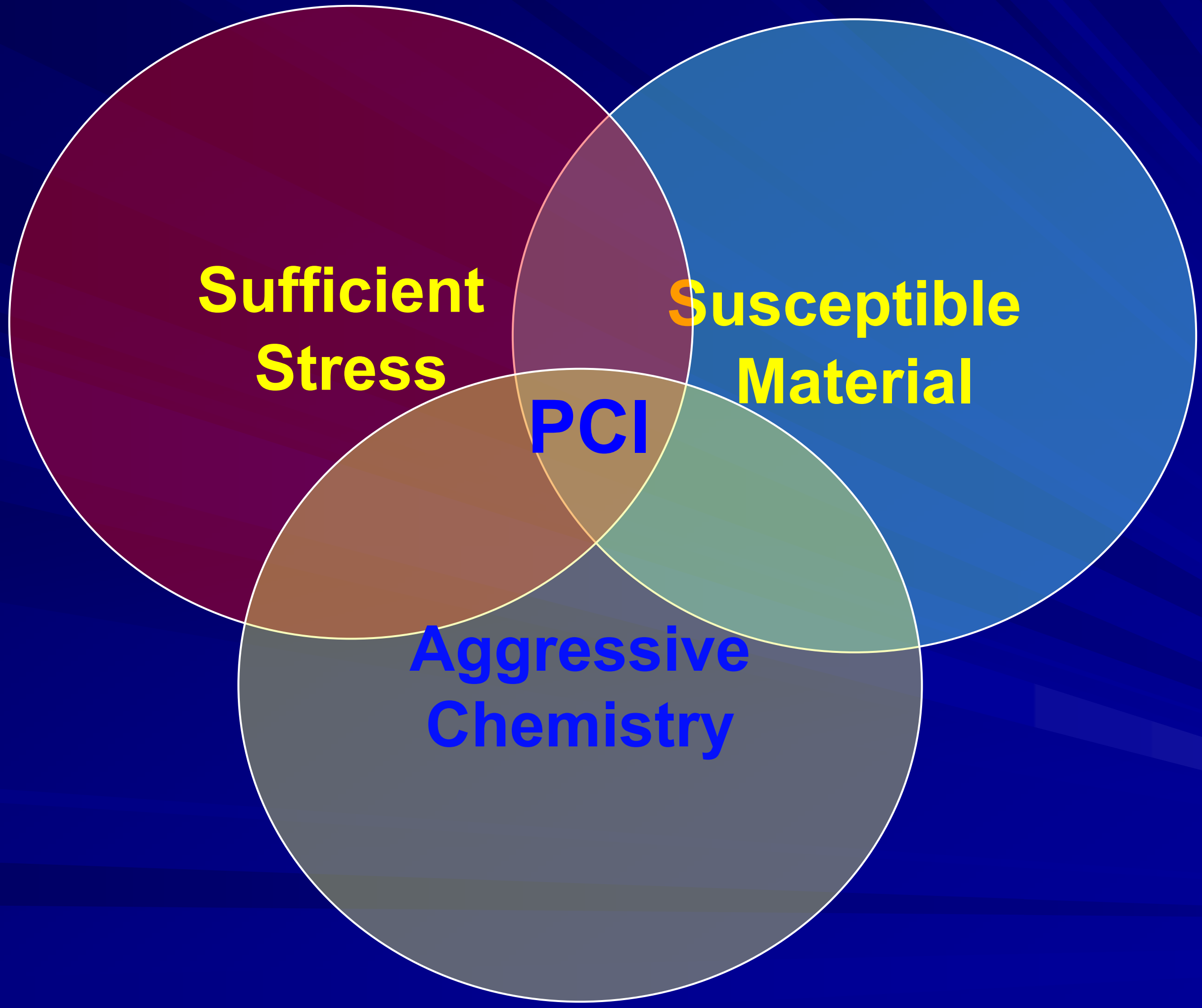


GE and Demo Ramp II PCI Failure Strains

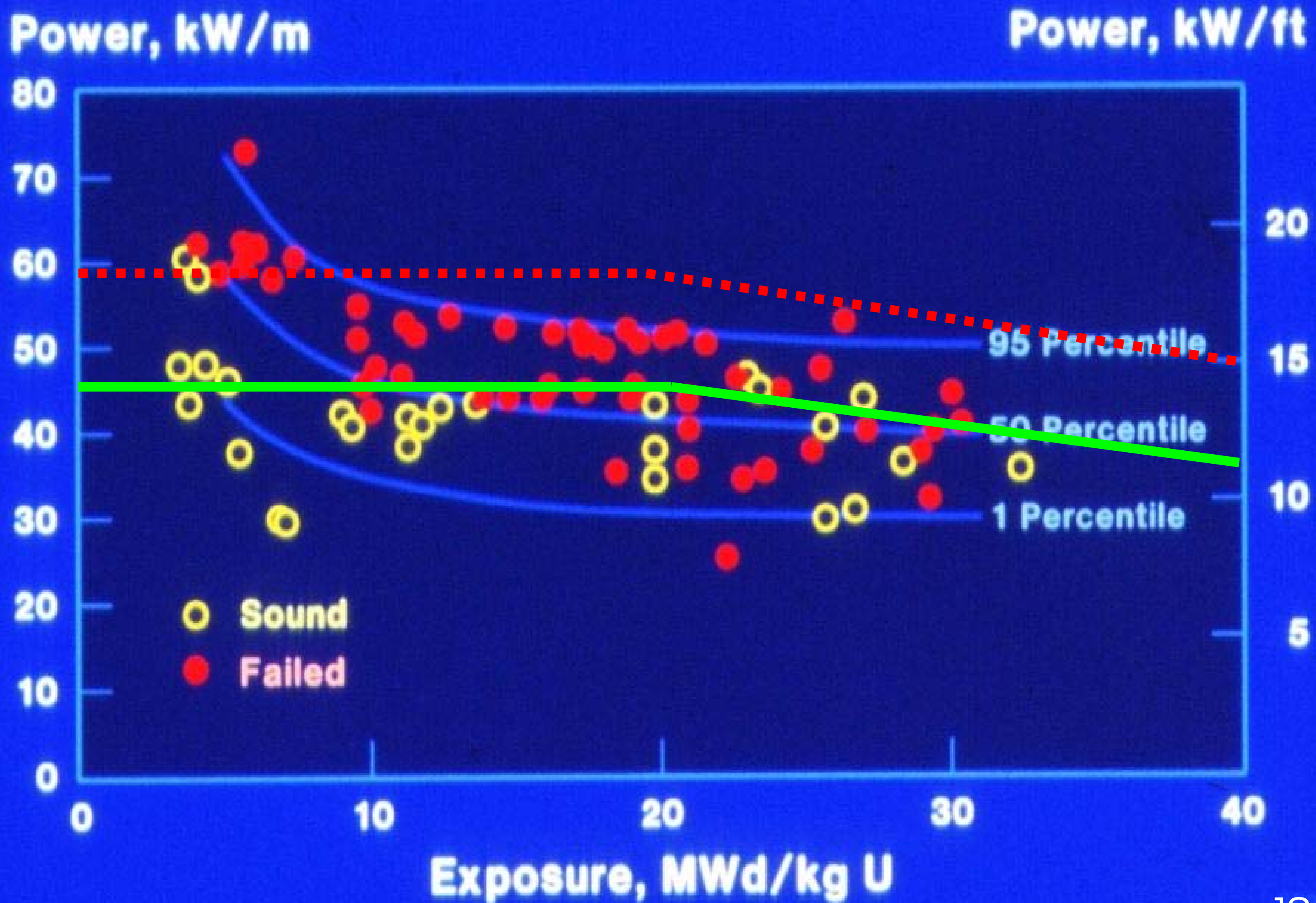


- All rods failed or damaged by PCI
- All strains much lower than 1%

Requirements for Stress Corrosion Cracking



BWR STANDARD FUEL



Oskarshamm 1 Event

- Control rod withdrawal test in Oskarshamm 1 BWR in 1975
  - Performed by ASEA-ATOM to demonstrate PCI resistance of standard 8x8 Zr-2 fuel cladding.
  - Single control blade withdrawn in 10% steps with 2 hour holds.
- Peak powers at failure nodes ranged from 9.1 to 11.3 kW/ft.
- 45 fuel rods in 14 bundles failed by PCI.

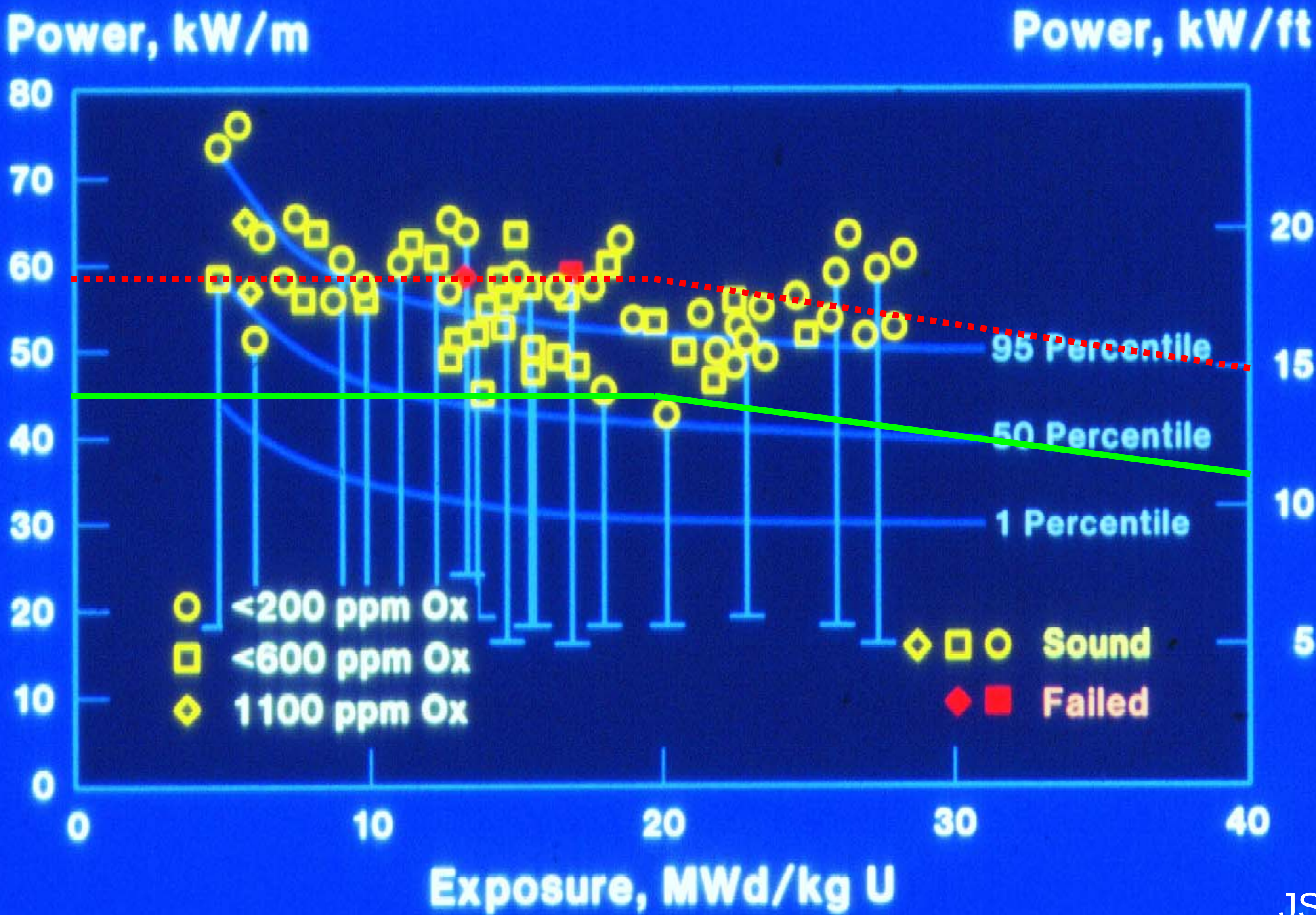
BWR Loss of Feedwater Heater

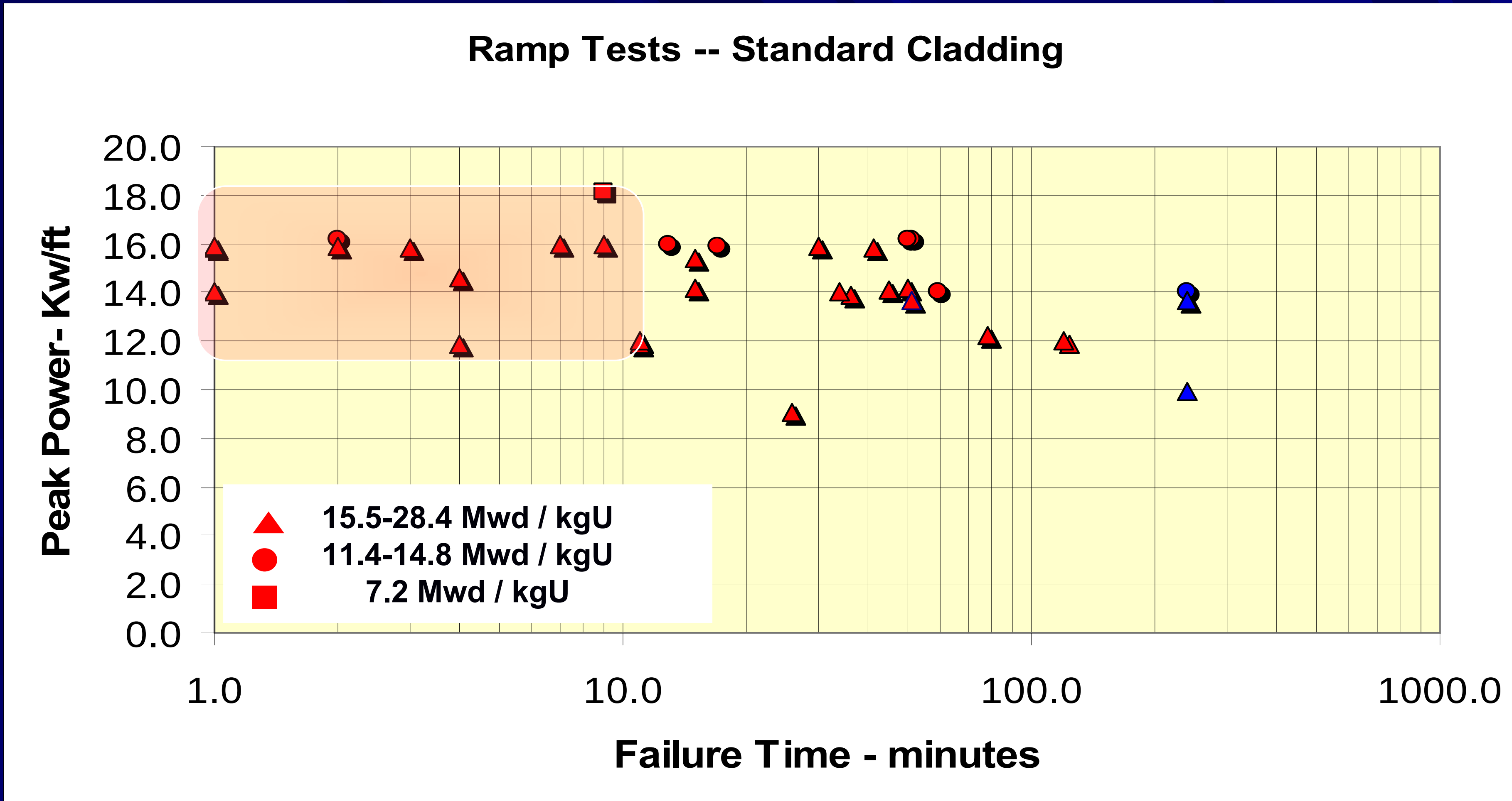
- Loss of feedwater heating results in core power increase due to core inlet subcooling.
- Most severe if feedwater heaters are bypassed
- Core power increases to ~120% of rated in one minute and is maintained until terminated by operator action.
- All fuel rods in the core are affected; peak rods can reach powers up to16 kW/ft depending on fuel design and plant state.

PCI Mitigation Options

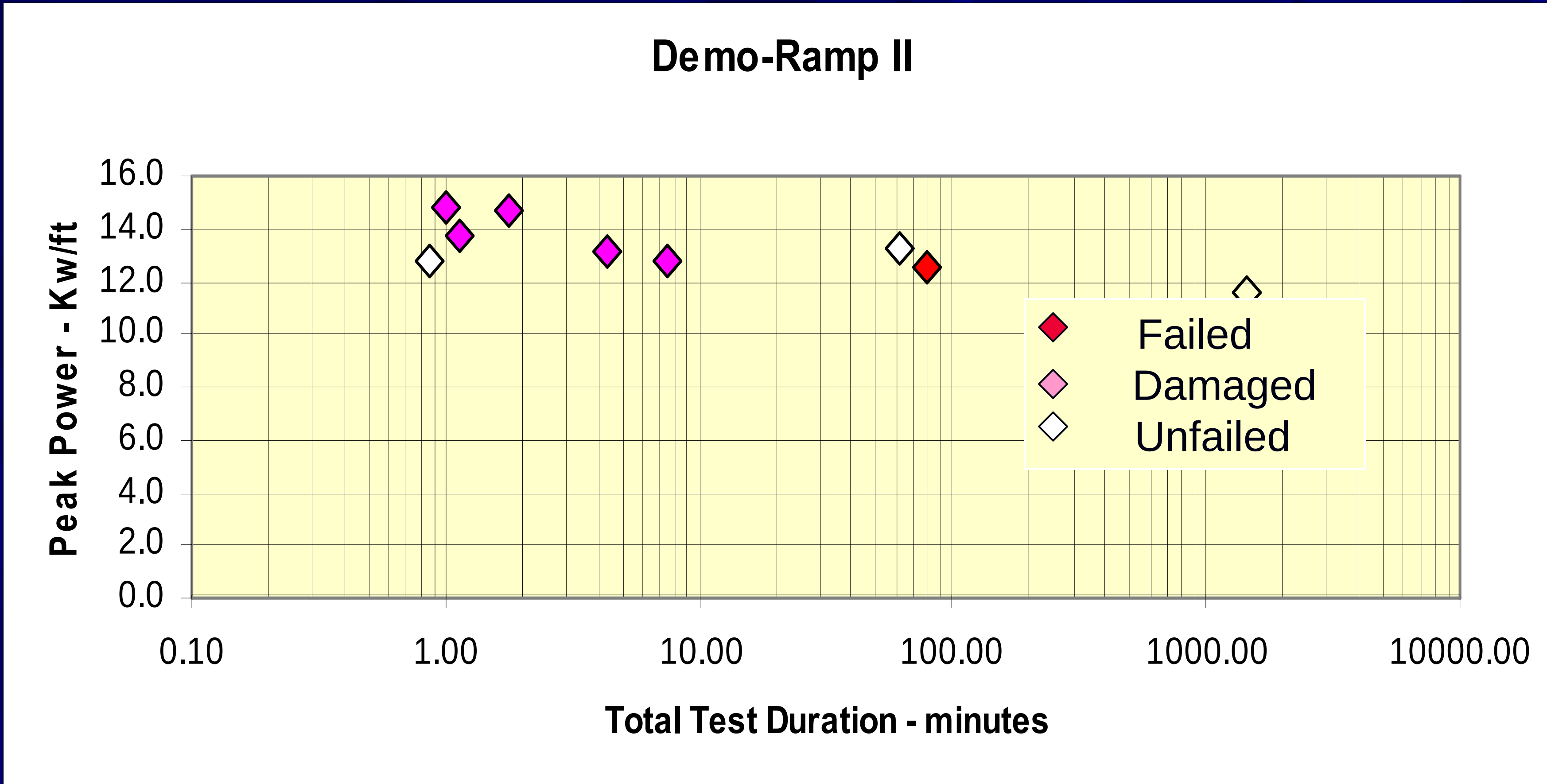
- Normal Operation
  - PCI resistant fuel
  - Preconditioning
- AOOs
  - PCI resistant fuel
  - Prompt operator action

PCI-Resistant Fuel



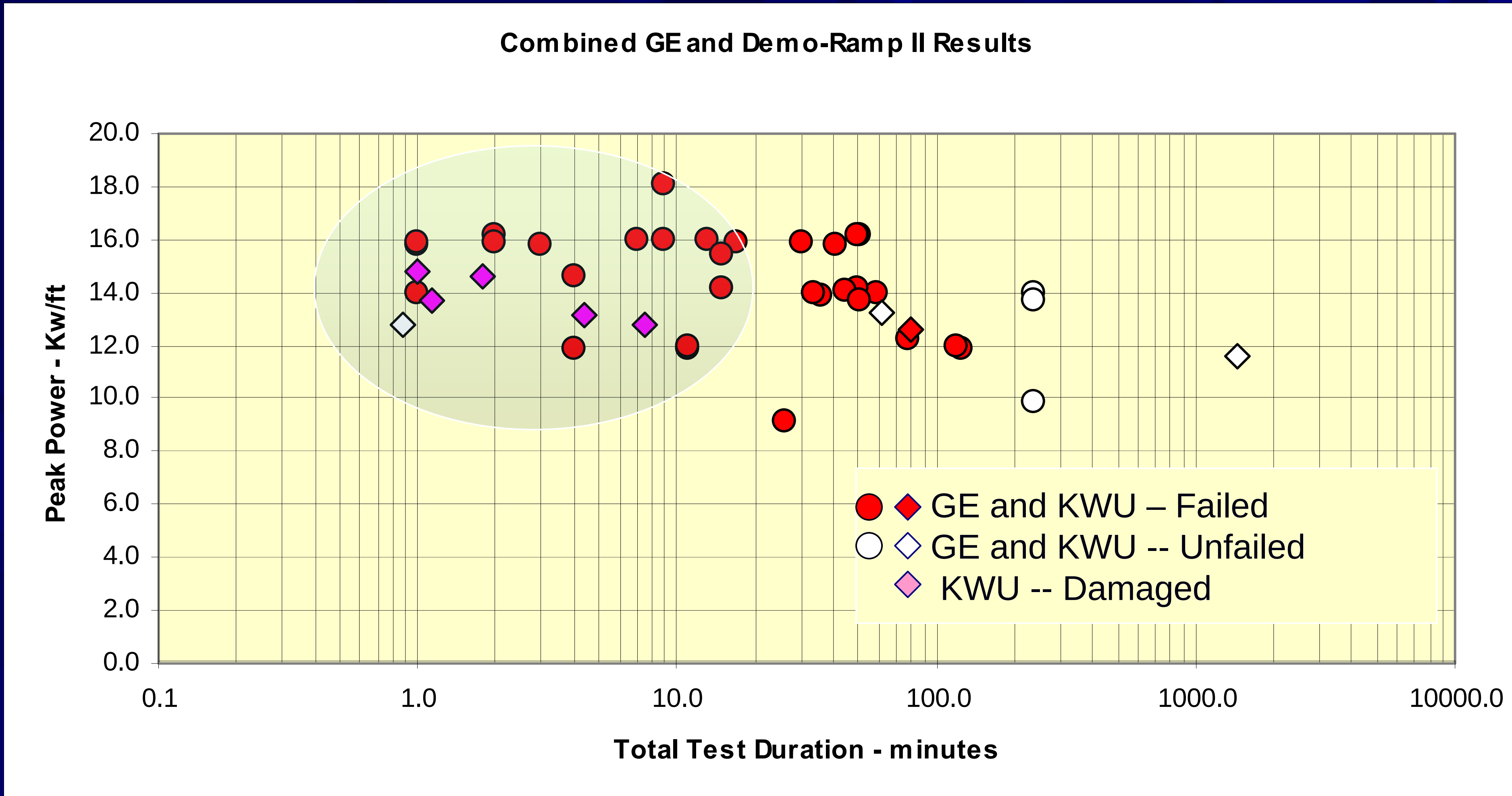


- **GE Fuel Rods**
- **Irradiated in power reactors at low power**
- **Power ramped in R2 reactor**
- **5/25 (19%) failed in 1 to 3 minutes**



KWU Fuel Rods

- Irradiated in power reactors at low power
- Power ramped in R2 test reactor. Ramps intentionally terminated
- Five partial failures -- 10-60% thru-wall, in 1 to 7 minute tests
- One non-failed during 0.87 minute test



Combined Ramp Test Results

- **Performance of GE and KWU test rods consistent**
- **Of the 36 rods tested:**
  - **8 (22%) failed or were damaged within 3 minutes**
  - **1 was not damaged during 0.87 minute test**

Conclusions

- PCI failures are driven by chemistry and stress, not by strain.
- Strains required to cause PCI failures in conventional fuel are much lower that the 1% strain criterion.
- Current T-M regulatory criteria do not protect conventional fuel from PCI failure during AOOs
- PCI crack nucleation and propagation rates are fast enough to cause large numbers of conventional fuel failures during AOOs within one to three minutes.
- The number of fuel rods at risk increases with EPU.

Recommendations

- PCI failure criteria should be based on measured failure powers and failure times, not calculated failure strains.

- PCI resistance of specific fuel designs should be determined by power-ramp testing.

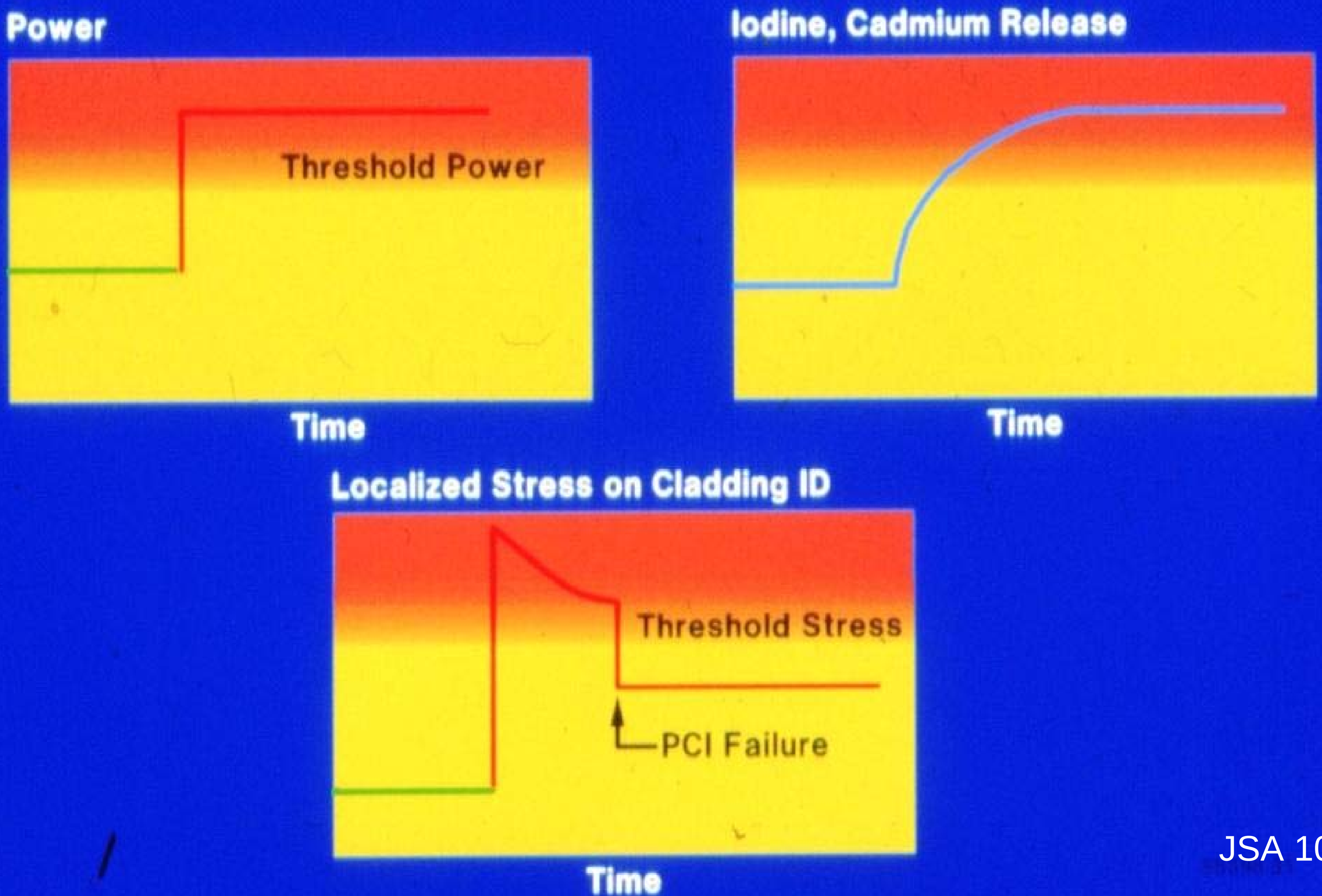
- Failure powers and failure times should be determined from statistically significant numbers of tests performed at conditions (power increase, peak power, time at peak power and burnup) expected during bounding AOOs

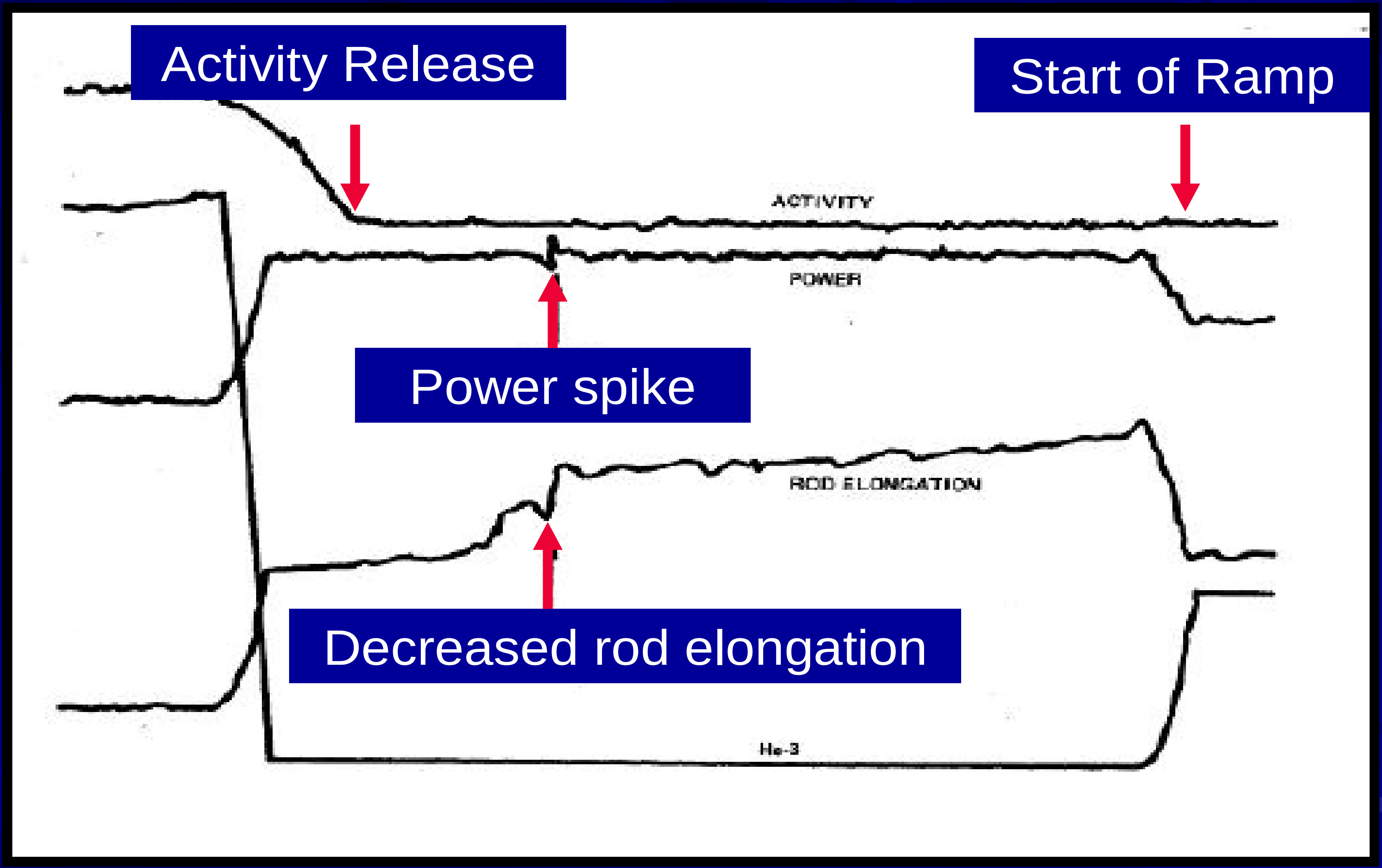
Backups

Tokar report to the ACRS on PCI -1979

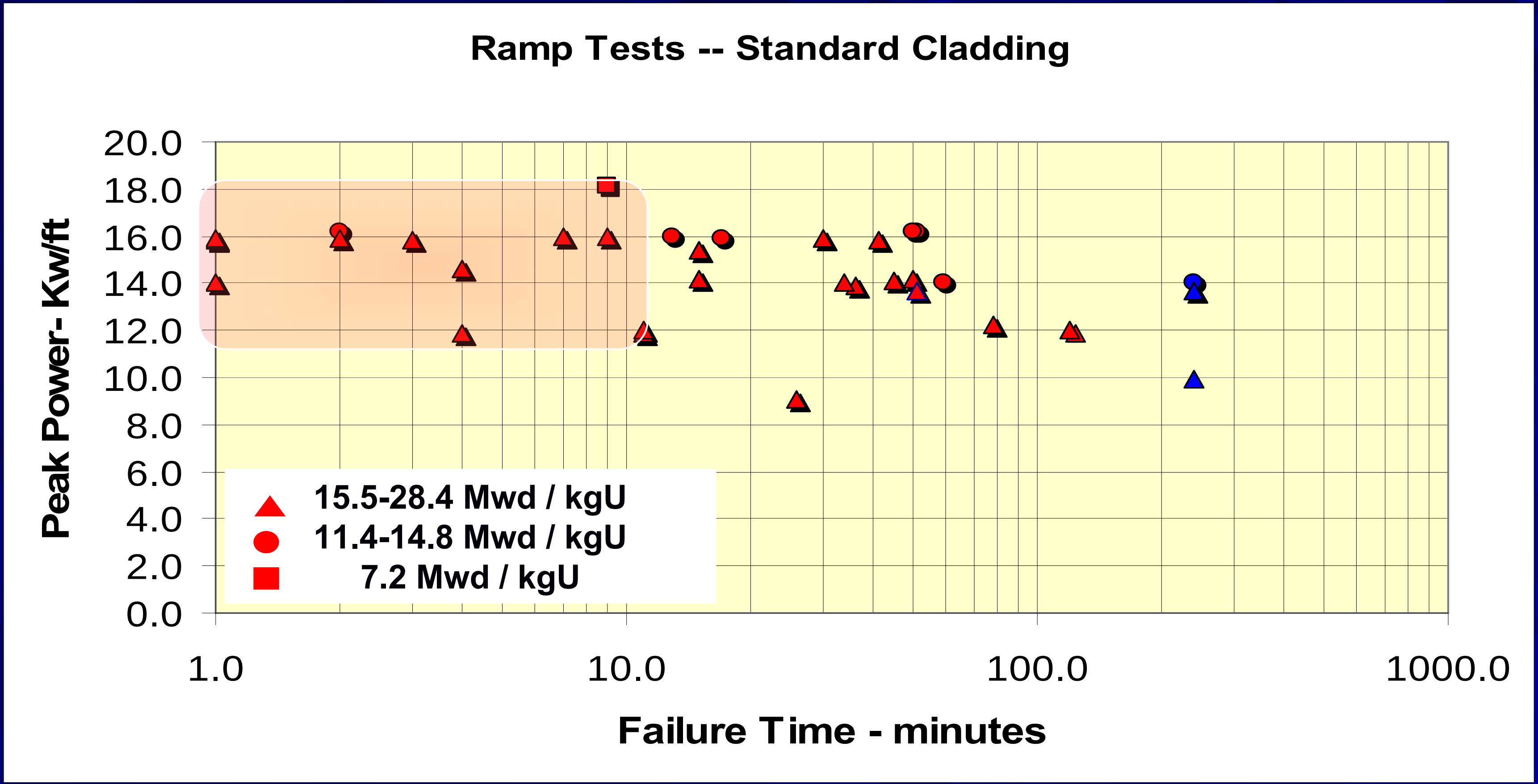
- “Current plant safety analyses are, therefore, deficient in the sense that they do not, in general, account for PCI, which is now well recognized as a significant fuel failure mechanism.”
- “As the result of our past and on-going efforts on PCI, we believe that the time is right to start introducing PCI fuel failure analyses into plant safety analyses.”
- “...a major segment of the LWR industry holds that PCI failures will not occur during the type of power increasing transients and accidents addressed in Chapter 15 of the Standard Review Plan because the time at the increased transient power is too short.”

POWER RAMP EFFECTS



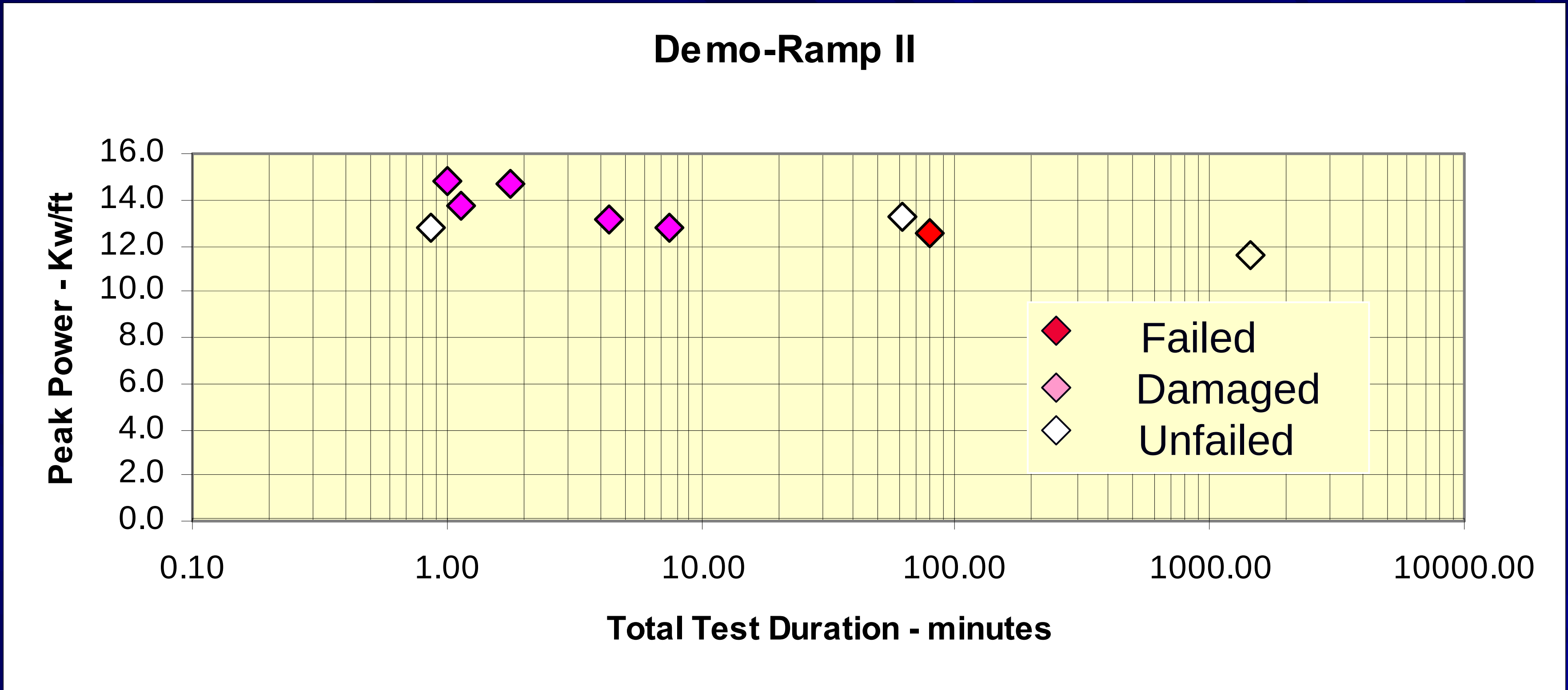


Ramp test time-to-failure detection methods



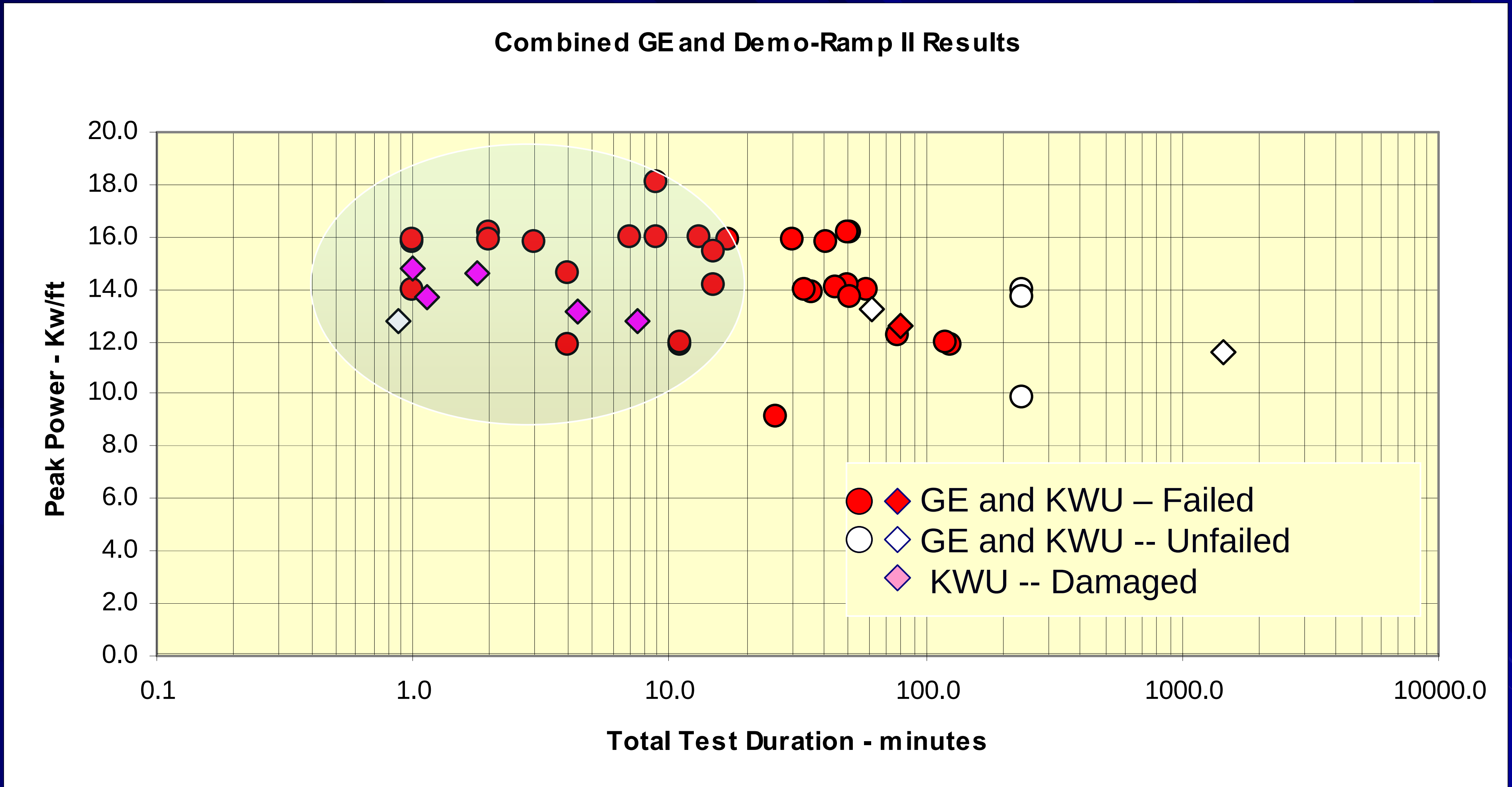
GE BWR Test Fuel Rods

- Irradiated in power reactors to burnups of 7- 28 Mwd / kgU at powers of 4-6 Kw/ft
- Pre test conditioned at 8 to 9 Kw/ft
- Power ramps of 2 to 8 Kw/ft at 2 to 100 Kw/ft-min
- 10 thru-wall PCI failures during 1 to 9 minute tests
- 17 thru-wall PCI failures during 10 to 110 minute tests
- 3 non-failed after 240 minute tests



KWU BWR Test Fuel Rods

- Irradiated in power reactors at 5 to 9 Kw/ft to burnups of 25 to 29 Gwd/t
- Pre-ramp conditioned at 9 Kw/ft
- Power ramps of 1.7 to 5.6 Kw/ft at rates of 1.6 to 10 Kw/ft-min
- One thru wall failure – during 79.8 minute test
- Five partial failures -- 10-60% thru-wall, during1 to 7 minute tests
- One non-failed during 0.87 minute test
- One non-failed during 61 minute and 1440 minute tests (same rod ramped twice)



Combined Ramp Test Results

- **Performance of GE and KWU test rods comparable**
- **Of the 16 tests with durations less than 10 minutes**
  - **9 failed with thru-wall PCI cracks – 5 failed within 3 minutes**
  - **6 had PCI cracks 10 to 60 % thru-wall – deepest occurred within 2 minutes**
  - **1 was not damaged during 0.87 minute test**

GEH Proprietary - Backups

GEH LFWH Analysis

NEDE-32538P-A
GE Proprietary Information

Table 4
LFWH Core Power and ACPR

Parameter	TRACG	PANACEA (Inlet Conditions from TRACG)
Core power to moderator for a 100°F LFWH (% of rated)	118.0 at 50 seconds	117.7
ACPR for a 100°F LFWH (M adjusted to the safety limit 1.07)	0.103 at 50 seconds	0.115

Browns' Ferry Plant Safety Analysis

FSAR BFN 16 Table 14.4-1Summary of Abnormal Operational Transients

Undesired Parameter Variation	Event Causing Transient	Scram Caused by
Nuclear system pressure increase	Generator trip without bypass	Turbine control valve fast closure
Nuclear system pressure increase	Turbine trip without bypass	Turbine stop valve closure
Nuclear system pressure increase	Main steam line isolation valve closure	Main steam line isolation valve closure
Nuclear system pressure increase	Loss of Condenser vacuum	Turbine stop valve closure
Nuclear system pressure increase	Bypass valve malfunction	Reactor vessel high pressure
Nuclear system pressure increase	Pressure regulator malfunction	Reactor vessel high pressure
Reactor water temperature decrease	Shutdown cooling malfunction decrease temperature	High Neutron flux
Reactor water temperature decrease	Loss of feedwater heater*	None
Reactor Water temperature decrease	Inadvertent pump start*	None

- Amendment 7 of GESTAR II SER transmittal letter states:
  - ***Should our criteria or regulations change so that our conclusions as to the acceptability of the report are invalidated, GE and /or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.“***

Fuel Design Criteria Licensing Clause

- Item 4 of Amendment 22 to GESTAR II states:

  " New-fuel-related licensing issues identified by the NRC will be evaluated to determine if the current criteria properly address the concern; if necessary, new criteria will be proposed to the NRC for approval."

- CONCLUSION

  – ***Obligation of licensees and fuel vendors to demonstrate that their fuel designs as operated will preclude known damage mechanism and meet GDC-10 and GDC-12 requirements***

PCI Summary

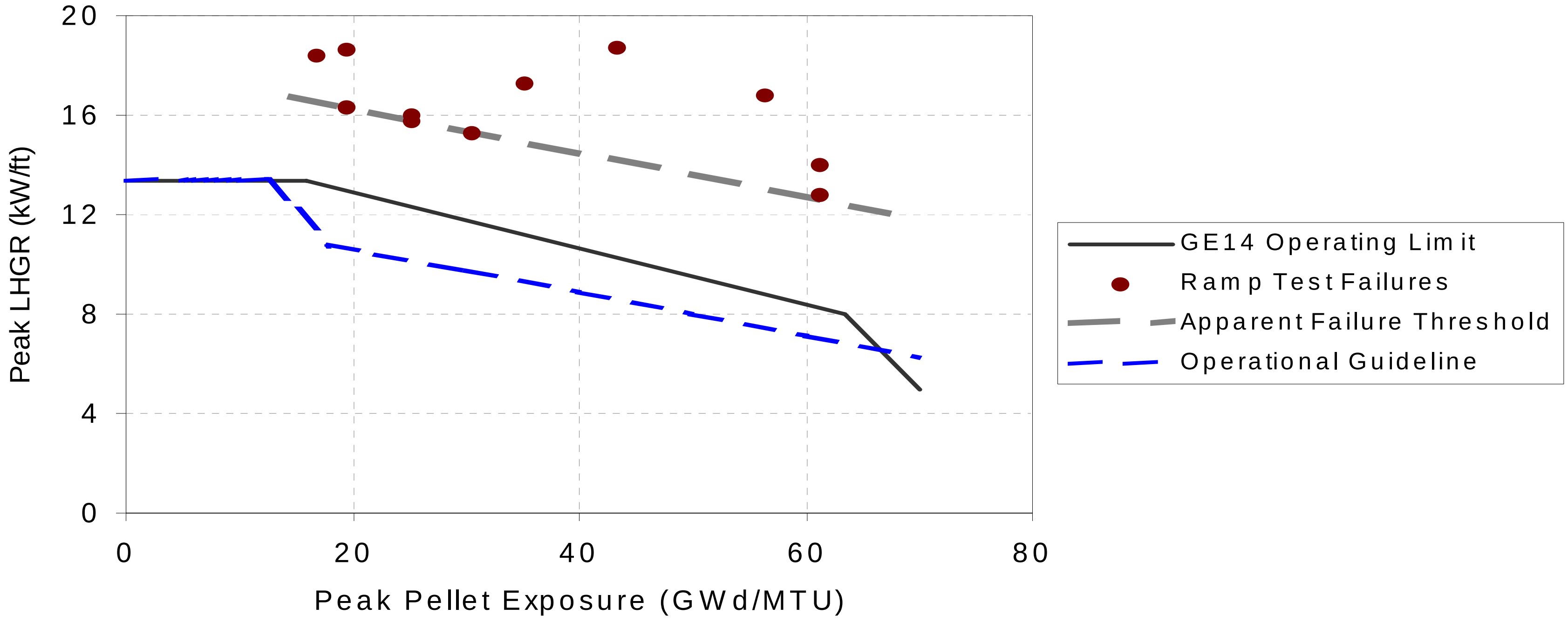
GE14 and SVEA-96+ fuel designs include barrier cladding which significantly reduces the PCI/SCC failure potential

No change in fuel duty/margin to the LHGR for the pre-EPU versus EPU conditions

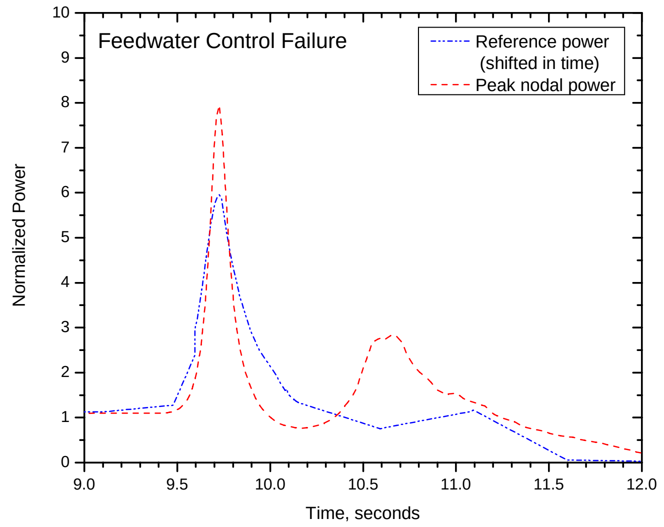
Operational guidelines provide additional margin to avoid PCI/SCC type fuel failures

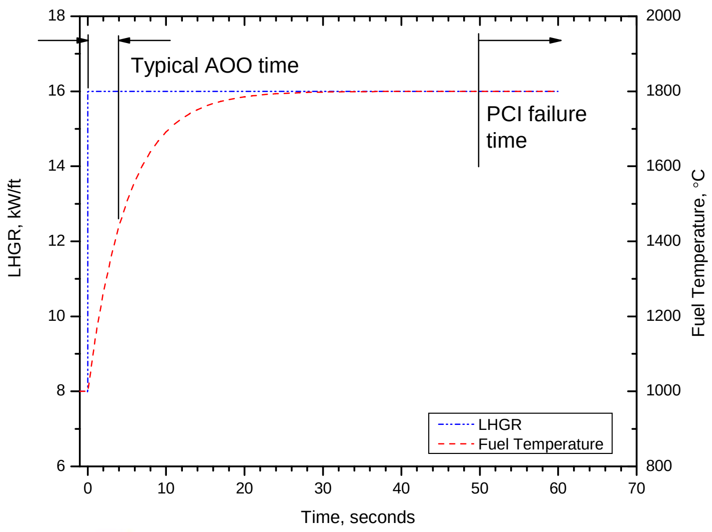
Hope Creek Uses Operating Guidelines to Reduce the Potential for PCI/SCC Type Fuel Failures

- Significant margin compared to ramp tests
  - Apparent failure stress threshold is ~60 ksi
  - Calculated stress at the typical operational guideline threshold is ~15 ksi



PCI Failure during AOOs



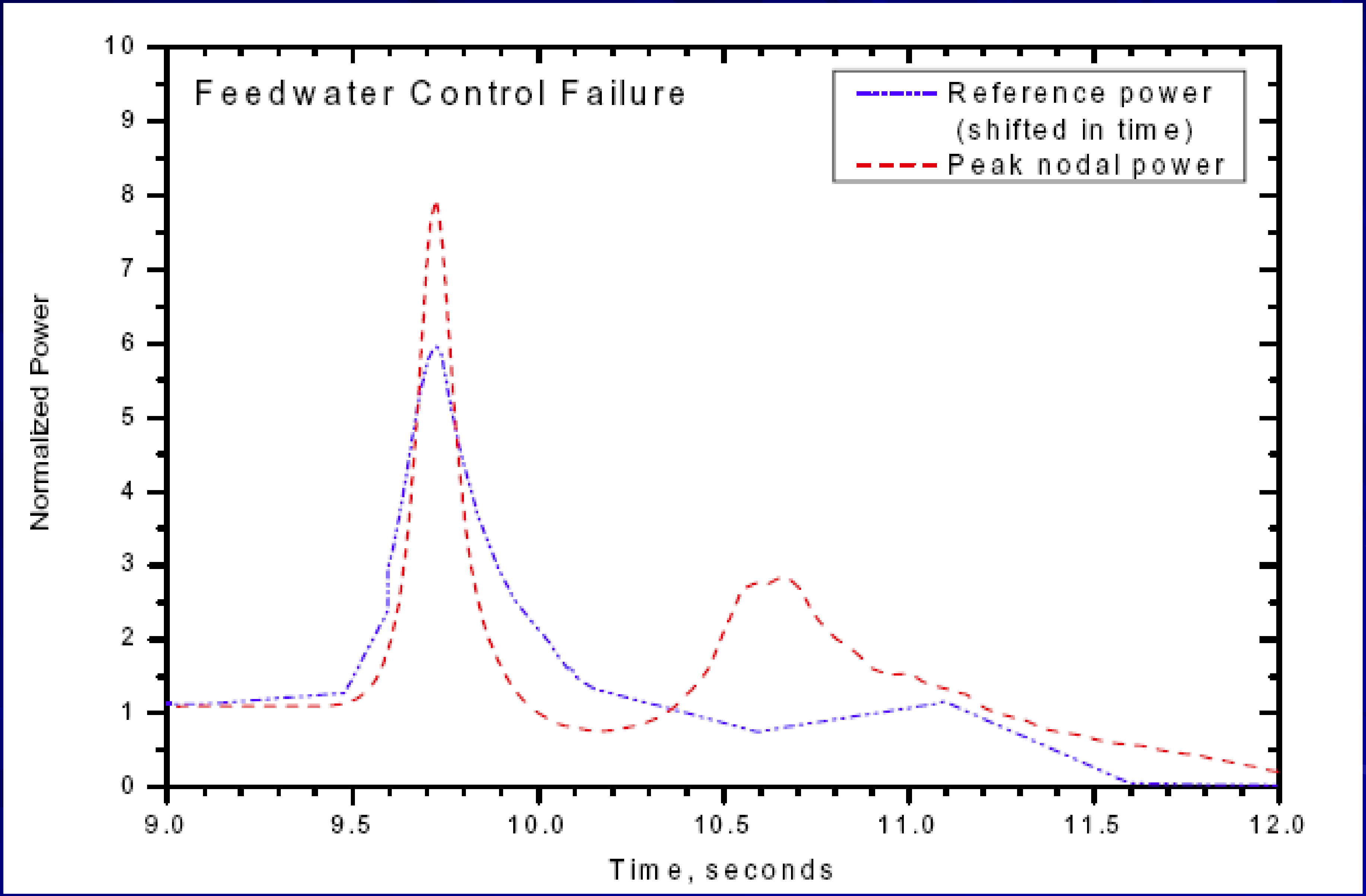


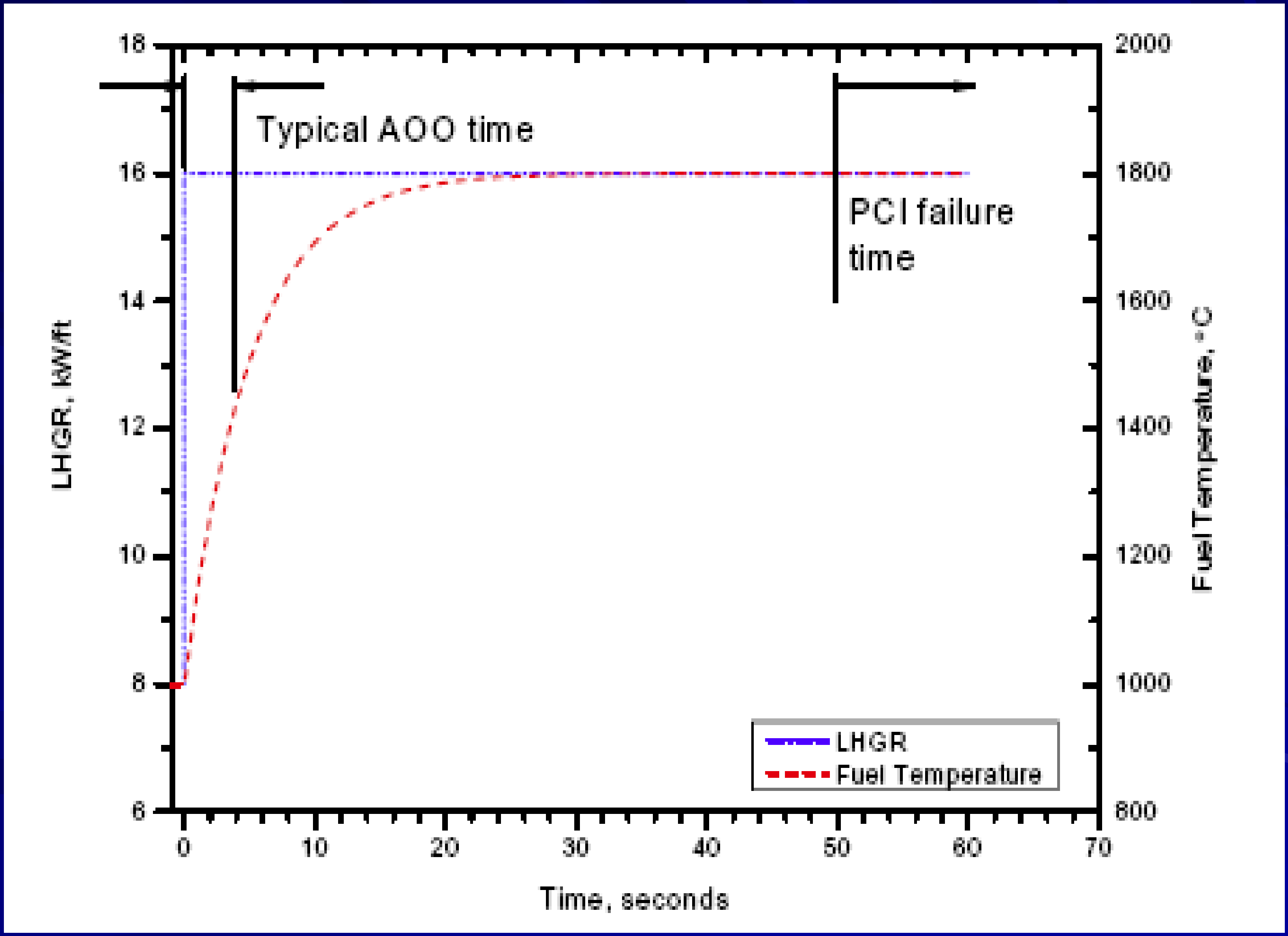
PCI failures can result from elastic loading, without plastic strain

- Requires time, temperature and stress
- Very localized and stochastic in nature

AOOs are events of short duration

- Not enough time for the corrosive fission products release and cause PCI/SCC


Feedwater Control Failure
Reference power
(shifted in time)
Peak nodal power
Normalized Power
Time, seconds
0
1
2
3
4
5
6
7
8
9
10
9.0
9.5
10.0
10.5
11.0
11.5
12.0



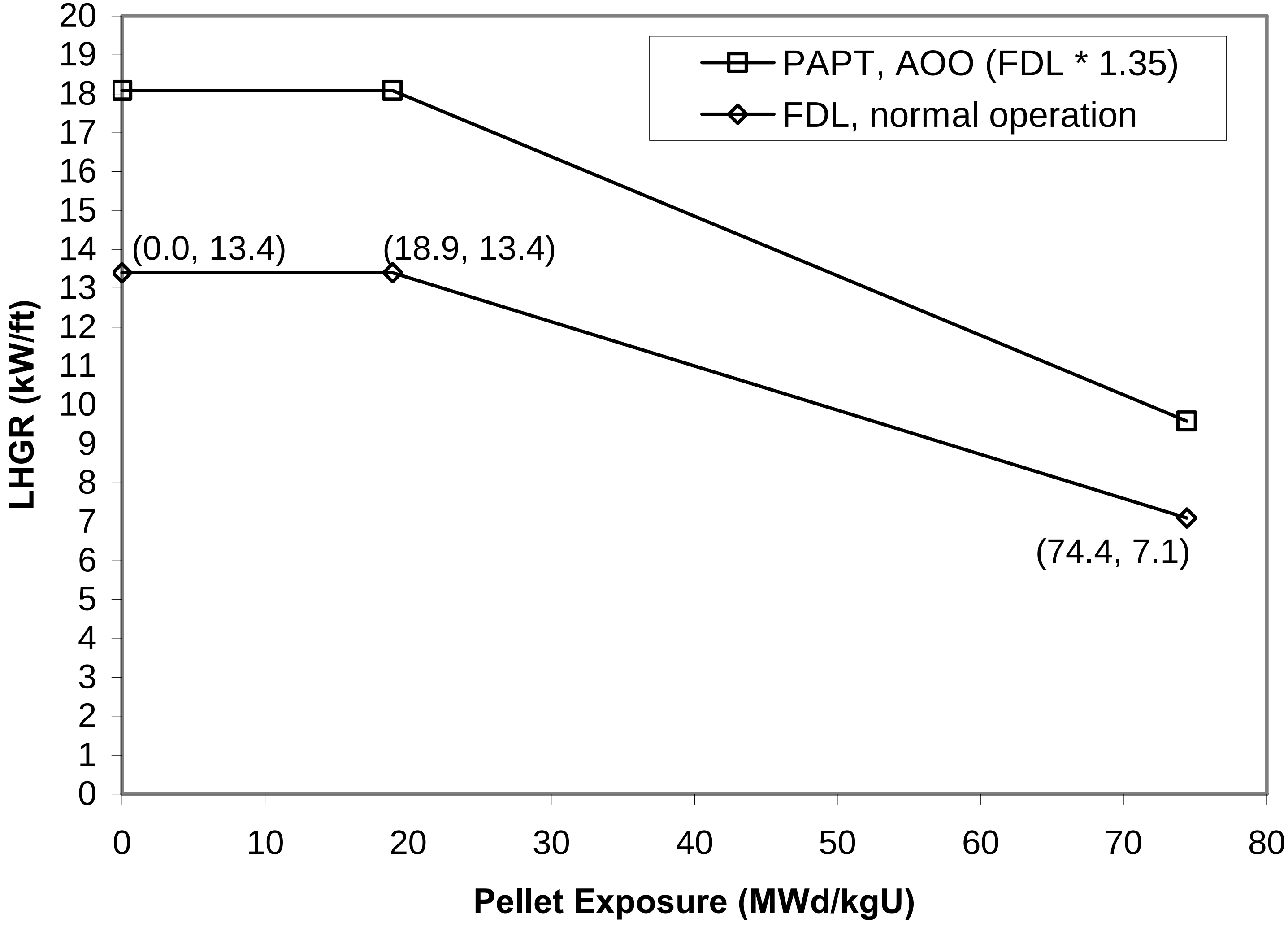
Typical AOO time
PCI failure time
LHGR, kW/ft
Fuel Temperature, °C
Time, seconds
LHGR
Fuel Temperature
6
8
10
12
14
16
18
800
1000
1200
1400
1600
1800
2000
0
10
20
30
40
50
60
70

AREVA Proprietary -- Backups

Thermal Mechanical Methods

Michael Garrett
Manager, BWR Safety Analysis

LHGR limits for ATRIUM™-10 fuel



Thermal Mechanical Methods

- Fuel thermal mechanical limits remain unchanged for Susquehanna CPPU operation
- 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 centerline melting) are not exceeded during anticipated operational occurrences (AOOs)
  - Neutronic design criteria ensure gadolinia rods are not limiting with respect to thermal mechanical criteria
  - Same FDL and PAPT limits for liner or non-liner cladding

Thermal Mechanical Methods
Liner Cladding and Standard Cladding

- Susquehanna uses standard (non-liner) cladding
  – Majority of ATRIUM-10 fuel supplied is non-liner
  – ATRIUM-10 failure-free operation in both Susquehanna units since introduction in 1997
  – FDL and PAPT limits are unchanged from pre-CPPU operation
- Use of liner cladding provides less restrictive maneuvering (power ramp rate) guidelines
  – Liner does not impact FDL or PAPT limits
  – Liner does not provide additional protection for SAFDLs

Thermal Mechanical Methods

- Cycle specific transient analyses performed to establish an operating limit LHGR that ensures PAPT limit is not exceeded during an AOO
- Potentially limiting AOOs were analyzed at CPPU conditions for Susquehanna
  - Limiting event for normal operation (loss of feedwater heating) resulted in overpower ratio of [24% (PAPT based on 35% overpower ratio)]
- The operating limit LHGR is specified in the COLR
- The core monitoring system is used to ensure the core is operated within the operating limit LHGR

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Protecting People and the Environment

PCI/SCC Regulatory Approach

ACRS Full Committee Meeting

June 3, 2009

Paul M. Clifford
Division of Safety Systems
Nuclear Reactor Regulation

ACRS Letter on Susquehanna EPU (December 20, 2007)

- The staff should develop the capability and perform a thorough review and assessment of the risk of pellet-cladding interaction (PCI) fuel failures with conventional fuel cladding, during anticipated operational occurrences (AOOs).
  - The staff should develop qualified analytical tools to demonstrate that operator actions will assure an acceptably low number of failures. If this can be demonstrated by analysis, then the required operator actions should be incorporated into the regulatory process through commitments or inclusion in the updated FSAR.

Staff Response to ACRS Letter (January 17, 2008)

- In response to recommendation 6, the NRC staff will investigate current computational capabilities to model the complex phenomena associated with non-uniform fuel pellet expansion and stress-corrosion cracking (SCC). As necessary, the staff will develop guidance related to an application methodology and regulatory approach for implementing a PCI/SCC fuel failure criteria.

Staff Concerns with ACRS Direction

Staff concerns with the specific direction:

- PCI/SCC phenomena difficult to model and requires tacit assumptions on chemical effects and initial crack depth.
- All domestic fuel designs susceptible to PCI/SCC
  - Various design features (e.g. doped pellets, low alloy Zr barrier, natural Zr barrier) provide varying levels of PCI/SCC resistance
  - Barrier fuel design provides PCI/SCC resistance, but not immune from failure during power maneuvering or AOOs
- Crediting prompt operator action in UFSAR Chapter 15

Important points to consider moving forward:

- Regulations specify performance requirements
  - Does not impose specific design features
- Regulations apply universally
  - Not restricted to a particular fuel of cladding design
- PCI/SCC not strictly an EPU issue or BWR issue

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PCI/SCC Work Priority

- PCI/SCC may yield fuel rod cladding failure (i.e., through wall crack releasing fission gas within plenum)
  – No challenge to core coolable geometry
  – No challenge to pressure vessel integrity
  – No challenge to containment integrity
  – No challenge to systems designed to mitigate transient and minimize offsite activity releases
- PCI/SCC safety significance does not warrant immediate action nor higher priority in staff workload planning than ongoing regulatory improvements.
  – Revision to 10 CFR 50.46(b) ECCS Acceptance Criteria
  – Revision to RG 1.183 Gap Source Terms
  – Revision to RG 1.77 RIA Acceptance Criteria

Change in Staff Position

- No regulations or Regulatory Guides specifically address PCI/SCC.
- Cladding failure mechanisms and SAFDLs defined within approved topical reports and captured within each plant's licensing basis via Technical Specifications and UFSAR.
- Any change to the treatment of PCI/SCC would constitute a **change in a regulatory staff position**.
  - Consider 10 CFR 50.109 "Backfitting" requirements.
  - Complete Regulatory Analysis (NUREG/BR-0058, Rev.04).

Alternative strategies:

1. Maintain current approach
2. PCI/SCC protection based on empirical failure threshold
3. PCI/SCC protection based on analytical models

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Proposed Strategy #1

Maintain Current Approach

- PROS:
  - Current approach provides reasonable level of protection during core-wide AOOs.
  - Staff resources devoted to more substantial regulatory improvements.
- CONS:
  - Potential fuel cladding breach during certain BWR AOOs due to PCI/SCC.
  - Lack of specific PCI/SCC guidance and regulatory criteria for future fuel designs.

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Proposed Strategy #2

PCI/SCC Protection based on Empirical Failure Threshold

- Revise SRP-4.2 guidance on PCI/SCC fuel failure mechanism and level of qualification to demonstrate no fuel failures during AOOs.
  - Quantification of PCI/SCC resistance of all fuel designs under AOO conditions.
  - Empirically derived fuel rod failure threshold based on change in rod power and elapsed time.
  - Calculated rod powers remain below empirical failure threshold during UFSAR Chapter 15 AOOs.
- PROS:
  - Strict compliance with GDC10
  - High confidence predictions on rod power history
  - Consistent with reactivity-initiated accident regulatory approach.
- CONS:
  - Will require empirical data from power ramp testing (facilities limited).
  - May necessitate changes in Operator procedures and training and/or PPS reactor trip setpoints.
  - Implementation costs expected to be high for industry and NRC.

Proposed Strategy #3

PCI/SCC Protection based on Analytical Models

- Revise SRP-4.2 guidance on PCI/SCC fuel failure mechanism and level of qualification to demonstrate no fuel failures during AOOs.
  - Verification and validation of analytical models capable of predicting, at high confidence levels, crack tip propagation and cladding failure under combined mechanical loading and chemical attach.
  - Calculated cladding stresses remain below analytical failure predictions during UFSAR Chapter 15 AOOs.
- PROS:
  - Strict compliance with GDC10
- CONS:
  - Will require empirical data from power ramp testing (facilities limited) to calibrate analytical models.
  - PCI/SCC phenomena difficult to model and requires tacit assumptions on chemical effects and initial crack depth.
  - No well-verified analytical models exist.
  - Standard modeling approach (95/95) likely to yield overly burdensome requirements.
  - May necessitate changes in Operator procedures and training and/or PPS reactor trip setpoints.
  - Implementation costs expected to be <u>very</u> high for industry and NRC.

Backfitting

- 10 CFR 50.109 “Backfitting” represents a regulatory hurdle for implementing changes in staff positions to currently licensed facilities.
- The rule requires a substantial increase in the overall protection of the public health and safety **and** that the direct and indirect costs of implementation for that facility are justified in view of this increased protection.
- It would be difficult to justify an exception to this rule under “compliance” or “adequate protection”.

Forward Fitting

- No regulatory expectation that requirements or staff positions remain stable for future requests for agency action/approval.
  - Expanding fuel failure mechanisms to explicitly account for PCI/SCC for future fuel designs is not a backfit.
- Due to ongoing fuel design enhancements, implementing forward-fit PCI/SCC requirements likely to encompass a majority of the fleet in a reasonable timeframe.
  - Application of forward-fit PCI/SCC requirements to licensing actions (e.g., EPUs) involving existing, approved fuel design?
- Regulatory Analysis needed to justify change in staff position.

QUESTIONS?

Backup Slides

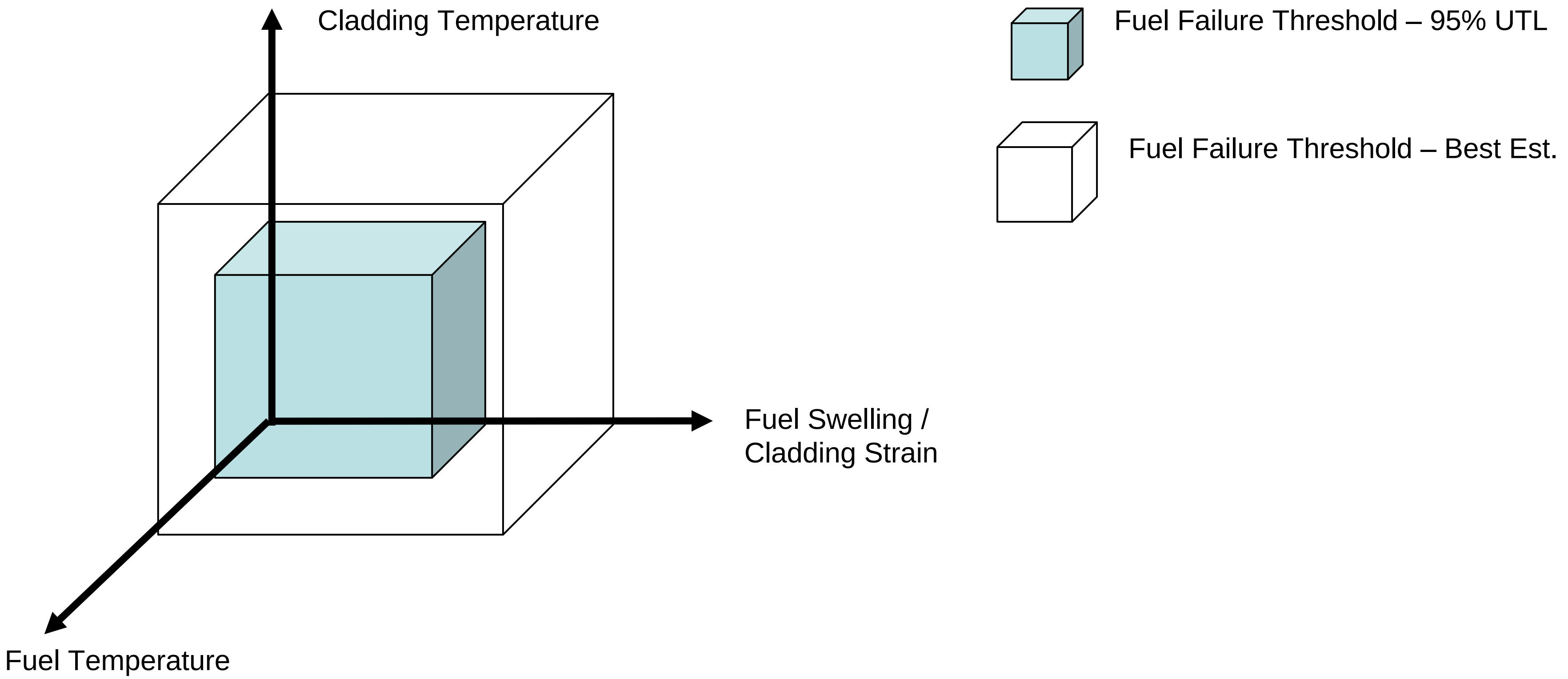
Low Safety Significance

- PCI/SCC may yield fuel rod cladding failure (i.e., through wall crack releasing fission gas within plenum)
  - No challenge to core coolable geometry
  - No challenge to pressure vessel integrity
  - No challenge to containment integrity
  - No challenge to systems designed to mitigate transient and minimize offsite activity releases

Low Probability of Occurrence

Limited envelope on magnitude of power excursion

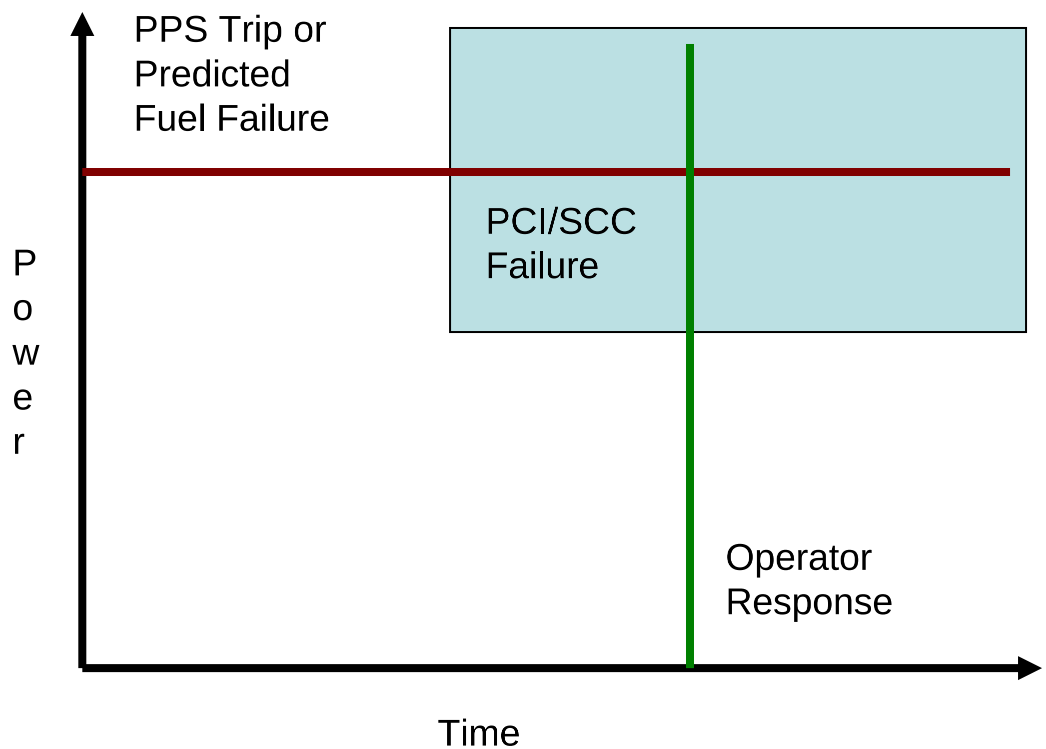
- Power level must remain below automatic trip setpoint.
- Power level must remain below level which results in predicted fuel failure calculated using conservative analytical models along with conservative assumptions and initial conditions.



Low Probability of Occurrence (cont.)

Limited envelope on duration of power excursion

- Duration beyond time necessary for PCI/SCC crack growth.
- Duration below timing for reasonable Operator response.



Current Staff Guidance

Standard Review Plan Section 4.2.II.1.B.vi, “Pellet/Cladding Interaction”

- *Two related criteria should be applied, but they are not sufficient to preclude PCI or PCMI failures. The first criterion limits uniform strain of the cladding to no more than 1 percent. In this context, uniform strain (elastic and inelastic) is defined as transient-induced deformation with gauge lengths corresponding to cladding dimensions; steady-state creepdown and irradiation growth are excluded. Mechanical testing must demonstrate that the irradiated cladding ductility at maximum waterside corrosion (hydride embrittlement) is well within the 1-percent strain criterion. Although observing this strain limit may preclude some PCI and PCMI failures, it will neither preclude the corrosion-assisted failures that occur at low strains nor the highly localized overstrain failures introduced by pellet chips on the outer fuel diameter. The second criterion states that fuel melting should be avoided. The large volume increase associated with melting may cause a pellet with a molten center to exert a stress on the cladding. Avoiding fuel melting can preclude such a PCI. Note that item 1.B.iv above invoked this same criterion to ensure that overheating of the cladding would not occur.*
- *Fuel vendors have introduced fuel design limits on power maneuvering and rate of power ascension to prevent PCI or PCMI. These design limits have primarily been based on power ramp data from test reactors for a specific fuel design. Recently, however, fuel vendors have been relying more on their predictions of cladding strain and less on their power ramp data to verify that PCMI will not occur. Convincing evidence exists that gaseous swelling and fuel thermal expansion is responsible for cladding strains at high burnup levels and perhaps at even moderate burnups. Therefore, PCI or PCMI analyses of cladding strain for AOO transients and accidents should apply approved fuel thermal expansion and gaseous fuel swelling models, as well as irradiated cladding properties.*

10 CFR 50.109 Backfitting

(a)(1) Backfitting is defined as the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission's regulations or the imposition of a regulatory staff position interpreting the Commission's regulations that is either new or different from a previously applicable staff position after:

10 CFR 50.109 Backfitting (con't.)

(3) Except as provided in paragraph (a)(4) of this section, the Commission shall require the backfitting of a facility only when it determines, based on the analysis described in paragraph (c) of this section, that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit **and** that the direct and indirect costs of implementation for that facility are justified in view of this increased protection.

10 CFR 50.109 Backfitting (con't.)

(4) The provisions of paragraphs (a)(2) and (a)(3) of this section are inapplicable and, therefore, backfit analysis is not required and the standards in paragraph (a)(3) of this section do not apply where the Commission or staff, as appropriate, finds and declares, with appropriated documented evaluation for its finding, either:

(i) That a modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee; or

(ii) That regulatory action is necessary to ensure that the facility provides adequate protection to the health and safety of the public and is in accord with the common defense and security; or

(iii) That the regulatory action involves defining or redefining what level of protection to the public health and safety or common defense and security should be regarded as adequate

US-APWR

Diversity and Defense-In-Depth

Topical Report

Safety Evaluation

To:
Advisory Committee on Reactor Safeguards

By:
Royce D. Beacom
Instrumentation, Controls and Electrical Engineering Branch (ICE1)
Office of New Reactors

Wednesday, June 3, 2009

Agenda

- Diversity and Defense-in-Depth Scope
- Findings and Conclusions
- Listing Sub-Committee Points of Discussion
- Addressing Each Point of Discussion

Diversity and Defense-in-Depth Scope

- Diversity with Safety and Non Safety Systems
  - Protection and Safety Monitoring System (Safety)
  - Plant Control and Monitoring System (Non-Safety)
- Both using the MELTAC Platform

Diversity and Defense-in-Depth Scope

- Functionality of the Diverse Actuation System (DAS) – (analog & non-safety)
  - Provides a defensive measure to cope with Anticipated Operational Occurrence (AOO) or Postulated Accident (PA) concurrent with Common Cause Failure in the PSMS which is beyond design basis

Diversity and Defense-in-Depth Scope

- Provides the ATWS Mitigation Function
- Provides Automatic Actuations where time is insufficient for manual operator action; MHI Proposed: < 10 mins
  - Delay from anticipated PSMS trip
  - Proper actuation of PSMS blocks DAS

Diversity and Defense-in-Depth Scope

- DAS Manual Actuation
  - Separate HSI Panel with conventional Controls and Indicators
  - Proposed < 30 min from Prompting Alarm
- Isolated signals from sensors, shared with PSMS, provided to Non-Safety DAS
- DAS Outputs to discrete portion of Power Interface Module (PIF)

Findings and Conclusions

- LBLOCA Coping Strategy
  - High quality, high reliability, measures of MELTAC within the RPS/ESFAS design
  - Low frequency of AOO and PA events
  - Supplemented with DAS leak protection
- LBLOCA Strategy unacceptable
  - Frequency of AOO / PAs still finite possibility
  - Leak-Before-Break doesn't apply here

Findings and Conclusions

- Protective action – Manual vs Automatic
  - MHI “Target” $\leq$ 10 minutes - Automatic
  - > 10 minutes – Manual Action is assumed
    - Differs from DI&C-ISG-02; < 30 Minutes – Auto
    - Insufficient information to assess manual action between 10 min and 30 min following the event
- Justification for manual actions within 30 minutes – US-APWR HSI Certification

Findings and Conclusions

- The staff concluded that the D3 approach, and the D3 analysis provided per NUREG-6303, had met the acceptable bases for conforming to the requirements and supporting industry standards.
- Subject to satisfactory completion of Application Specific Action Items (ASAI)

Sub-Committee Points of Discussion

The subcommittee meeting identified these points of discussion for the staff:

- How D3 fits in the Overview of US-APWR
- Separate approval of D3 from US-APWR?
- Bypassing DAS; PSMS actuation & Startup/ Shutdown
- Concept of two DAS Subsystems

Sub-Committee Points of Discussion

- ASAI on partial output failure from CCF
- Three DAS Inputs to Rx & Turbine Trip

How D3 fits in the Overview of US-APWR

- New reactor design only – US-APWR
  - MHI intent was for Operating Fleet also
- These Topical Reports are stand alone and will have separate SER's:
  - Safety I&C Sys. Description & Process, D3, MELTAC, HSI System Design & Process

How D3 fits in the Overview of US-APWR

- The safety evaluations of these Technical Reports will be included in the DCD SER
  - Defense-In-Depth Coping Analysis
  - Software Program Manual (Application SW)
- Application Specific Action Items (ASAIs) can be addressed in the following:
  - Directly in the DCD (Rev); ITAAC; COL Action Item

Separate approval of D3 from US-APWR?

- For attributes approved, level of detail is sufficient
  - Will not expect additional detail in DCD
- Staff is confident ASAIs are sufficient
  - Will address additional D3 Info needed
    - Particularly pointers to Coping Analysis

Separate approval of D3 from US-APWR?

- If Applicant/ Licensee cannot meet ASAIs
  - It is their risk to proceed with design or
  - Take exception to TR
    - Provide alternative path for staff approval

Bypassing DAS; PSMS actuation & During Startup/ Shutdown

- Diverse Actuation System (DAS) is bypassed when PSMS actuates:
  - If proper feedback from actuated components is received
  - Prevents unexpected competition between systems

Bypassing DAS; PSMS actuation & During Startup/ Shutdown

- DAS is bypassed at same time as PSMS
  – Required in Modes 1,2 & 3 and Pressurizer Pressure > P11
- However, enabled by different means:
  – PSMS is automatically interlocked
  – DAS is enabled by manual switch

Concept of DAS Subsystems

- Two subsystems balances two issues:
  - – Reliability & Spurious actuation of the DAS
- There are no 50.55(a)(h)/ IEEE 603 safety requirements applicable since this is a non-1E system
    - i.e. single failure, independence, EQ, quality etc.
  - – GL 85-06, ATWS QA applies (Not App. B)

ASAI on partial output failure from CCF

- Concept captured by the D3 Task Working Group of the DI&C Steering Committee.
- The staff position states that a simple failure of the total system may not be the worst case failure, particularly when analyzing the time required for identifying and responding to the condition. For example, a failure to trip may not be as limiting as a partial actuation of the emergency core cooling system, with indication of a successful actuation

ASAI on partial output failure from CCF

- At least two requirements from 50.59(a)(h)/ IEEE-603 are applicable:
  - **Completion of Protective Action**. The safety systems shall be designed so that, once initiated automatically or manually, the intended sequence of protective actions of the execute features shall continue until completion

ASAI on partial output failure from CCF

– **System Status Indication.** Display instrumentation shall provide accurate, complete, and timely information pertinent to safety system status. This information shall include indication and identification of protective actions of the sense and command features and execute features.

ASAI on partial output failure from CCF

- Staff has not proposed a method for addressing this issue – nor should they
- All digital upgrades eventually will have to address this issue

Three DAS Inputs to Automatic Reactor and Turbine Trip

- The DAS has three diverse automatic actuation functions to shut down the reactor and to achieve secondary system core heat removal.
  - High Pressurizer Pressure
  - Low Pressurizer Pressure
  - Low SG Level

Three DAS Inputs to Automatic Reactor and Turbine Trip

- If Chapter 15 event credits a specific reactor trip, an automatic DAS trip would occur on the same trip function
- Refer to the D3 Coping Analysis (MUAP-07014)
  - Discussion of event evaluation methods
  - Results of each event evaluation with CCF