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

July 28, 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 560th FULL COMMITTEE MEETING HELD ON MARCH 5-7, 2009 IN ROCKVILLE, MARYLAND

The minutes of the subject meeting were certified on April 21, 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

April 21, 2009

MEMORANDUM TO: Sherry Meador
Advisory Committee on Reactor Safeguards

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

SUBJECT: MINUTES OF THE 560th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), MARCH 5-7, 2009

I certify that based on my review of the minutes from the 560th 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	04/ 21 /09	04/ 21 /09

OFFICIAL RECORD COPY

CERTIFIED Date Certified: 04/21/2009

TABLE OF CONTENTS
MINUTES OF THE 560th ACRS MEETING

MARCH 5-7, 2009

Appendices

During its 560th meeting, March 5-7, 2009, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following report, letters, and memoranda:

REPORT

Report to Dale E. Klein, Chairman, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Draft Final Regulatory Guide 5.71, "Cyber Security Programs For Nuclear Facilities," dated March 19, 2009

LETTERS

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

- Draft Final Rule 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," dated March 13, 2009

- Draft Final Regulatory Guide 5.73, "Fatigue Management for Nuclear Power Plant Personnel," dated March 19, 2009

- Crediting Containment Overpressure in Meeting the Net Positive Suction Head Required to Demonstrate that the Safety Systems can Mitigate the Accidents as Designed, dated March 18, 2009

MEMORANDA

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

- Draft Final Revision 3 to Regulatory Guide 10.4, dated March 12, 2009

- Proposed Revision to Regulatory Guide 1.141 (DG-1213), dated March 12, 2009

- Proposed Revisions to Regulatory Guide 1.205 (DG-1218) and Standard Review Plan Section 9.5.1.2, dated March 12, 2009

MINUTES OF THE 560th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

ROCKVILLE, MARYLAND

The 560th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on March 5-7, 2009. Notice of this meeting was published in the *Federal Register* on February 19, 2009 (72 FR 7707-7709). 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. Other attendees can be found at the sign-in sheets in Appendix III.

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. Draft Final Regulatory Guide 5.71 (formerly DG-5022), "Cyber Security Programs for Nuclear Facilities"

[Note: Mrs. Christina Antonescu was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff to discuss the Draft Final Regulatory Guide (RG) 5.71, "Cyber Security Programs for Nuclear Facilities," and NRC staff's resolution of stakeholders' comments. 10 CFR 73.54 establishes performance-based requirements to ensure that the functions of critical systems and critical digital assets are protected from cyber attack using a graded approach. RG 5.71 conveys NRC staff positions for developing a program that provides an effective protection mechanism against cyber attacks. It also provides NRC staff positions regarding the minimum set of elements needed within a program to protect a facility, the networks within it, the plant systems, the digital assets that implement system functions, and operating systems and applications within those digital assets that accomplish the functions to be protected. A generic cyber security plan template, NEI-08-09, is also being jointly-developed between staff and industry for later consideration as a reference in RG 5.71.

The Committee issued a report to the NRC Chairman, dated March 19, 2009, recommending that RG 5.71 not be published until it is revised to: (i) provide a reference Digital I&C (DI&C) computer, communication, and network security framework that identifies assets, associated plant functions, vulnerabilities, and interaction and access pathways; (ii) include examples and more specific guidance on how the stated requirements can be met; (iii) ensure that the guidance distinguishes between DI&C system and non-real-time information technology system architectures; and (iv) address the issues of threat assessment, dependency analysis, and the use of probabilistic risk assessment (PRA).

III. Draft Final Revisions to 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events"

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

The Committee met with representatives of the NRC staff to discuss Draft Final rule 10 CFR 50.61a, "Alternative Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." The discussion on the technical basis of the rule centered on the studies used to show that the rule was generally applicable across the entire fleet of pressurized water reactors (PWRs). The screening limits in the proposed rule were developed through detailed studies of three PWRs: Beaver Valley, Oconee, and Palisades. The generalization studies then considered five additional plants in order to infer the broader applicability of the rule. The events that offered the most challenge to vessel were found to be driven by factors that are similar across the fleet.

The main features of 10 CFR 50.61a include: limitations on applicability; less restrictive screening criteria; evaluation of plant-specific flaw distributions; and implementation of new embrittlement models and surveillance data evaluations. The staff's conclusion was that 10 CFR 50.61a is appropriate for providing protection of public health and safety and for reducing unnecessary regulatory burden.

The Committee issued a letter to the Executive Director for Operations on this matter, dated March 13, 2009, recommending that the rule be approved and that the staff verify and document the capability of nondestructive examination procedures used to characterize flaw distributions in reactor vessels. The Committee also concluded that an effort is needed to plan for the most effective use of surveillance samples in tracking embrittlement trends.

IV. Draft Final Regulatory Guide 1.200 (formerly DG-1200), "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"

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

The Committee met with representatives of the NRC staff to discuss the Draft Final Revision 2 of Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." Revision 2 of Regulatory Guide 1.200 endorses a national consensus standard, jointly developed by the American Nuclear Society (ANS) and the American Society of Mechanical Engineers (ASME), regarding the content of a technically acceptable PRA. It also endorses PRA peer review guidance developed by the Nuclear Energy Institute (NEI). The staff discussed the resolution of public comments and the major outstanding industry issues. One of the industry comments is regarding independence of the peer reviewers. Industry claims that achieving the level of independence recommended by this guide could be a problem because of a lack of PRA experts. The staff also discussed the future work on standard development that would require future revisions to RG 1.200. The ACRS members' questions were related to alternative risk metrics, non-light water reactor PRA standards, and handling of uncertainty. The Committee plans to issue a letter on this matter during its April 2-4, 2009, meeting.

V. Nuclear Power Plant Personnel"

[Note: Ms. Yoira Diaz-Sanabria was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff, the Professional Reactor Operator Society (PROS), the Nuclear Energy Institute (NEI), and the International Brotherhood of Electrical Workers (IBEW) to discuss Draft Final Regulatory Guide 5.73, "Fatigue Management for Nuclear Power Plant Personnel." RG 5.73 endorses NEI 06-11, Revision 1, "Managing Personnel Fatigue at Nuclear Power Reactor Sites," with certain exceptions. The first exception relates to periodic overtime. The staff feels that allowing overtime on the day off could lead to rule violations, and that there is already sufficient flexibility to make this a rare situation. The second exception relates to outage activities at multi-unit sites. The staff disagrees with the NEI position that licensed operators responsible for an operating unit (and the operators who provide relief for them) can be considered to be "working on outage activities."

A representative of PROS stated that RG 5.73 will make utilities change the way they schedule staff on the front and back end of an outage, possibly having a negative impact on their ability to safely execute an outage. PROS recommended that instead of defining an outage as having a reactor unit disconnected from the grid, an outage be defined as the time period beginning one week prior to disconnecting the unit from the grid and ending when the unit is reconnected and achieves 75 percent power. This would require a rule change as well as a change to the regulatory guide. The PROS representative also recommended that for multi-unit sites, the concept of "outage unit" be replaced with a concept of "outage site." This would allow multiple unit facilities with combined control rooms to modify all site personnel schedules to accommodate the outage. The advantages of this include a simpler set of work rules and more opportunity to keep work teams intact over the outage.

A representative of NEI discussed industry implementation of the fatigue management rule and stated that the regulatory guide's exceptions to NEI 06-11 are not necessary. It is not clear what problem is being addressed by the exception to outage activities. Regarding overtime, the NRC staff position adds to the complexity of schedule changes and would result in a significant distraction for first-line supervision.

A representative of IBEW agreed with the PROS proposal on the definition of outages. Regarding the issue of periodic overtime, IBEW supports the NEI position. Regarding multi-unit outages, IBEW is still working to address the issue of multi-unit outages.

The Committee issued a letter to the Executive Director for Operations on this matter, dated, March 19, 2009, recommending that Draft Final Regulatory Guide 5.73 be issued as final. The Committee also recommended that the staff closely track industry pilot applications and confirm that practical scheduling and time monitoring programs achieve the desired fatigue management objectives within an integrated framework that maintains stable shift manning and workforce controls throughout all plant operating modes.

VI. International Human Reliability Analysis (HRA) Empirical Pilot Study

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

The Committee met with representatives of the NRC staff, Sandia National laboratories (SNL), the Organization for Economic Cooperation and Development Halden Reactor Project (HRP), and Paul Scherrer Institute (PSI) to discuss International Human Reliability Analysis (HRA) Empirical Study. The NRC staff described the benchmark study of HRA methods using control room simulator data; the NRC and industry efforts to improve HRA guidance; interactions with national and international experts to pursue testing and benchmarking of HRA methods; and further support for Halden Reactor Project initiatives. The NRC staff also discussed the value of simulator exercises in providing insights on issues related to human behavior that can be used to improve HRA methods.

Representatives of HRP discussed the new data analysis approach that was developed to optimize the comparison of HRA methods predictions to empirical observations and improve the usefulness of simulator data for HRA purposes. They also discussed experimental work that identified the extent of crew-to-crew variability, the significance of teamwork factors, and the importance of event dynamics in determining crew performance.

The representative of SNL discussed the qualitative analysis of HRA methods and its comparison with Halden crew data. All HRA methods identified some of the important factors that affected crew performance, but most HRA analyses failed to identify important factors for some Human Failure Events (HFEs). Additionally, several methods have significantly over- or underestimated the difficulty of some HFEs.

The representative of PSI discussed the quantitative comparisons between HRA predictions and the empirical data. There were significant limitations on the quantitative results. The quantitative comparisons supplement the qualitative comparisons and insights. The overall evaluation of the HRA methods is based on both qualitative and quantitative insights; however, the qualitative insights should be weighted more strongly. This was an information briefing and no Committee action was necessary at this time.

VII. Subcommittee Reports

Plant License Renewal Subcommittee Report (Indian Point License Renewal Application)

The Chairman of the Plant License Renewal Subcommittee provided a report to the Committee summarizing the results of the March 4, 2009, meeting with the NRC staff and representatives of Entergy Nuclear Operations Inc. (Entergy) to review the Draft Safety Evaluation Report (SER) with Open Items related to the license renewal application for the Indian Point Nuclear Generating Station, Units 2 (IP2) and 3 (IP3).

The current operating licenses for the IP2 and IP3 expire on September 28, 2013, and December 12, 2015, respectively. Entergy submitted the license renewal application on April 23, 2007. The staff's draft SER, issued in January 2009 contained 20 open items. During the meeting, Entergy described the plant, its operating history, the license renewal review methodology, the aging management programs, and its commitment tracking system. The staff discussed the open items and stated that 13 out of the 20 open items have been closed. The staff and Entergy are in the process of resolving the remaining seven open items.
A public interest group, Riverkeeper, provided written comments regarding its concerns on the Indian Point license renewal and made oral statements during the meeting.

The Committee plans to review the final SER related to the license renewal application for the Indian Point Nuclear Generating Station, Units 2 and 3 in September 2009.

US-APWR Subcommittee Report

The Chairman of the US-APWR Subcommittee provided a report to the Committee summarizing the results of the February 18-20, 2009, site visits to Westinghouse offices in Monroeville, PA, and Mitsubishi Electric Power Products, Inc. (MEPPI) facilities in Cranberry Township, PA. On February 18, 2009, the Subcommittee participated in a tour and demonstration of the Westinghouse AP1000 digital control room simulator. On February 19, 2009, the Subcommittee met with representatives of the NRC staff and MEPPI to discuss three topical reports related to large-break loss-of-coolant-accident (LOCA), small-break LOCA, and non-LOCA methodologies associated with the US-APWR Design. On February 20, 2009, the subcommittee participated in a tour and demonstration of the Mitsubishi APWR digital control room simulator. The Committee plans to continue its review of the topical reports and draft SERs related to the US-APWR design certification in future meetings.

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 January 23, 2009, to conclusions and recommendations included in the December 18, 2008, ACRS letter on the technical basis and rulemaking strategy for the revision of 10 CFR 50.46(b), "Loss of Coolant Accident Embrittlement Criteria for Fuel Cladding Materials." 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 for the March ACRS Meeting

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

Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through June 2009 was discussed. The objectives were:

- 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

Biennial ACRS Report on the NRC Safety Research Program

The biennial ACRS report on the NRC Safety Research Program is due to the Commission in March 2010. Drs. Shack and Powers have the lead in coordinating the preparation of the report. Proposed assignments provided by Dr. Powers were discussed and members should provide input to Drs. Powers, Shack, and Nourbakhsh by September 15, 2009. A draft report will be prepared for Committee consideration during the October 2009 meeting.

ACRS Meeting With the Commission

The ACRS is scheduled to meet with the Commission on June 4, 2009 to discuss items of mutual interest. A list of topics proposed by the ACRS is:
Overview
Accomplishments
Future Plant Activities
License Renewal/Power Uprates
Ongoing/Future Activities
Containment Overpressure Credit Issue
Pressurized Thermal Shock Rule

Digital I&C Matters
Options to Revise NRC Regulations Based on ICRP Recommendations

Draft Regulatory Guides and Proposed Standard Review Plan

The staff plans to issue the following Draft Regulatory Guides (DG) and proposed Standard Review Plan (SRP) for public comment and would like to know whether the Committee wants to review these documents prior to being issued for public comment.

Proposed Revision 1 to Regulatory Guide 1.205 (DG-1218), "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants"

The staff issued the initial RG 1.205 in May 2006 to provide a method acceptable to the staff to comply with the requirements in 10 CFR 50.48(c) (commonly known as the National Fire Protection Association (NFPA) 805 Rule). At this time, 48 reactor units are transitioning to NFPA 805 using guidance provided in current version of RG 1.205. This includes pilot plants Oconee Units 1, 2, 3 and Shearon Harris. The non-pilots will benefit from the lessons learned from the pilot reviews. The revision of the RG is one of several vehicles to share lessons learned from the pilot reviews with other plants. The staff received the pilot license amendment requests in June 2008. Pilot plants supplemented their submittals with additional information. Based on the lessons learned from the pilot reviews, the staff is revising RG 1.205.

Draft Regulatory Guides (DG) 1213, "Containment Isolation Provisions for Fluid Systems"

This Guide describes updated methods that the staff considers acceptable for use in complying with the Commission's requirements for containment isolation of fluid systems. DG-1213 is a proposed Revision 1 to Regulatory Guide 1.141 of the same title. The changes to the Guide consist of adding provisions in the Regulatory Positions to reflect operating experience. The American National Standards Institute (ANSI) N271-1976 Standard is still applicable, subject to the provisions listed in DG-1213.

Proposed Standard Review Plan (SRP) Section 9.5.1.2, "Risk-Informed, Performance-Based Fire Protection Program"

This guidance is being issued as an alternate to the existing guidance currently provided under SRP section 9.5.1. This is stand alone guidance and is provided for the benefit of licensees of existing plants who choose to adopt Risk-Informed, Performance-Based Fire Protection Program that meets the requirements of National Fire Protection Association (NFPA) Standard 805. The NRC staff will also incorporate the approved SRP Section 9.5.1.2 into the next revision of RG 1.205 and any related guidance documents.

Draft Final Regulatory Guide

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

Draft Final Revision 3 to Regulatory Guide 10.4, "Guide for the Preparation of Applications for Licenses to Process Source Material"

Regulatory Guide 10.4 describes the information currently acceptable to the NRC staff for the review of applications for new, renewal, or amended licenses to process source material in research and development, shielding, and alloy manufacturing. The proposed Revision 3 to Regulatory Guide 10.4 was issued for public comment as DG-0020 in September 2008. The public comment period closed November 10, 2008. No comments were received.

Pellet Clad Interaction Failures Under Expended Power Uprate (EPU) Conditions

During the meeting on March 3, 2009, the Materials, Metallurgy, and Reactor Fuels Subcommittee met with the staff to discuss pellet clad interaction failures under EPU conditions. This issue was raised during the ACRS review of the Susquehanna EPU application.

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

The members plan to visit a plant in Region II. Since the ACRS is in the process of reviewing the activities associated with the Watts Bar Unit 2 operating license, it was suggested that the members consider touring the Watts Bar plant (Unit 1 Operating and Unit 2 - 60% complete).

The meeting was adjourned at 12:00 noon on March 7, 2009.

performed at the B&W NOG facilities in Lynchburg, Virginia. The 2007 request for the then-proposed license transfer was previously noticed in the **Federal Register** on Friday, August 1, 2008; 73 FR 45089, with a notice of an opportunity to request a hearing. No hearing requests were submitted.

On November 12, 2008, the NRC issued an Order approving the proposed license transfer. This order was accompanied by a Safety Evaluation Report (SER) documenting the basis for the license transfer approval, and a license amendment. This license amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended, and NRC's rules and regulations as set forth in 10 CFR Chapter 1. The license was transferred from BWXT to B&W NOG on January 11, 2009.

Further Information

In accordance with 10 CFR 2.390 of NRC's ''Rules of Practice,'' details with respect to this action, including the SER and accompanying documentation included in the license Amendment package, are available electronically, at NRC's Electronic Reading Room, at *http://www.nrc.gov/reading-rm/adams.html*. From this site, you can access NRC's Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. The ADAMS accession numbers for documents related to this action are:

Documents	ADAMS Accession No.
November 14, 2007: Initial Application	ML080920759
December 7, 2007: Request for Additional Information (RAI) Request I	ML073340643
December 10, 2007: RAI Response I	ML073460400
December 17, 2007: Meeting Minutes	ML080090688
January 7, 2008: Application Supplement	ML080160257
January 7, 2008: Application Supplement	ML080160149
January 11, 2008: Application Supplement	ML080230599
February 1, 2008: RAI Request II	ML080280551
February 1, 2008: Proprietary Determination I	ML080150394
February 15, 2008: RAI Response II	ML080920674
February 29, 2008: Response to Proprietary Determination	ML080640268
March 19, 2008: Application Acceptance	ML080710555
March 31, 2008: Proprietary Determination II	ML080790072
April 24, 2008: RAI Request III	ML081050308
June 27, 2008: Response to RAI Request III	ML082810598
October 29, 2008: SER	ML082600362
Application supplements via e-mails	**ADAMS Accession No.**
December 12, 2007	ML081190572
December 12, 2007	ML081190669
December 12, 2007	ML081190672
December 13, 2007	ML081190671
December 13, 2007	ML081190670
January 9, 2008	ML081190624
January 14, 2008	ML081190661
March 13, 2008	ML081190657
August 19, 2008; Response to NRC questions	ML082690414
August 26, 2008: NRC Additional Questions	ML082690283
September 5, 2008: Response to NRC Questions	ML082690224
Miscellaneous	**ADAMS Accession No.**
July 17, 2008: FEDERAL REGISTER notice	ML073511429
September 23, 2008: Environmental Assessment State Consultation	ML083010195
November 12, 2008: Order approving Transfer with SER	ML082910211
January 9, 2009: Extension of Order	ML083530787

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, to *pdr.resource@nrc.gov*.

These documents may also be viewed electronically on the public computers located at NRC's PDR, O 1 F21, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852. The PDR reproduction contractor will copy documents for a fee.

Dated at Rockville, Maryland, this 10th day of February, 2009.

For the U.S. Nuclear Regulatory Commission.

Peter Habighorst,

Branch Chief, Office of Nuclear Material Safety and Safeguards, Division of Fuel Cycle Safety and Safeguards.

[FR Doc. E9–3505 Filed 2–18–09; 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 March 5–7, 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).

Thursday, March 5, 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.–10:15 a.m.: Draft Final Regulatory Guide 5.71 (formerly DG–5022), ''Cyber Security Programs for Nuclear Facilities'' (Open/Closed)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft final Regulatory Guide 5.71, NRC Staff's resolution of stakeholders' comments, and related matters. [**Note:** A portion of this session may be closed to protect information classified as National Security Information as well as Safeguards Information pursuant to 5 U.S.C. 552b(c)(1) and (3).]

10:30 a.m.–12:15 p.m.: Draft Final Revisions to 10 CFR 50.61: ''Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events'' (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft final revisions to 10 CFR 50.61 on the Pressurized Thermal Shock Events, NRC staff's resolution of public comments, and related matters.

1:15 p.m.–2:45 p.m.: Draft Final Regulatory Guide 1.200 (formerly DG–1200), ''An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities'' (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft final Regulatory Guide 1.200, NRC staff's resolution of public comments, and related matters.

3 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters discussed during this meeting as well as a proposed report on containment overpressure issue.

Friday, March 6, 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.–10 a.m.: Draft Final Regulatory Guide 5.73 (formerly DG–5026), ''Fatigue Management for Nuclear Power Plant Personnel'' (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft final Regulatory Guide 5.73, NRC staff's resolution of public comments, and related matters.

10:15 a.m.–12:15 p.m.: International Human Reliability Analysis (HRA) Empirical Pilot Study (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and international stakeholders regarding the HRA Empirical Pilot Study and related matters.

1:15 p.m.–2 p.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings and other matters related to the conduct of the 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.]

2 p.m.–2:15 p.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

2:15 p.m.–2:45 p.m.: Subcommittee Reports (Open)—The Committee will hear reports by and hold discussions with the Chairmen of the Plant License Renewal Subcommittee and the US–APWR Subcommittee regarding interim review of the Indian Point license renewal application and the NRC staff's Safety Evaluation Report with Open Items; and selected Topical Reports associated with the US–APWR design, respectively.

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

Saturday, February 7, 2009, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

8:30 a.m.–2:30 p.m.: Preparation of ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

2:30 p.m.–3 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 and protect information classified as National Security Information as well as Safeguards Information pursuant to 5 U.S.C. 552b(c)(1) and (3). In addition, it may be necessary to close a portion of the meeting to discuss organizational and personnel matters that relate solely to the internal personnel rules and practices of the ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy pursuant to 5 U.S.C. 552b(c)(2) and (6).

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, as well as the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Girija Shukla, Cognizant ACRS staff (301–415–6855), between 7:15 a.m. and 5 p.m. (ET). ACRS meeting agenda, meeting transcripts, and letter reports are available through the NRC Public Document Room at *pdr.resource@nrc.gov,* or by calling the PDR at 1–800–397–4209, or from the

Publicly Available Records System (PARS) component of NRC's document system (ADAMS) which is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* or *http://www.nrc.gov/reading-rm/doc-collections/ACRS/.*

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

Dated: February 12, 2009.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E9–3498 Filed 2–18–09; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request

Upon Written Request, Copies Available From: Securities and Exchange Commission, Office of Investor Education and Advocacy, Washington, DC 20549–0213.

Extension:

Form ADV–E; Sec File No. 270–318; OMB Control No. 3235–0361.

Notice is hereby given that, pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*) the Securities and Exchange Commission (the ''Commission'') has submitted to the Office of Management and Budget a request for extension of the previously approved collection of information discussed below.

Form ADV–E (17 CFR 279.8) is the cover sheet for accountant examination certificates filed pursuant to rule 206(4)–2 (17 CFR 275.206(4)–2) under the Investment Advisers Act of 1940 (15 U.S.C. 80b–1 *et seq.*) by investment advisers retaining custody of client securities or funds. The annual burden is approximately three minutes per respondent.

The estimate of burden hours set forth above is made solely for the purposes of the Paperwork Reduction Act and is not derived from a comprehensive or even representative survey or study of the cost of SEC rules and forms.

The information provided on Form ADV–E is mandatory. Responses will not be kept confidential. An agency may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid control number.

Please direct general comments regarding the above information to the following persons: (i) Desk Officer for the Securities and Exchange Commission, Office of Management and Budget, Room 10102, New Executive Office Building, Washington, DC 20503 or e-mail to: *Shagufta_Ahmed@omb.eop.gov*; and (ii) Charles Boucher, Director/CIO, Securities and Exchange Commission, C/O Shirley Martinson, 6432 General Green Way, Alexandria, VA 22312; or send an e-mail to: *PRA_Mailbox@sec.gov*. Comments must be submitted to OMB within 30 days of this notice.

February 11, 2009.

Florence E. Harmon,

Deputy Secretary.

[FR Doc. E9–3431 Filed 2–18–09; 8:45 am]

BILLING CODE 8011–01–P

SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request

Upon Written Request, Copies Available From: U.S. Securities and Exchange Commission, Office of Investor Education and Advocacy, Washington, DC 20549–0213.

Extension:

Rule 17Ac2–1, SEC File No. 270–95, OMB Control No. 3235–0084.

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*), the Securities and Exchange Commission (''Commission'') has submitted to the Office of Management and Budget (''OMB'') a request for approval of extension of the existing collection of information provided for in the following rule: Rule 17Ac2–1 (17 CFR 240.17Ac2–1) under the Securities Exchange Act of 1934 (15 U.S.C. 78a et seq.) (''Exchange Act'').

Rule 17Ac–2, pursuant to Section 17A(c) of the Exchange Act, generally requires transfer agents to register with their Appropriate Regulatory Agency (''ARA''), whether the Commission, the Comptroller of the Currency, the Board of Governors of the Federal Reserve System, the Federal Deposit Insurance Corporation, or the Office of Thrift Supervision, and to amend their registrations if the information becomes inaccurate, misleading, or incomplete.

Paragraph 1 of Rule 17Ac–2, requires transfer agents to file a Form TA–1 application for registration with the Commission where the Commission is their ARA. Transfer agents must also file an amended Form TA–1 application for registration if the existing on their Form TA–1 becomes inaccurate, misleading, or incomplete. The Form TA–1s must be filed with the Commission electronically, absent an exemption, on EDGAR pursuant to Regulation S–T (17 CFR 232).

The Commission receives on an annual basis approximately 100 applications for registration on Form TA–1 from transfer agents required to register with the Commission. Included in this figure are amendments to Form TA–1 as required by Paragraph (c) of Rule 17Ac2–1 to address information that has become inaccurate, misleading, or incomplete. Based on past submissions, the staff estimates that the average number of hours necessary to comply with the requirements of Rule 17Ac–1 and Form TA–1 is one and one-half hours with a total burden of 150 hours.

An agency may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid control number.

Comments should be directed to: (i) Desk Officer for the Securities and Exchange Commission, Office of Information and Regulatory Affairs, Office of Management and Budget, Room 10102, New Executive Office Building, Washington, DC 20503 or by sending an e-mail to: *Shagufta_Ahmed@omb.eop.gov*; and (ii) Charles Boucher, Chief Information Officer, Securities and Exchange Commission, c/o Shirley Martinson, 6432 General Green Way, Alexandria, VA 22312 or send an e-mail to: *PRA_Mailbox@sec.gov*. Comments must be submitted within 30 days of this notice.

Dated: February 11, 2009.

Florence E. Harmon,

Deputy Secretary.

[FR Doc. E9–3433 Filed 2–18–09; 8:45 am]

BILLING CODE 8011–01–P

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

February 11, 2009

AGENDA
560th ACRS MEETING
MARCH 5-7, 2009

THURSDAY, MARCH 5, 2009, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 – 10:15 A.M. Draft Final Regulatory Guide 5.71 (formerly DG–5022), "Cyber Security Programs for Nuclear Facilities" (Open/Closed) (GEA/CEA)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff regarding draft final Regulatory Guide 5.71, NRC staff's resolution of stakeholders' comments, and related matters.

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

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

10:15 – 10:30 A.M. * BREAK *****

3) 10:30 – 12:15 P.M. Draft Final Revisions to 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events" (Open) (WJS/MLB/CLB)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding draft final revisions to 10 CFR 50.61 related to Pressurized Thermal Shock Events, NRC staff's resolution of public comments, and related matters .

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

12:15 – 1:15 P.M. ***** LUNCH *****

4) 1:15 – 2:45 P.M. Draft Final Regulatory Guide 1.200 (formerly DG-1200), "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Open) (DCB/HPN)

4.1) Remarks by the Subcommittee Chairman
4.2) Briefing by and discussions with representatives of the NRC staff regarding draft final Regulatory Guide 1.200, NRC staff's resolution of public comments, and related matters.

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

2:45 –3:00 P.M. ***** BREAK**

5) 3:00 – 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

5.1) Draft Final Regulatory Guide 5.71, "Cyber Security Programs for Nuclear Facilities" (GEA/CEA)
5.2) Draft Final Revisions to 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events" (WJS/MLB/CLB)
5.3) Draft Final Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (DCB/HPN)
5.4) Containment Overpressure Credit Issue (WJS/MVB/ZA)

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

6) 8:30 – 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (MVB/EMH/SD)

7) 8:35 – 10:00 A.M. Draft Final Regulatory Guide 5.73 (formerly DG-5026), "Fatigue Management for Nuclear Power Plant Personnel" (Open) (JWS/HJV)

7.1) Remarks by the Subcommittee Chairman
7.2) Briefing by and discussions with representatives of the NRC staff regarding draft final Regulatory Guide 5.73, NRC staff's resolution of public comments, and related matters.

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

10:00 – 10:15 A.M. ***** BREAK *****

8) 10:15 – 12:15 P.M. International Human Reliability Analysis (HRA) Empirical Pilot Study (Open) (GEA/HPN)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff and international stakeholders regarding the HRA Empirical Pilot Study 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 *****

9) 1:15 – 2:00 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed) (MVB/EMH)

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

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

[NOTE: A portion of this 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) 2:00 – 2:15 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.

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

11.1) Report by and discussions with the Chairman of the Plant License Renewal Subcommittee regarding Interim Review of the Indian Point License Renewal Application and the Safety Evaluation Report (SER) with Open Items that were discussed during the Subcommittee meeting on March 4, 2009 (OLM/PW)

11.2) Report by and discussions with the Chairman of the US-APWR Subcommittee regarding selected Topical Reports associated with the US-APWR Design that were discussed during the Subcommittee meeting on February 19, 2009,

as well as insights gained from the tour of the Mitsubishi nd Westinghouse simulators on February 18 and 20, 2009 (OLM/NMC)

2:45 –3:00 P.M. ***** BREAK**

12) 3:00 – 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
12.1) Draft Final Regulatory Guide 5.71, "Cyber Security Programs for Nuclear Facilities" (GEA/CEA)
12.2) Draft Final Revisions to 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events" (WJS/MLB/CLB)
12.3) Draft Final Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (DCB/HPN)
12.4) Containment Overpressure Credit Issue (WJS/MVB/ZA)
12.5) Draft Final Regulatory Guide 5.73, "Fatigue Management for Nuclear Power Plant Personnel" (JWS/HJV)

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

13) 8:30 – 2:30 P.M. Preparation of ACRS Reports (Open)
Continue discussion of the proposed ACRS reports listed under Item 12.

14) 2:30 – 3:00 P.M. Miscellaneous (Open) (MVB/EMH)
Discussion of matters related to the conduct of Committee activities and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTES:

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

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

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

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
560th FULL COMMITTEE MEETING

March 5-7, 2009

PLEASE PRINT

TODAY'S DATE: March 5, 2009

	NAME	NRC ORGANIZATION
1	Timothy Kolb	NRR/DIRS
2	Greg Bowman	OE
3	Carole Revelle	NRR/DIRS
4	NANCY L. SALGADO	NRR/DIRS/IOLB
5	Howard Benowitz	OGC
6	Mike Boggi	NRR/IOLB
7	John Ridgely	RES/DE/RGDB
8	Autumn Szabo	RES/DRA
9	David Desaulniers	NRO/DCIP
10	Fred Brown	NRC/NRR
11	Sean Peters	NRC/RES/DRA
12	NATHAN SIU	NRC/RES/DRA
13	GARETH PARRY	NRC/NRR/DRA
14	Taesuk Hwang	NRR/DRA (foreign assignee)
15	Ann Ramey-Smith	NRO/DCIP/COLP
16		
17		
18		
19		
20		
21		
22		
23		
24		
25		
26		
27		
28		

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
559th FULL COMMITTEE MEETING

March 5-7, 2009

PLEASE PRINT

TODAY'S DATE: March 5, 2009

	NAME	AFFILIATION
1	Mitchel R. Taggart	PROS
2	Russell Smith	NEI
3	~~Timothy K~~	
4	Todd Newkirk	IBEW
5	Charles Hassler	IBEW Local 94
6	John Butler	NEI
7	Ron Boring	SANDIA
8	April Whaley	Idaho National Lab
9	Katrina Groth	Univ of Maryland
10	Jeffrey Julius	Scientech
11	Andreas Bye	HALDEN PROJECT
12	Pierre LE BOT	EDF R&D
13	Luca PODOFILLINI	PSI, Switzerland.
14	SALVATORE MASSAIU	IFE - NORWAY (HALDEN PROJECT)
15	JOHN FORESTER	SANDIA LABS
16	Vinh DANG	PAUL SCHERRER INSTITUTE
17	Helene PESME	EDF R&D (France)
18		
19		
20		
21		
22		
23		
24		
25		
26		
27		
28		

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
560th FULL COMMITTEE MEETING

March 5-7, 2009

PLEASE PRINT

TODAY'S DATE: March 6, 2008

NAME	NRC ORGANIZATION
Russell Sydnor	RES/DE/DICB
Ian Jung	NRO/DE/ICE2
John N Ridgely	RES/DE/RGDB
Stewart Bailey	NRR/DE
Dan Santos	RES/DE
STELLA OPARA	NSIR/DSP
MONIKA COFLIN	NSIR/DSP
Robert Hardies	RES/DE/CIB
Matthew A. Mitchell	NRR/DCI/CVIB
JOHN LUBINSKI	NRR/DCI
Karl Sturzebecher	NRC/RES
Veronica Rodriguez	NRC/NRR
Stephen Dinsmore	NRC/NRR/DRA
BENJAMIN PARKS	NRC/NRR/DSS
Matthew Panicker	NRC/NRR/DSS/
Gary S. Mizuno	OGC/R+FC
Pat Purtscher	NRC/DCI/CVIB
Jocelyn Mitchell	RES/DSA
[illegible]	RES/DRA
TAREK ZAKI	RES
GARETH PARRY	NRR
JS Hyslop	RES
DONALD DUBE	NRO/DSRA
Yuri Orechwa	NRR/DSS
[illegible]	RES
John Monninger	NRC/RES
Taesuk Hwang	NRR/DRA (foreign assignee)
Sud Basu	RES/DSA

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
561st FULL COMMITTEE MEETING

~~April 2-4, 2009~~ March 5-7, 2009

PLEASE PRINT

TODAY'S DATE: March 6, 2008

NAME	NRC ORGANIZATION
Jared Wermiel	NRC/NRR/DE
Stewart Bailey	NRC/NRR/DSS
Ian Jung	NRC/NRO/DE
David Desaulniers	NRC/NRO/DCIP
DAVID SKEEN	NRC/NRR/DE
Nancy L. Salgado	NRC/NRR/DIRS/IOLB
Allen Howe	NRC/NRR/DORL
Russell Sydnor	NRC/RES/DE
Bill KEMPER	NRC/NRR/DE
Daniel J Santos	NRC/RES/DE
Michael Junge	NRO/DCIP/COLP
William B Kennedy	NRR/DPR/PRTA
WADE Richards	NIST
TED QUAY	NRC/NRR/DPR
Kathryn Brock	NRC/NRR/DPR/PRTA
William Schuster	NRC/NRR/DPR/PRTA
Cindy Montgomery	NRC/NRR/DPR/PRTA
Beth Reed	NRC/NRR/DPR/PRTA
Alexander Adams Jr	NRC/NRR/DPR/PRTA
Duane Hardesty	NRC/NRR/DPR/PRTA
Linh Tran	NRC/NRR/DPR
Charles Boutin	NIST
ANDREA VALENTIN	NRC/RES/DE
Roy MATHEW	NRC/DE
NITIN PATEL	NRC/DE
BILL HORIN	WINSTON + STRAWN

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
560th FULL COMMITTEE MEETING

March 5-7, 2009

PLEASE PRINT

TODAY'S DATE: March 6, 2008

NAME	NRC ORGANIZATION
Kevin Coyne	RES/DRA/PRAB
Don Helton	RES/DRA/PRAB
Stephen Dinsmore	NRR/DRA
See Meng Wong	NRR/DRA

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
559th FULL COMMITTEE MEETING

March 5-7, 2009

PLEASE PRINT

TODAY'S DATE: March 6, 2008

NAME	AFFILIATION
W^m GROSS	NUCLEAR ENERGY INSTITUTE
Stephen Byrne	Westinghouse
James Hall	AREVA
JIN CHUNG	MNES
Jay Amin	Luminant Power
Kevin Holthaus	Omaha Public Power District
Jenny Weil	McGraw-Hill
W Abrien	ISL
Jack Spanner	EPRI
Steven Dolley	Platts
Spyros Traiforos	LINK
KAMIAR JAMALI	NRC/NRO
Rick Grantom	STPNOC
Dean Rategh	SCIentech

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

March 10, 2009

AGENDA
561st ACRS MEETING
APRIL 2-4, 2009

THURSDAY, APRIL 2, 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 Final Safety Evaluation Report (SER) for the Vogtle Nuclear Plant (Open) (JDS/CLB)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff and Southern Nuclear Operating Company (SNC) regarding the Vogtle Nuclear Plant License Renewal Application, 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 – 12:00 P.M. Digital Instrumentation and Control (I&C) Interim Staff Guidances (ISGs) (Open) (CB/CEA)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding Digital I&C ISGs on Highly Integrated Control Room - Human Factors, Licensing Process Issues, 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 – 2:30 P.M. License Renewal Application and Final Safety Evaluation Report for the National Institute of Standards and Technology (NIST) Reactor (Open) (JDS/PW)
   - 4.1) Remarks by the Subcommittee Chairman
   - 4.2) 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 final SER, and related matters.

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

2:30 – 2:45 P.M. ***** BREAK**

5) 2:45 – 4:15 P.M. Draft Final Regulatory Guide 1.211, "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants" (Open) (OLM/CEA)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff and the Nuclear Utility Group on Equipment Qualification (NUGEQ) regarding draft final Regulatory Guide 1.211 and related matters.

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

4:15 – 4:30 P.M. ***** BREAK**

6) 4:30 – 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

6.1) License Renewal Application and Final SER for the Vogtle Nuclear Plant (JDS/CLB)

6.2) Digital I&C Interim Staff Guidances on Highly Integrated Control Room - Human Factors and Licensing Process Issues (CB/CEA)

6.3) License Renewal Application and Final SER for the National Institute of Standards and Technology Reactor (JDS/PW)

6.4) Draft Final Regulatory Guide 1.211, "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants" (OLM/CEA)

6.5) 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" (DCB/HPN)

FRIDAY, APRIL 3, 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. Risk Metrics for New Light-Water Reactor Risk-Informed Applications (Open) (GEA/HJV/HPN)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff regarding risk metrics for new light-water reactor risk-informed applications 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 – 11:00 A.M. Subcommittee Reports (Open)

9.1) Report by and discussions with the Chairman of the Plant License Renewal Subcommittee regarding interim review of the Three Mile Island Unit 1 License Renewal Application and the SER with Open Items that were discussed during the Subcommittee meeting on April 1, 2009 (JWS/CLB)

9.2) Report by and discussions with the Chairman of the Plant License Renewal Subcommittee regarding interim review of the Susquehanna Steam Electric Station License Renewal Application and the SER with Open Items that were discussed during the Subcommittee meeting on April 1, 2009 (WJS/PW)

9.3) Report by and discussions with the Chairman of the Reliability and PRA Subcommittee regarding revisions to NUREG-1855, Appendix A, "Example Implementation of the Process for the Treatment of PRA Uncertainty in a Risk-Informed Regulatory Application," that were discussed during the Subcommittee meeting on March 27, 2009 (GEA/HJV)

10) 11:00 – 11:45 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) 11:45 – 12:00 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.

12:00 – 1:00 P.M. * LUNCH *****

12) 1:00 – 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
- 12.1) License Renewal Application and Final SER for the Vogtle Nuclear Plant (JDS/CLB)
- 12.2) Digital I&C Interim Staff Guidances on Highly Integrated Control Room - Human Factors and Licensing Process Issues (CB/CEA)
- 12.3) License Renewal Application and Final SER for the National Institute of Standards and Technology Reactor (JDS/PW)
- 12.4) Draft Final Regulatory Guide 1.211, "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants" (OLM/CEA)
- 12.5) 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" (DCB/HPN)

SATURDAY, APRIL 4, 2009, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

13) 8:30 – 1:00 P.M. **(10:30-10:45 A.M. BREAK)** Preparation of ACRS Reports (Open)
Continue discussion of the proposed ACRS reports listed under Item 12.

14) 1:00 – 1:15 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 IV

LIST OF DOCUMENTS FROM THE
560TH ACRS MEETING MARCH 5-7, 2009

Agenda Item 2:
Draft Final Regulatory Guide 5.71 (formerly DG–5022), "Cyber Security Programs for Nuclear Facilities"

1. Table of Contents
2. Proposed Agenda
3. Status Reports
4. References
   - RG-5.71 "Cyber Security Programs for Nuclear Facilities" (former DG-5022)

NOTE: This document is a safeguard document and should be treated as such by all members and consultants.

Agenda Item 3:
Draft Final Revisions to 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events"

5. Agenda
6. Status Report
7. Documents
   - Final OMB Supporting Statement Related to Final Rule
   - Final Regulatory Analysis Related to Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (10 CFR 50.61a)
   - Final Rule Related to Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (10 CFR 50.61a)
   - Summary and Analysis of Public Comments on Proposed and Supplemental Proposed Rule Language

Agenda Item 4:
Draft Final Regulatory Guide 1.200 (formerly DG-1200), "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"

8. Table of Contents
9. Proposed Schedule
10. Status Report
11. Attachments:
   - Memorandum dated February 10, 2009, from Michael Case, Director, Division of Engineering, Office of Nuclear Regulatory Research, to Edwin M. Hackett, Executive Director, ACRS, Subject: Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (ML090410042)

Appendix IV

LIST OF DOCUMENTS FROM THE
560TH ACRS MEETING MARCH 5-7, 2009

Agenda Item 7:
Draft Final Regulatory Guide 5.73 (formerly DG-5026), "Fatigue Management for Nuclear Power Plant Personnel"

12. Proposed Schedule
13. Status Report

Agenda Item 8:
International Human Reliability Analysis (HRA) Empirical Pilot Study

14. Table of Contents
15. Proposed Schedule
16. Status Report
17. Attachment:
    - Erasmia Lois, Vinh N. Dang, John Forester, Helena Broberg, Salvatore Massaiu, Michael Hildebrandt, Per Øivind Braarud, Gareth Parry, Jeff Julius, Ronald Boring, Ilkka Männistö, Andreas Bye, "*INTERNATIONAL HRA EMPIRICAL STUDY -PILOT PHASE REPORT* - Description of Overall approach and First Pilot Results from Comparing HRA Methods to Simulator Data," OECD HALDEN REACTOR PROJECT, HWR-844, Rev. 2, October 2008

Regulatory Guide 1.200 Revision 2

Presented to:

Advisory Committee on Reactor Safeguards

Mary Drouin (301-251-7574, mary.drouin@nrc.gov)
Gareth Parry (301-415-1464, gareth.parry@nrc.gov)
US Nuclear Regulatory Commission

March 5, 2009

Purpose of Meeting

- Discuss Revision 2 to Regulatory Guide (RG) 1.200
  - Currently documented as DG-1200 (referred to in presentation as RG 1.200, Revision 2)

- Request letter approving issuance for use

Agenda

- Purpose of RG 1.200
- History of RG 1.200
- RG 1.200
- History of Standards and Industry Guidance
- Staff Endorsement
- Stakeholder Comments
- Schedule and Future Work

Purpose of Regulatory Guide 1.200

- Provides one acceptable approach for determining that the technical adequacy of the PRA is sufficient to support the risk-informed decision-making
- When used in support of an application, should obviate the need for an in-depth review of the PRA by NRC staff
  - Provide for a more focused and consistent review process
- A major technical guidance document in achieving Phase 3 of the staff's phased approach to PRA quality to support risk-informed regulatory activities

History of RG 1.200

- **November 2002**, DG-1122 (draft Revision 0 to RG 1.200) issued for public comment
- **February 2004**, Revision 0 to RG 1.200 issued for trial use
- **September 2006**, DG-1161 (draft Revision 1 to RG 1.200) issued for public comment
- **January 2007**, Revision 1 to RG 1.200 issued for use
- **August 2004**, DG-1138 (draft on staff position on external events) issued for public comment
- **June 2008**, DG-1200 (draft Revision 2 to RG 1.200) issued for public comment
- **March 2009**, Revision 2 to RG 1.200 to be issued for use

Regulatory Guide 1.200 Structure

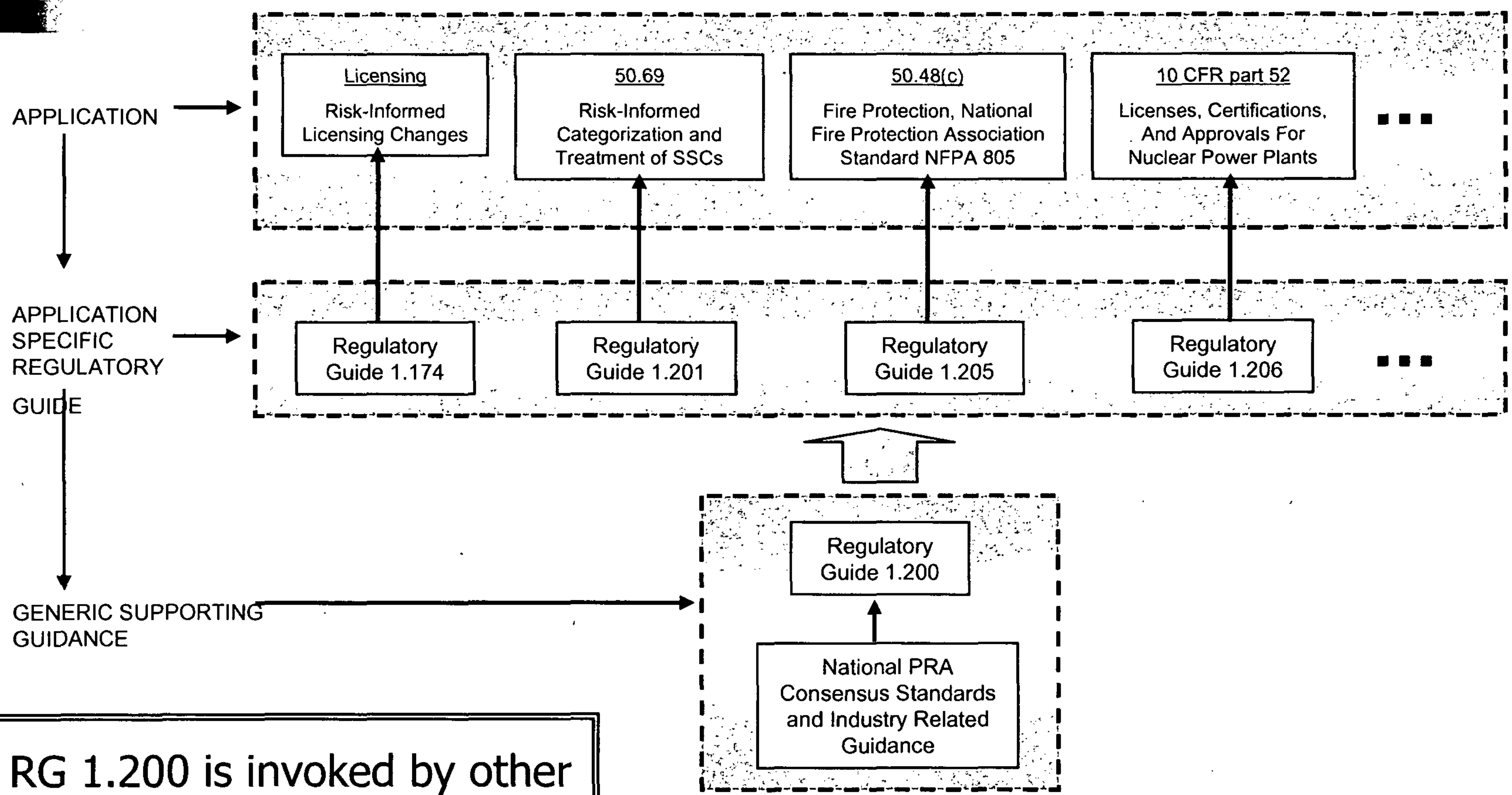
- Main Body
  - Provides staff position on one acceptable approach for what constitutes a technically acceptable PRA
- Appendices
  - Provides staff position (endorsement) on national consensus PRA standards and industry PRA peer review guidance

⇒ ***Majority of staff positions in the main body have not changed since Revision 0 – both NRC and stakeholders, in general, have understanding and are comfortable with the language***

Regulatory Guide 1.200 Content (main body)

- Describes the relationship of RG 1.200 to other guidance documents
- Provides staff position on what constitutes a technically acceptable PRA
- Provides staff position on how to use a national consensus standard and industry peer review in meeting staff position on a technically acceptable PRA
- Provides staff position on demonstrating that the PRA used in regulatory applications is of sufficient technical adequacy
- Provides staff position on the documentation to support a regulatory application

Relationship of RG 1.200 to Other Guidance Documents



⇒ RG 1.200 is invoked by other regulatory guides

Scope of RG 1.200

- Primarily addresses currently operating light water reactors (LWRs), and new LWRs applying for DC and COL
- Addresses CDF, LERF and LRF
- Addresses all plants operating states
- Addresses both internal and external hazard groups

Scope of RG 1.200 (cont'd)

- Provides approach for a technically acceptable PRA, does not provide a staff position on other risk analysis approaches
- Defines PRA

  For a method or approach to be considered a PRA, the method or approach (1) provides a quantitative assessment of the identified risk in terms of scenarios that result in undesired consequences (e.g., core damage or a large early release) and their frequencies, and (2) is comprised of specific technical elements in performing the quantification. A method that does not provide a quantified assessment of the defined risk or does not include the technical elements specified in Regulatory Position 1.2 is not considered to be a PRA.

- Technical acceptability defined in terms of technical elements and their associated attributes and characteristics

RG 1.200: Use of National Consensus Standards and Industry Peer Review

- RG 1.200 allows the use of national consensus PRA standard to demonstrate conformance with the staff's position on what constitutes a technically acceptable PRA
- Standard provides requirements on what a technically acceptable PRA needs to include
  - A peer review is needed to determine if the intent of the requirements in the standard have been met
  - RG 1.200 provides staff's position on what constitutes an acceptable peer review
- Use of a standard has to address the staff's concerns (as addressed in Appendix A to RG 1.200)

RG 1.200: Use of Consensus Standard

- Technical requirements written to different "capability categories"
- Use of the capability categories has caused confusion
- While technical requirements in a PRA may vary, current good practice (Category II) is adequate for majority of applications
- Staff recommends that next revision of the standard address a single category, current good practice

RG 1.200: Peer Review

- NRC has to have confidence in industry peer reviews to achieve a primary purpose of RG 1.200:
  - obviate the need for an in-depth review of the PRA by NRC staff
- Staff position for a technically acceptable peer review addresses:
  - Peer review process
    - Has to be current with both the PRA and the standard
  - Team Qualifications
    - Has to have credibility (e.g., expertise, independence)
  - Documentation
    - Has to document the strengths and weaknesses of the PRA
- Use of a industry peer review process has to address the staff's concerns (as addressed in Appendices B-D to RG 1.200, Revision 2)

History of Standards and Industry Guidance

Standard/Industry Guidance			NRC Endorsement	
Document	**Scope**	**Date**	**Document**	**Date**
ASME RA-S-2002	• At-power • Internal events • Internal flood • CDF and LERF	April 2002	DG 1122	Nov 2002
Addendum A	Same	Dec 2003	RG 1.200, Rev 0	Feb 2004
Addendum B	Same	Dec 2005	DG-1161/RG 1.200 Rev 1	Sep 2006/Jan 2007
Addendum C	Same	July 2007	---	---
ANS 53.21	External hazards	2004	DG-1138	Aug 2004
Revision 1	Same	March 2007	---	----
ASME/ANS RA-S-2008	• Internal hazards • External hazards • CDF and LERF • At-power	April 2008	DG-1200	June 2008
Addendum A	• **Internal hazards** • **External hazards** • **CDF and LERF** • **At-power**	**Feb 2009**	**RG 1.200, Rev 2**	**March 2009**
NEI 00-02	• At-power • Internal events • Internal flood • CDF and LERF	March 2000	RG 1.200, Rev 0	Feb 2004
Revision 1, Self Assessment	same	Nov 2006	RG 1.200, Rev 1	Jan 2007
NEI 05-04, Peer Review Update	• At-power • Internal events • Internal flood • CDF and LERF	Aug 2006	RG 1.200, Rev 1	Jan 2007
Revision 2	**Same**	**Nov 2008**	**DG-1200, RG 1.200, Rev 2**	**March 2009**
NEI 07-12, Int Fire Peer Review	Internal Fire	Dec 2007	DG-1200	June 2008
Draft H	**same**	**Nov 2008**	**RG 1.200, Rev 2**	**March 2009**

RG 1.200: Staff Endorsement of Standards and Industry Guidance

- Staff position categorized as "no objection," "no objection with clarification," or "no objection subject to the following qualification," and defined as follows:
  - **No objection**. The staff has no objection to the requirement.
  - **No objection with clarification**. The staff has no objection to the requirement. However, certain requirements, as written, are either unclear or ambiguous, and therefore the staff has provided its understanding of these requirements.
  - **No objection subject to the following qualification**. The staff has a technical concern with the requirement and has provided a qualification to resolve the concern.
- The staff clarification or qualification to the requirement is indicated in either bolded text (i.e., **bold**) or strikeout text (i.e., ~~strikeout~~); that is, the necessary additions or deletions to the requirement for the staff to have no objection are provided.

RG 1.200: Appendix A Contents

- A-1: General Requirements
- A-2: Internal Events
- A-3: Internal Flood
- A-4: Internal Fire
- A-5: Seismic Events
- A-6: Screening
- A-7: High Winds
- A-8: External Flood
- A-9: Other Hazards
- A-10: Seismic Margins

RG 1.200: Appendix A
Table A-1, General Requirements

- Majority of staff concerns addressed
- Remaining issue on peer review
  - Need to assess the appropriateness of the assumptions
  - Need to review all the applicable requirements
  - Need a minimum list of topics to be reviewed
  - Need to document what was reviewed

RG 1.200: Appendix A
Table A-2 thru A-4, Internal Hazards

- Internal Hazards: Internal events, internal flood, internal fire
- Majority of staff concerns addressed

Remaining issues:

- Internal Events: Failure to repair
  - Data collection and estimation should use both plant-specific and industry data where appropriate
- Internal Flood: Flood-induced failure mechanisms
  - Some level of assessment needs to be included in the analysis
- Internal Fire: equipment selection
  - Supporting requirement needs to state what to do

RG 1.200: Appendix A
Tables A-5 thru A-9, External Hazards

- External Hazards: Seismic, screening and conservative analyses, high winds, external floods, other external hazards, seismic margins
- Majority of staff concerns addressed
- No major "qualifications" remain
- Remaining issue on tornado wind hazard
  - Basic elements of the analysis need to be provided as requirements and not as a "note"
- Seismic margins – staff has not endorsed, outside of scope of RG 1.200

RG 1.200: Appendices B-D NEI Peer Review Guidelines

- Majority of staff concerns addressed
- NEI 00-02 and NEI 05-04
  - Self-assessment performed against ASME RA-Sb-2005
  - Standard has changed since 2005 (e.g., revised requirements, new requirements)
  - PRA may have changed
  - Self-assessment needs to be against both the current PRA and the current standard
- NEI 07-12
  - Peer review needs to be performed using Addendum A to the standard
  - Every applicable requirement needs to be reviewed

Stakeholder Comments on DG-1200

- Majority of comments were of a "technical edit" nature
- Majority of comments were accepted by the staff
- Numerous comments not applicable to the RG

Major outstanding industry issues:

- Do not finalize until fire and external hazard parts of the standard have been fully piloted
- Acceptability of seismic margin as a seismic PRA
- Self-assessment is historical
- Assessment of non-routine activities
- Level of expertise for tornado hazard analysis
- Use of bounding for fire scenarios for Capability Category II
- Independence of peer reviewers

Schedule and Future Work

- Commission due date of March 31, 2009, for Revision 2 of RG 1.200
- Other technical concerns are being addressed by ASME and ANS
  - Addressed in either future addendum or revision to ASME/ANS RA-Sa-2009
- **PRA** Standards under development
  - Low power shutdown
  - Level 2
  - Level 3
  - New Reactors
  - Advanced non-LWRs
- RG will continue to be revised/updated to stay current with published standard

Alternate Fracture Toughness Requirements for Protection against Pressurized Thermal Shock (PTS) Events Rule (10 CFR 50.61a)

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150–AI01

[NRC–2007–0008]

Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events

AGENCY: Nuclear Regulatory Commission.

ACRS Full Committee Meeting
March 5, 2009

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

Rulemaking Working Group
Alternate PTS Rule

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

• Barry Elliot	NRR/DCI
• Matthew Mitchell	NRR/DCI
• Stephen Dinsmore	NRR/DRA
• Lambros Lois	NRR/DSS
• Veronica Rodriguez	NRR/DPR
• Mark EricksonKirk	RES/DE
• Robert Hardies	RES/DE
• Nihar Ray	NRO/DE
• Geary Mizuno	OGC

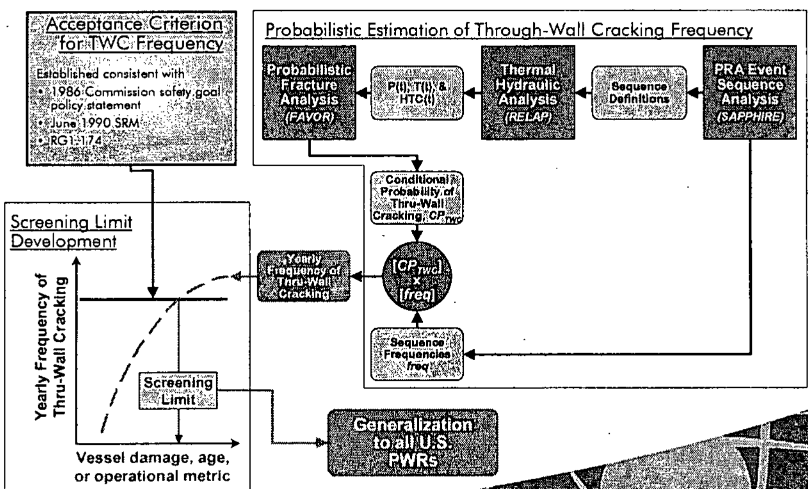
Agenda
Alternate PTS Rule

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

- Main Topics:
  - Technical Basis for the Rule
  - Generalization Study
  - Current PTS Rule and motivation for developing the Alternate PTS Rule
  - Alternate PTS Rule

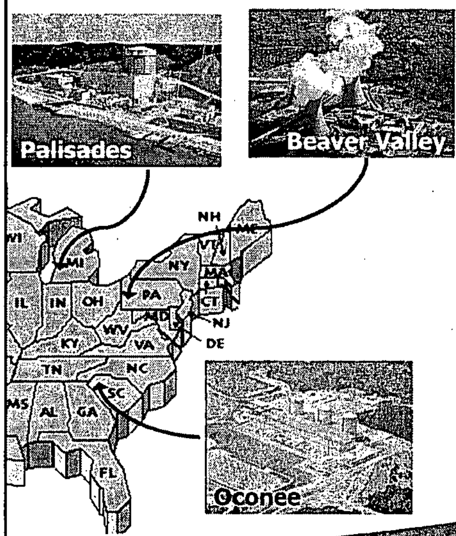
Technical Basis
Overall Model

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



Technical Basis
Detailed Study Plants (Baseline)

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



- Detailed analysis of 3 pressurized water reactors (PWRs)
  - All PWR manufacturers
    - 1 Westinghouse (W)
    - 1 Combustion Engineering (CE)
    - 1 Babcock & Wilcox (B&W)
  - 1 plant from original (1980s) PTS study
  - 2 plants very close to the current PTS screening criteria
- Generalization to all PWRs
  - Characteristics of materials and transients that dominate failure frequencies
  - Examination of 5 more high embrittlement PWRs

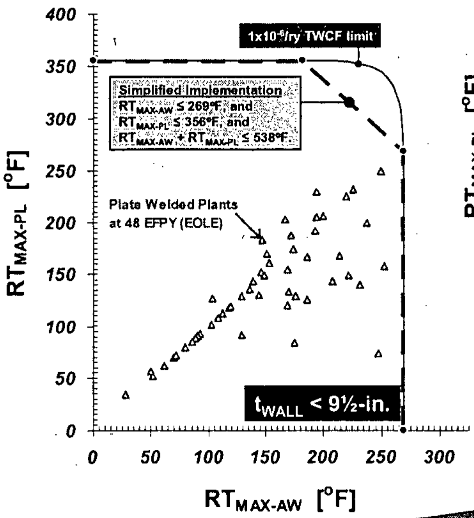
Technical Basis
Summary of Findings – 3 Study Plants

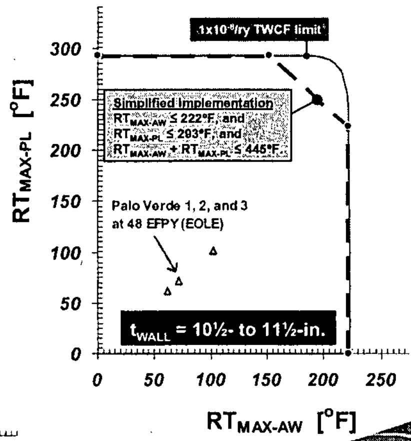
U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Only the most severe transients modeled contribute to risk
  - The characteristics of these transients are similar across the operating PWR fleet
  - Operator actions, while accounted for in our analysis, are not important for the scenarios that dominate RT_{MAX} limits
- Axial flaws, and their associated material properties, dominate risk
- Study plant results support development of embrittlement-based through-wall cracking frequency (TWCF) estimation formulae useful for all plants

Technical Basis

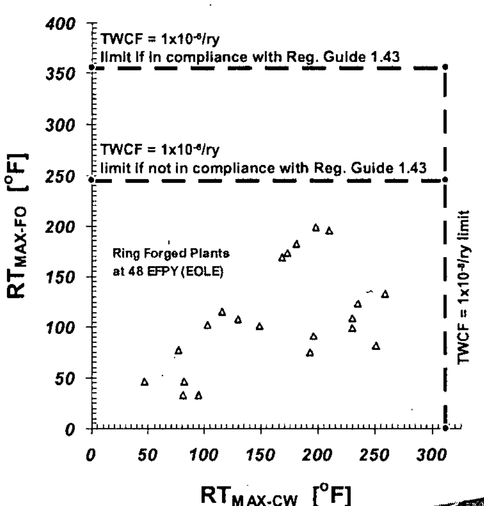
RT Limits – Implementation for Plate Plants



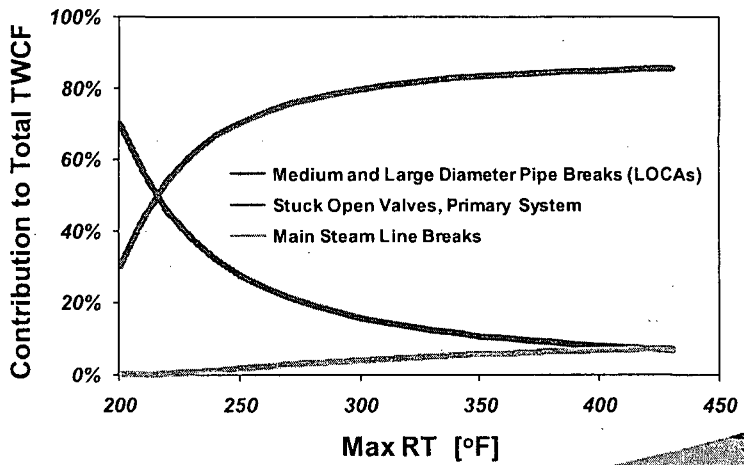


Technical Basis

RT Limits – Implementation for Forgings



Technical Basis
Important Transient Classes



Generalization Study
Methodology

- Original detailed study - 3 plants
  - Beaver Valley (W – 3 Loop)
  - Oconee (B&W)
  - Palisades (CE)
- Chose 5 more high embrittlement plants
  - Salem, Unit 1 (W – 4 Loop)
  - Three Mile Island, Unit 1 (B&W)
  - Fort Calhoun (CE)
  - Diablo Canyon, Unit 1 (W – 4 Loop)
  - Sequoyah, Unit 1 (W – 4 Loop)
- Questionnaire used to collect information on the 5 additional plants

Generalization Study

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

Medium and Large LOCAs

- Dominate risk at higher embrittlement
  (75% contributor at new RT-limits)
- Failures driven by factors that are similar across fleet
  - Rate of cooling of the primary system water exceeds that achievable by the reactor pressure vessel (RPV) wall, so the transient severity depends on:
    - Steel thermal conductivity
    - Vessel diameter and thickness

    Not by the thermal hydraulic (TH) characteristics of the transient (i.e., is vessel-limited)
  - Emergency core cooling systems operate automatically. Therefore operator actions do not play a role in these transients
- These factors suggest generalization is possible

Generalization Study

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

Stuck-Open Primary Valves

- Dominate risk at low embrittlement
- Failures driven by factors that are similar across the fleet
  - Low reactor coolant temperatures at time of re-pressurization
  - Re-pressurization to the safety valve setpoint
- Rapid operator action (i.e., high pressure injection (HPI) throttling) can influence this scenario; however, even if credit for operation action was removed, the screening criteria will not change
- These factors suggest generalization is possible

Generalization Study

Main Steam Line Breaks

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

- Slight effect at very high embrittlement
- Failures driven by factors that are similar across the fleet
  - Rate of cooling of the primary system water exceeds that achievable by the RPV wall
  - Temperature in primary cannot fall below 212°F because of secondary side interaction.
- Failures, if they occur, happen before operator action is probable
- These factors suggest generalization is possible

Background

Current PTS Rule – Technical Summary

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

- The current PTS rule, 10 CFR 50.61, has provided a sound, conservative methodology for ensuring adequate protection from PTS events since its promulgation in 1985

- However,10 CFR 50.61 is fundamentally based on 1980s technology and is not based on the best available information and analyses regarding potential RPV failure due to PTS

Background
Current PTS Rule – Regulatory Summary

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

- The level of conservatism in 10 CFR 50.61 imparts a degree of unnecessary regulatory burden on licensees when compared to our best current understanding of PTS events and the risks they pose
- Under 10 CFR 50.61, approximately 8 to 12 operating PWRs would not meet the screening criteria of the rule through 60 years of operation

Background
Alternate PTS Rule – Objectives

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

- The objectives of the alternate PTS Rule, 10 CFR 50.61a, include:
  - Adequate protection of public health and safety
  - Regulatory efficiency, effectiveness, and openness
  - Remove unnecessary regulatory burden

Alternate PTS Rule
Overview

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

- 10 CFR 50.61a structured similarly to 10 CFR 50.61
- Similarity emphasized to facilitate implementation by both the industry and the NRC staff
- Differences between the two rules reflect critical features

Alternate PTS Rule
Key Features

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

- Key features of 10 CFR 50.61a include:
  - Limitations on applicability
  - Less restrictive screening criteria
  - Evaluation of plant-specific flaw distributions
  - Implementation of new embrittlement models and RPV surveillance data evaluations

Alternate PTS Rule

Limitations on Applicability

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

- Technical basis for 10 CFR 50.61a based on evaluation of currently operating PWR designs
- New reactor designs may by subject to different PTS event frequencies or severities – hence, 10 CFR 50.61a not explicitly applicable
- Improvements in RPV manufacturing expected to obviate need for application of 10 CFR 50.61a to new reactors

Alternate PTS Rule

Less Restrictive Screening Criteria

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

- Modified material property parameter (RT_{MAX}) used instead of RT_{PTS}
- Technical basis for 10 CFR 50.61a demonstrates that PWR facilities can safely operate to higher levels of RPV embrittlement
- Hence, less restrictive screening criteria implemented in 10 CFR 50.61a

Alternate PTS Rule

Plant-Specific Flaw Distributions

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

- Less restrictive screening criteria in 10 CFR 50.61a due, in large part, to use of a more realistic flaw distribution in technical basis development
- Important to verify facilities implementing 10 CFR 50.61a are consistent with this assumption
- Requirements established to evaluate data acquired via ASME Code-required inservice inspections to verify plant-specific flaw distribution

Alternate PTS Rule

Embrittlement Models and Surveillance Data

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

- 10 CFR 50.61a embrittlement models based on:
  - a significantly enhanced RPV surveillance database
  - a combined statistical analysis of data and a mechanistic understanding of radiation embrittlement
- Enhanced RPV surveillance data evaluations:
  - are more statistically rigorous
  - ensure embrittlement models are not behaving non-conservatively

Alternate PTS Rule
Conclusion

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

- 10 CFR 50.61a provides:
  - an effective option for those facilities projected to exceed the screening criteria of 10 CFR 50.61, while
  - ensuring that adequate protection of public health and safety is maintained based on the rule's thorough, state-of-the-art technical basis and specific requirements incorporated in the rule, where necessary, to ensure facility compliance with the rule's technical basis

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

RG-5.71
Cyber Security Programs for Nuclear Facilities
(DG-5022)

Karl Sturzebecher
Digital Instrumentation and Controls Branch
Division of Engineering
Office of Nuclear Regulatory Research

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

Agenda

- RG-5.71 Development
- Technical Approach
- Path Forward
- Backup Slides
  - Comment Response
  - NUREG/CR 6847

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

RG-5.71 Development

New Rule 10 CFR 73.54

- Protection of digital computer and communication systems and networks from cyber attacks
  - Safety-related and important to safety functions
  - Security functions
  - Emergency preparedness functions
  - Support systems, which if compromised, impact above

- Approved by Commission 1/09

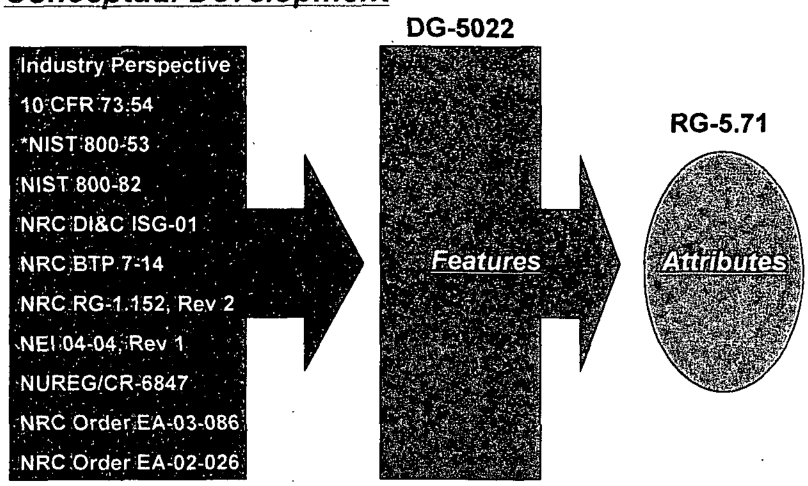
- Anticipate OMB approval April/May

3

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

RG-5.71 Development

Conceptual Development



***Merger of IEC 15408 (Parts 1-3) and IEC 17799**

4

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

RG-5.71 Development

Stakeholder Comments

- Participation by NERC, FERC, DHS, NIST, Joe Weiss, vendors, licensees, NEI
- 7/11/08 Stakeholder Meeting (208 comments)
  - High number of questions, assumptions, move and delete comments
- 12/4/08 Stakeholder Meeting (14 comments)
  - Cyber security plan needs to be clearer
  - Should leverage existing NRC/industry regulations, programs, and processes
  - Should use a graded approach
  - Physical and logical security boundaries do not have a one-to-one correspondence
- 1/12/09 Stakeholder Meeting (6 comments)
  - Reorganize document to discuss plan first, next program, then security controls
  - Emphasize performance-based attributes
- 2/11/09 Stakeholder Meeting (final closure)

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

Technical Approach

Time Frame	Security Engineering Paradigm	Technical Environment
1960s – 1970s	COMPUSEC – computer security COMSEC – communications security	Digital mainframes Analog communications
1980s – mid 1990s	INFOSEC – information security	Distributed computing LANs Digital communications
Mid 1990s – today	Cyber security -Management controls -Operational controls -Technical controls	Convergence of computing and telecommunications Advances in digital technology, ASICS, PLDs, FPGAs, etc.

Cyber security: combination of : (1) inherent technical features and functions that collectively contribute to a system, system of systems, and enterprise achieving and sustaining confidentiality, integrity, and availability, and (2) implementation of standardized operational and management controls that define the nature and frequency of interaction between users, systems, and system resources, the purpose of which is to achieve and sustain and known secure state at all times, and prevent accidental and intentional theft, destruction, alteration or sabotage of system resources.

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

Technical Approach

Purpose of RG-5.71

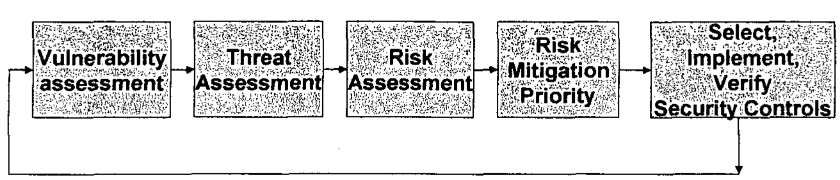
- Per 10 CFR 73.54 establish performance based requirements to ensure that the functions of critical systems and critical digital assets are protected from cyber attack throughout the system engineering lifecycle, using a graded approach

7

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

Technical Approach (3.1, 3.9)

- **Vulnerability**
  - Inherent weakness in a system, system of systems, or enterprise, its design, implementation, operation, or operational environment
- **Threat**
  - Potential for a vulnerability to be exploited, accidentally or intentionally, a function of the opportunity, motive, expertise, and resources (OMER) needed and available to effect the exploitation
- **Risk**
  - Likelihood of a vulnerability being exploited and a threat instantiated, plus the worst-case severity consequences

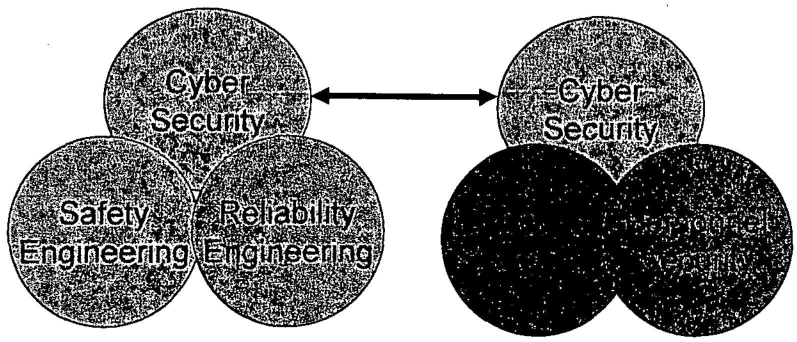


Ongoing throughout lifecycle

8

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

Technical Approach (3.4.1.2)

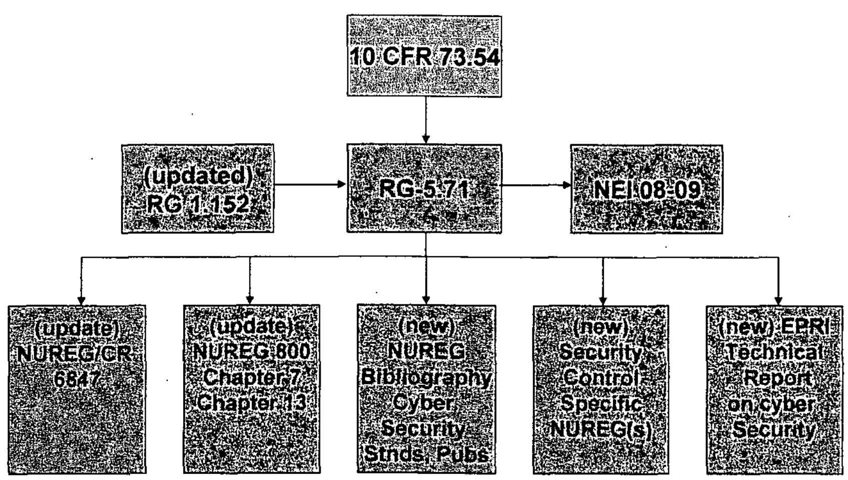


3.4.1.2 The licensee should perform concurrent security engineering lifecycle activities, to achieve high assurance that safety, reliability, and security engineering activities are coordinated.

9

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

Technical Approach



10

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

Technical Approach (3.4)

Cyber Security Plan
↓
Cyber Security Program
↓
Security Controls

Management Controls	Operational Controls	Technical Controls
-System and service acquisition (vendor supply chain) -System engineering lifecycle activities -Risk management -Policies and implementing procedures	-System hardening -Media protection -Physical and operational environment protection -Personnel security -System and information integrity -Attack mitigation -Configuration management	-Access control -Audit and accountability -System and communication protection -Identification and authentication -Defense in depth
Management of people, processes, and business activities that could impact the security of a Product, service, or system	Control interaction among people, technology, and operational environment to ensure system remains in known secure state	Inherent security features and functions implemented through technology

11

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

Technical Approach

Performance based

- RG-5.71 specifies attributes ("what") for which applicant must demonstrate high assurance
- Cyber security plan, policies, and implementing procedures specify details ("how"), along with applicable NUREGs
- Rationale:
  - Security architecture is site specific, tied to each system, its design, implementation, operation, and operational environment
  - Security engineering is a concurrent engineering activity, ties into existing system engineering methodology and business practices
  - Rapid evolution of cyber security technology
  - Constantly changing attack methods and threat environment
  - Security sensitive information doesn't belong in a public document
  - Approach is similar to other federal security rules and NERC cyber security standards
- " ...defense technologies are widely available to mitigate threats but have not been uniformly adopted due to associated costs, perceived need, operational requirements, and regulatory constraints."
  - Director of National Intelligence Annual Threat Assessment, provided to U.S. Senate Select Committee on Intelligence, 2/12/09, p. 39.

12

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

Technical Approach (3.6)

Most Common Categories of Exploits (accidental or intentional)

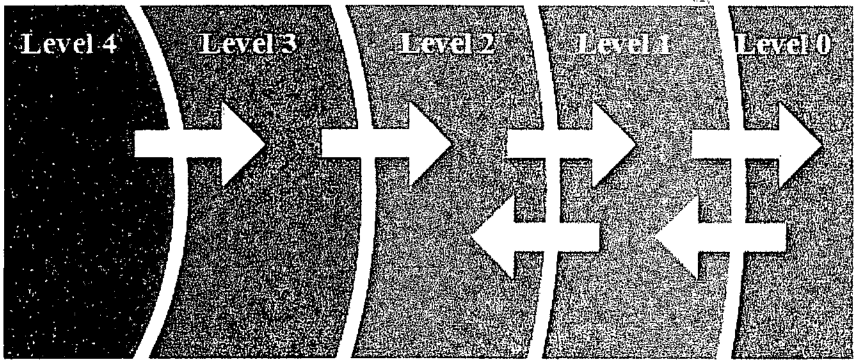
-Action, command, response triggering -Blocking access to system resources -Browsing, surveillance (pre-cursor event) -Corruption of resource management information -Deletion of information -Denial of service, network flooding, system saturation, lack of capacity planning -EMI/RFI -Environmental, facility, power faults or tampering -Illegal operations, transactions, modes/states -Inference, aggregation -Insertion of bogus data or commands -Lack of contingency planning, back-ups	-Masquerading, IP spoofing -Modification of information or commands -Lack of fault tolerance, error detection or correction -Overwriting information or commands -Password guessing, spoofing, compromise -Replay, reroute, misroute messages -Site or system specific vulnerabilities -Theft of information or service -Trojan horse -Unauthorized access or use of system resources -Uncontrolled, unprotected portable systems, media, archives, hardcopies -Unpredictable COTS behavior -Virus, worm, zombie, bot net

13

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

Technical Approach (3.5)

An example of such a defensive architecture is one that includes a series of concentric defensive levels of increasing security



Security Architecture: Concentric Ring Model

14

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

Technical Approach (3.5)

ISO/OSI Reference Model	**Sample Protocols**	**Sample Security Controls**
7: Application Layer	FTP, HTTP, SMTP, SNMP, Telnet, APIs	Prohibit use of Telnet, require HTTPS, Digital certificates, system hardening
6: Presentation	Context and syntax management	**Information hiding**
5: Session	Session management and Synchronization	Digital certificates
4: Transport	TCP, UDP	**Peer entity authentication**
3: Network	IP, X.25, ATM	IPSec, **partitioning, wrappers**
2: Data Link	IEEE 802.3, Frame relay	Asymmetric block encryption
1: Physical	V.90, OC-3, SONET, RS-422	Electrically isolate signals, channels, etc.

Defense in depth strategy: apply multiple different technical and operational security controls to all layers of the protocol stack.

15

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

Technical Approach

Sample Implementation of Technical Controls

Access Control 3.4.3.1	Authentication 3.4.3.4
• Domain and type enforcement	• Biometrics
• Least privilege	• Data origin
• Wrappers	• Digital certificate
• Role based	• Kerberos
• Time based	• Unilateral
• Origin based	• Mutual
• Encryption	• Peer entity
• Information hiding	• Smart cards
• Partitioning	• Non-repudiation of origin, receipt
Arbitrate initiator request (person or process) to perform an operation on a target resource	Establish the claimed identity of a user, process, device, or other entity

16

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

Technical Approach (3.3)

Incorporating the Cyber Security Program into the Physical Protection Program

10 CFR 73.54(b)(3) security program a component of the physical protection program

- Security organization is responsible for protecting the facility from physical and cyber attacks up to and including the design-basis threat
- Align key personnel who are responsible for the management and oversight of the licensee's cyber security program
- Flexibility in regard to solid line/dotted line reporting chain

17

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

Path Forward

RG-5.71 Next Steps

- Respond to ACRS comments
- Complete development of generic cyber security plan template NEI-08-09
- Conduct licensing reviews
- Develop and implement oversight process

Requesting ACRS letter endorsing issuance for use

18

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

Backup: Comment Response

- **Cyber security should not be located in the physical security organization.**
  - Response: The rule, specifically 10 CFR 73.54(b)(3) requires this. However, we understand this concern and have allowed flexibility in regard to the dotted line/solid line reporting structure between cyber and physical security.
- **Need to ensure that cyber security requirements are carried forward all through the supply chain.**
  - Response: We will add "..including all suppliers, vendors, and maintenance contractors." to the end of the first bullet under 3.4.1.1. We will reword the second bullet under 3.4.1.1 to read "...vendor, supplier, and maintenance security and development lifecycles."
- **Need to emphasize the importance of configuration management, especially during hardware/software upgrades.**
  - Response: We believe the configuration management requirements stated in 3.4.1.2, which references Chapter 7 of the SRP and BTP-14, 10 CFR 54, 10 CFR 59, and section 3.10 of this document, address this concern.
- **Need to add more definitions in the glossary.**
  - Response: The additional definitions provided in this slide set will be added to the glossary.

19

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

Backup: Comment Response

- **Need to include more examples and diagrams**
  - Response: The new diagrams and tables provided on slides 8-11 and 13-16 will be added to the document.
- **Need to emphasize the deliberate exploitation of vulnerabilities.**
  - Response: This point has been added to slides 8 and 13, which will be added to the document.
- **Need to add acceptance criteria**
  - Response: The burden of proof that a security control or set of controls is acceptable and meets the high assurance test lies with the applicant. That said, a security control would be considered acceptable if:
    - The security control selected is appropriate for the vulnerability it is intended to mitigate.
    - The implementation, configuration, operation, and execution of the security control are sufficiently robust and resilient to mitigate the threat of the vulnerability being exploited.
    - The implementation, configuration, operation, and execution of the security control are consistent with industry best practices, national and international consensus standards, applicable NUREGs, site specific policies and procedures, and the due diligence criteria.
    - The security control is consistent and compatible with the overall site security architecture
  - [This statement will be added as the third paragraph in Section 3.4.]

Due diligence: (Black's Law Dictionary) such a measure of prudence, activity, or assiduity, as is properly to be expected from, and ordinarily exercised by, a reasonable and prudent person under the particular circumstances, not measured by any absolute standard, but depending on the relative facts of the special case.

20

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

Backup: NUREG/CR 6847

Conduct Assessment Planning and Select the Multidisciplinary Assessment Team

↓

Stage 1: Examine Plant-Wide Cyber Security Practices

↓

Stage 2: Identify Critical Digital Assets

↓

Stage 3: Conduct Cyber Security Vulnerability Analysis

↓

Stage 4: Conduct Assessments of Susceptibility

↓

Stage 5: Conduct Risk Assessment Activities

↓

Stage 6: Conduct Risk Management Activities

↓

Make Decisions and Take Actions to Enhance the Cyber Security of the Critical Digital Assets

↓

Incorporate the Method into an Ongoing Cyber Security program

Summary of Overall Conclusions

ACRS PRA Full Committee Meeting
March 6, 2009

Presented by
Gareth Parry

Confirmation of the Value of Simulator Exercises

- Provide insights on issues related to human behavior that can be used to improve HRA methods:
  - Understanding of the context for the crew actions, including its dynamic aspects (HFE 2A and 5B1)
  - Recognizing the significance of crew to crew variability
  - Identifying potential failure mechanisms
- HRA methods take these into account to varying degrees
  - HEPs in the context of a PRA represent the average taken over the aleatory variables (e.g., specific crew) and bounding or representative context (e.g., plant conditions)
  - Crew to crew variability is not considered for most methods
  - Many methods do not explicitly consider failure mechanisms

Variability in Quantitative Results

- Despite the care taken to provide a detailed description of the scenarios, the HEP evaluations show significant variability, though less than in the ISPRA study
- The variability was present for both the easy (e.g., HFE 4A) and the difficult (e.g., HFE 1B and HFE 5B1) HFEs
- Some variability is explainable by differences in assumptions
- Some outlier estimates (e.g., HFE 1A for ATHEANA) are understood
- The variability is not correlated across the HFEs (i.e., the highest HEP for different HFEs is not from the same analysis)

Understanding the Sources of Variability

- Variability should not be unexpected since the methods have very different bases, including:
  - Identification of failure mechanisms at a fairly detailed level (e.g., ATHEANA, MERMOS, CBDT)
  - Identification of generic failure types (e.g., CREAM, HEART)
  - Task analysis (e.g., THERP, ASEP)
  - PSF approaches (e.g., SPAR-H)
- The different methods <u>require</u> differing degrees of qualitative analysis and depth of understanding of human behavior

Understanding the Sources of Variability (cont'd)

- Factors affecting variability include:
  - The capability of method to capture the significant influences on behavior
  - Inherent pessimism or optimism of the method
  - Whether the method has been applied as intended
  - The team experience in HRA and with the method applied
  - The degree of expertise in human performance needed to apply the method
  - The depth of qualitative analysis undertaken to understand the underlying dynamics of the scenario and factor it into the estimation

Differences in Qualitative Assessment

- The nature of the qualitative analysis required is different from method to method
  - At one extreme, the qualitative assessment is focused on identifying failure mechanisms, including the contextual factors that drive them
  - At the other, it is focused on determining the strength of a PSF
- The guidance for performing a qualitative assessment that is systematic and thorough appears to be inadequate for most methods
  - Leads to lack of reproduceability and traceability

Initial Findings

- All methods, but particularly the more simplified methods, need additional guidance on:
  - How to develop the qualitative analysis
    - What are the operational conditions that could influence performance and how will they relate to the procedures, training, etc.
    - Guidance on how to discuss these issues with operators and trainers to obtain important information
  - How to judge the level of PSFs
  - How to document the quantification
- Many methods need to include additional PSFs to adequately predict crew performance

Use of Study Results to Address SRM

- The results of this study, and the follow-on for the LOFW scenarios, will be used to identify the strengths and weaknesses of the various methods in order to:
  - Identify methods that are acceptable for specific regulatory uses
  - Identify improvements to those methods
  - Identify limitations on their use
  - Development of potential hybrid methods
  - The required skill set for application of the methods

Predicted HEPs and Quantitative Comparisons

ACRS Full Committee Meeting

March 6, 2009

presenter: V.N. Dang, PSI, Switzerland

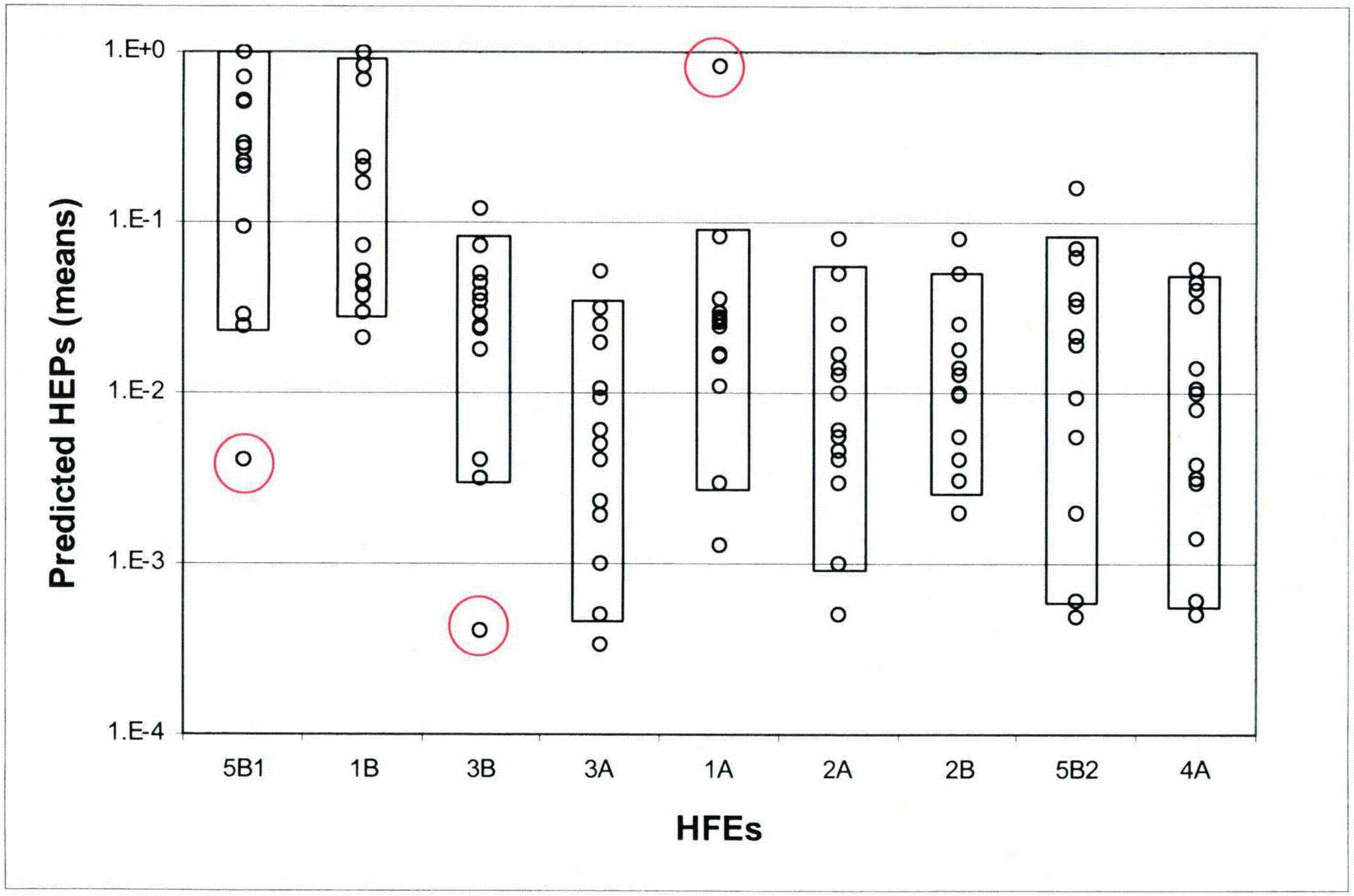
Empirical HRA Study Results
"Quantitative" comparisons

- Two types of quantitative comparisons between HRA predictions and empirical data
  1. predicted ranking vs. empirical ranking
  2. predicted HEPs vs. empirical HEPs

- Significant limitations of quantitative results, especially the small set of observations
  - Quantitative comparisons supplement the qualitative comparisons and insights
  - The overall evaluation of the HRA methods is based on both qualitative and quantitative insights
  - However, the qualitative insights should be weighted more strongly

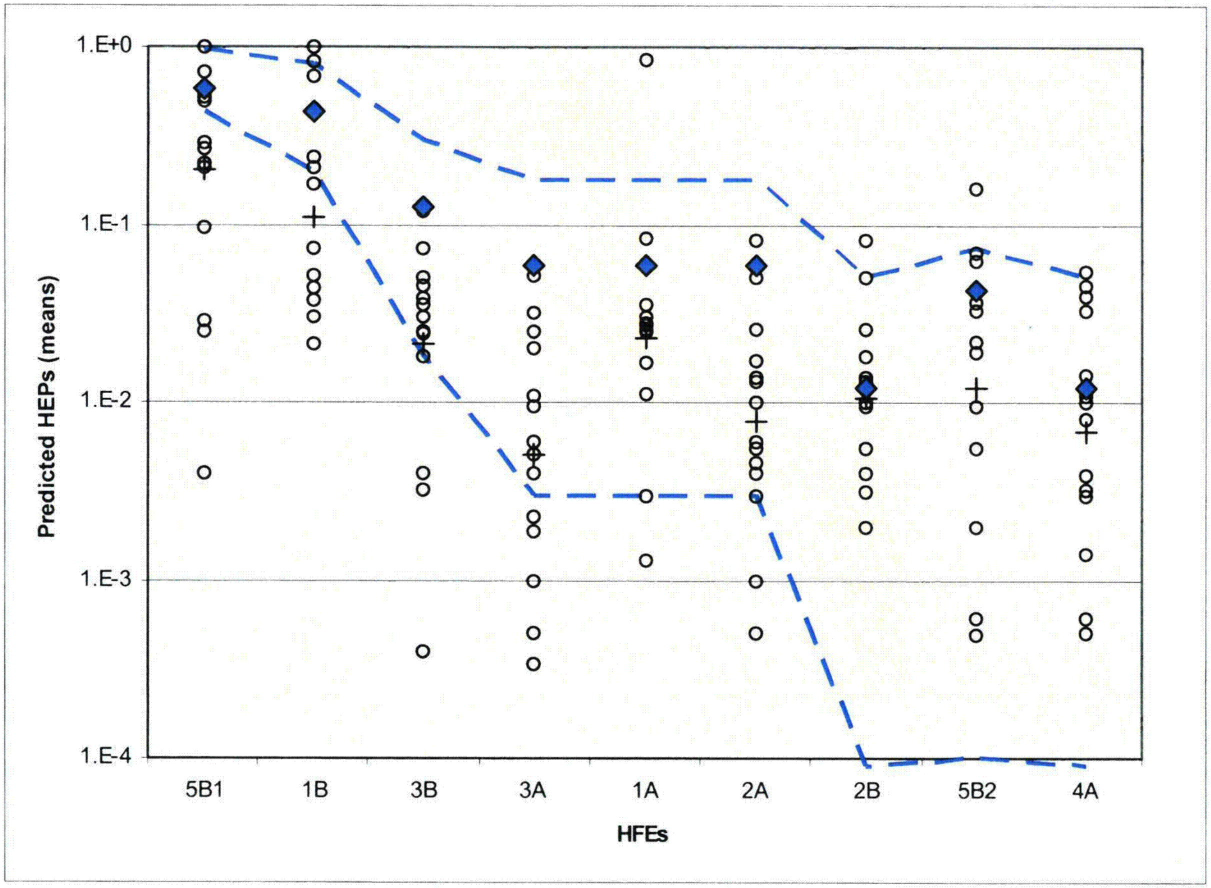
Range of predicted mean HEPs

Boxes drawn around range, 1 maximum value and 1 minimum value excluded from each range.



- **After exclusion, most ranges span < 2 orders of magnitude**
- **Many outliers relatively close to the range.**
- **Exceptions are highlighted.**
- **These are being examined to determine causes**
  - **method or assumption or combination**

Predicted HEPs vs. empirical HEPs (Bayesian results)



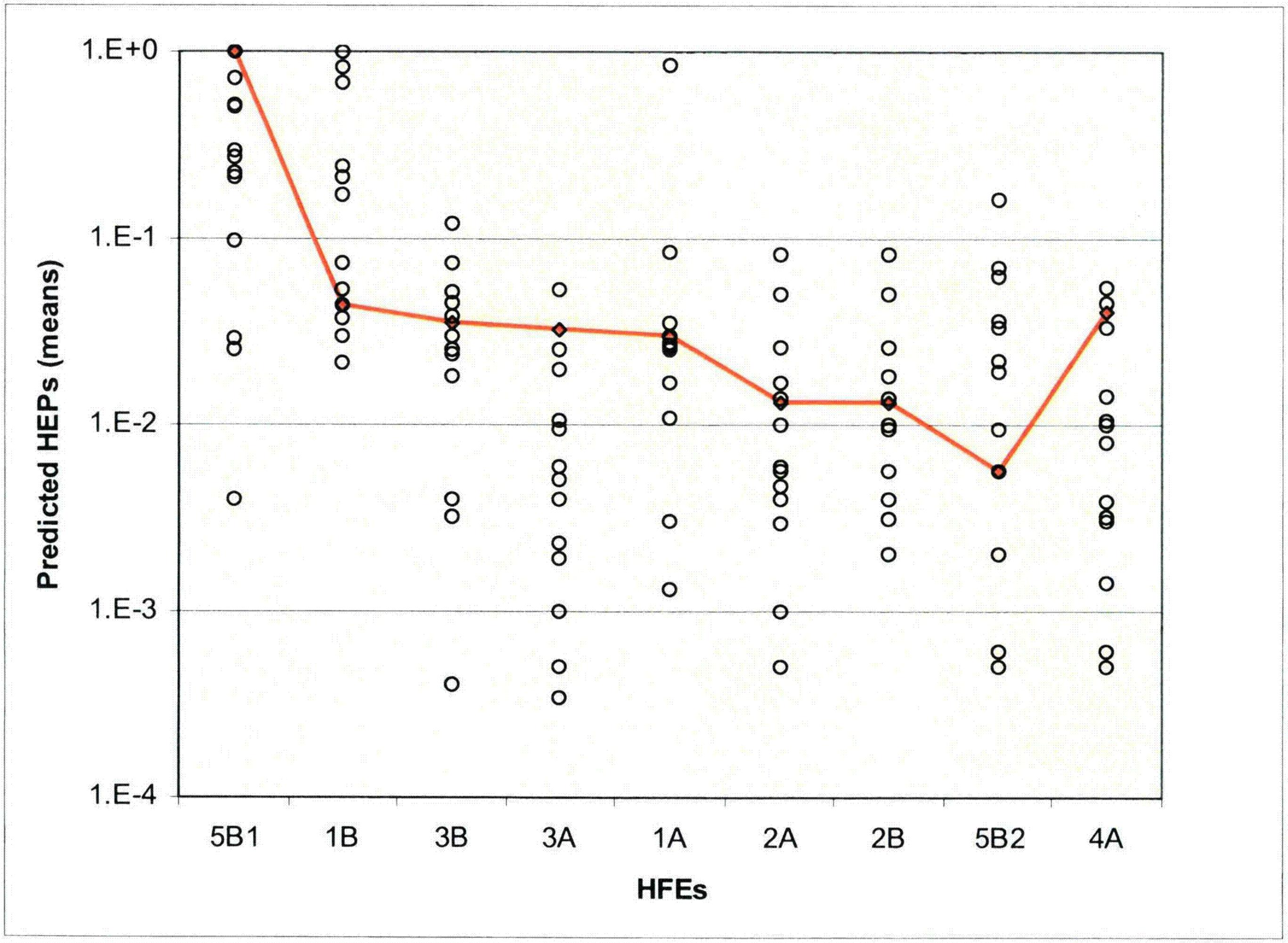
- **Many methods underestimated HEPs of difficult HFEs** (5B1 and 1B, also 3B, 3A)
- **Rest of HFEs: nearly all predictions (mean values) fall within bounds, but very broad bounds**
- **Consistency of predicted ranks**
  - **(by individual method)**
  - **separate, important criterion for HRA methods**

Breadth of Bayesian confidence bounds are due to small data set.

This show limits of comparisons based on empirical (Bayesian) HEPs.

Consistency of predictions against difficulty

"Difficulty" ranks combine failure counts and qualitative analysis of observations



Conclusions : quantitative comparisons

- Some quantitative data, subject to significant uncertainties, was obtained. A Bayesian analysis was performed.
- Empirical reference data (qualitative and quantitative)
  - a unique aspect of this study
  - allowed comparisons and evaluations of the methods that were not possible in previous benchmarks.
- Caution is warranted in interpreting the quantitative comparisons results
  - uncertainties in this reference data
  - the limited number of HFEs
- Qualitative evaluation remains the more valuable outcome of work so far

  quantitative comparison "rounds out" the overall evaluations of the methods

Due to limitations of the quantitative data, one should not select among the HRA methods based on these quantitative comparisons.

Qualitative Analysis and Comparison with Halden Crew Data

ACRS PRA Full Committee Meeting

March 6, 2009

Presented by

John Forester

Sandia National Laboratories

UNITED STATES NUCLEAR REGULATORY COMMISSION

Overview

- How we performed the qualitative analysis
  - Summarized each HRA method's qualitative predictions of crew performance on each human failure event (HFE)
  - Compared with crew data
  - Summarized results of the comparison and addressed performance characteristics of the method
- General findings

Summarized HRA Team Qualitative Results

- Translated driving factors (PSFs) identified from method results into common set of PSFs identified as relevant for this study
  - Different methods address different PSFs
  - Capture the factors predicted to affect crew performance in a common terminology (Summary Table of Driving Factors)
- Summarized their "operational expressions or stories"
- Assessment team **had not** seen the Halden results
- Summaries were sent to each team for review
  - Addressed any team comments

Summary Table of Driving Factors

EPRI Caused Based Decision Tree (CBDT) Assessment of HFE 5B1

PSF	Comments	Influence
Adequacy of time	Inadequate given the available cues and when another opportunity to check pressurizer pressure comes up in the procedures	-2 (MD)
Time pressure		0
Stress		0
Scenario complexity		0
Indication of conditions	Since pressurizer pressure will not immediately show pressure to be decreasing, it was assumed that the crew would not immediately recognize the stuck open PORV. The PORV position indication shows closed even though it is open.	-2(MD)
Execution complexity		0
Training		0
Experience		0
Procedural guidance	Crew does not get another direction to check pressure (when it will be meaningful) until they enter ECA-3.1, which is expected to be too late given the time frame.	-2(MD)
HMI		0
Work processes		N/A
Communication		N/A
Team dynamics		N/A
Scale for PSF Rating	*N= Generally good, 0= Not a driver, -1=minor negative driver, -2 = strong negative driver, MD= Main negative driver, N/A= not addressed by the method*	

Compared HRA Team Predictions to Halden Crew Data

- Compared PSF ratings from HRA teams to those in crew data (for each HFE)
  - Did they agree on positive and negative PSFs and the driving factors
- Compared operational expressions/stories
  - Did their operational story agree with what occurred in the crew data?
- Compared ranking of HRA team human error probability (HEP) predictions to HFE difficulty rankings
  - Did the HEPs (rankings) correspond to the way the crews performed?

Summarized the Results of the Comparison – Addressed Performance Characteristics of the Method

- Discussion of predictive power of the method
  - Identification of driving factors
  - Predicted operational expression
  - Assessment of HEPs
- Insights on guidance and traceability
- General conclusion on the strengths and weaknesses of the method based on the comparison
- Insights produced by the method for error reduction

General Findings

- All methods identified some of the important factors that drove crew performance
  - But most HRA analyses failed to identify important factors for some HFEs
- Several methods significantly over- or underestimated the difficulty of some HFEs
  - Many methods tended to make optimistic predictions for the more difficult HFEs
  - Some methods provided relatively pessimistic values for the easier HFEs
- Factors driving variability between HRA team predictions:
  - The depth of the analysis to develop qualitative understanding
  - The performance shaping factors (PSFs) used by the method
  - Judgments of the degree of influence of the PSFs

Backup slides

Example of Specific Finding

- Inconsistency in interpretation of the diagnosis component of a human action.
  - Some methods (e.g., SPAR-H, ASEP, THERP, CBDT) have an explicit diagnosis component.
  - Seems to have been interpreted as a high level "figure out what the accident is" activity
    - It's an SGTR – ignored diagnosis for later HFEs in scenario
  - Lower level cognitive activities, such as interpreting the plant status in the context of the step by step procedure are not being given enough attention
  - Simulator experiments show this to be important
- Can hurt the predictive power of the method

Difference (DT+ASEP - Halden)

PSF	HFE2A	HFE3A	HFE4A	HFE2B	HFE3B	HFE5B1	HFE5B2
Adequacy of Time							
Time Pressure							
Stress	-2	1	-2	-1	-1	-2	-2
Scenario Complexity	2	-1	-1	1	1	3	-1
Indications of Conditions	-3			-3			
Execution Complexity		-1	-3		-1	-3	-1
Training						1	
Experience							
Procedural Guidance	-1	-3	-3	-3	-3	1	
Human- Machine Interface	-1			-1			
Work Processes							
Communication							
Team Dynamics							

DT+ASEP > Halden - Method optimistic
DT+ASEP = Halden - Agreement
DT+ASEP < Halden - Method Pessimistic
Not covered

Difference in Rating for PSFs Between Two Applications of SPAR-H

Difference (NRC - INL)

PSF	HFE2A	HFE3A	HFE4A	HFE2B	HFE3B	HFE5B1	HFE5B2
Adequacy of Time			2				
Time Pressure							
Stress	1	1	1				
Scenario Complexity					1	2	1
Indications of Conditions						2	
Execution Complexity				-1		-1	
Training	1	1	1				
Experience	1	1	1				
Procedural Guidance					-1	-1	-1
Human- Machine Interface	-1	-1	-1				
Work Processes							
Communication							
Team Dynamics							

NRC > INL

NRC = INL

NRC < INL

Not covered

The International HRA Empirical Study
Phase 2 Empirical Results

Presentation to:
Advisory Committee on Reactor Safeguards
Rockville, MD - 6 March 2009

Salvatore Massaiu
Andreas Bye
Industrial Psychology Division
OECD Halden Reactor Project

Summary

- A new data analysis approach has been developed that:
  - optimizes the comparison of HRA methods predictions to empirical observations
  - improves the usefulness of simulator data for HRA purposes
- The experiment identified:
  - the extent of crew-to-crew variability and the significance of teamwork factors
  - the importance of events dynamics in determining crew performance

Data analysis process

- From individual observations to HRA reference data

Observations - raw data
- Audio/video
- Simulator logs
- Interviews
- On-line performance ratings (OPAS)
- On-line comments
- Crew self-ratings
- Observer ratings

Crew-HFE performances
- HFE success/failure
- Description of operation
- Observed PSFs

Overall-HFE performances
- Aggregated operational descriptions
- Aggregated PSFs

Aggregated operational descriptions - HFE 2B

Mode	Crews	Result	Deviation/comment
These crews only use SG PORVs (as dump is not available due to steam line isolation)	A, C, D, F, I, K, L, M, N.	Cooldown completed in 5-7 minutes	- Crew D enters step 7 already meeting the table conditions and does not need to cool down. - Crews A and C do not cool down at maximum speed (7:10 and 8:05 respectively)
Wait for completion of local actions for isolation before starting step 7 (wait 2 to 6 minutes)	B, G, J.	Full ruptured SG at start of cooldown, but normal cool down time (5-7 min.)	- Crews J does not cool down at maximum speed (7:25)
These crews tried to use steam dump, forgetting the steam line isolation. Afterwards they used the SG PORVs	E, H.	Cool down in 9:12 (E) and 11:50 (H).	

Aggregated PSFs - HFE 2B

PSF	Comment	Rating
Time pressure	No time pressure for almost all crews	0
Stress	Signs of stress carried over from the previous phase in 2 crews with difficulties	-1
Scenario complexity	Some crews encountered difficulties in understanding why the dump was not working	-1
Indication of conditions		N
Execution complexity	Some crews had problems with operating the SG PORVs at maximum or setting them correctly upon completion	-1
Training	Generally good training on cooldown	N
Experience	Experience level did not differentiate between performance levels	0
Procedural guidance	The crews typically based their cooldown strategy on following the procedure.	N
HMI		N
Work processes	Mostly thorough work. Some minor issues on not reading notes and warnings for 2 crews who exhibited operational difficulties.	N
Communication	Normally good communication. Some problems in information exchanges for the less performing crews (also captured under the PSF 'Team dynamics')	N
Team dynamics	Less well performing crews (but also some well performing) showed lack of adequate leadership and support (e.g. SSs too involved, too passive), and/or lack of coordination and discussion. 3 crews waited too long for local actions, 4 other teams with poor team dynamics performed less well.	-1(md)

Legend: md = main driver, -1 = negative driver, 0 = not a driver, N = nominal-positive driver

HFEs difficulty and ranking

HFE	Near misses	Failures	Difficulty	Comment on difficulty
1A	-	1/14	…	…
2B	4	0/14	…	…
5B1	-	7/7	Very difficult	[…] the RCS pressure for 6 out of 7 crews was increasing when applying E-3 step 18 ("check RCS pressure – increasing", the step directly after the end of depressurization) or at least stable when applying step 19 ("Check if SI flow should be terminated"). Only after the HFE time window clearer indications of RCS leakage will appear to the crew. This is the most difficult HFE of this set.

HFEs Ranking: 5B1 > 1B > 3B > 3A > [1A, 2A, 2B] > 5B2 > 4A

Conclusion

- The reference data (empirical results format) allow the comparison of both factorial and scenario-based HRA methods:
  - Accounts for the dynamic nature of crew performance (operational descriptions, observational PSF ratings)
  - Not a mere table comparison (operational details, comments on what and why)
- Significant crew-to-crew variability:
  - Strong interaction with process dynamics
  - Importance of teamwork factors
  - Procedures do not cover all situational variations in detail

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Status Report: Benchmark Study of HRA Methods Using Control Room Simulator Data

Erasmia Lois, PhD
Human Factors and Reliability Branch
Division of Risk Analysis

Presentation to:
Advisory Committee on Reactor Safeguards
6 March 2009

Study Objectives

- Assess HRA methods and HRA practices in light of NPP control room simulator data
  - Characterize the methods
  - Identify strengths and weaknesses in predicting simulator results
  - Provide the technical basis for improving HRA methods and their application
  - Support addressing Commission direction on HRA

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

The Issue

- Differences in the underlying frameworks, data, and quantification algorithms of HRA methods yield different human error probabilities and different insights regarding the potential drivers of error/failure
  - Models are based on formal and informal theories of error but have not been tested with empirical data.
  - In regulatory applications sensitivity analyses are used to enhance the robustness of HRA results.

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

NRC Efforts to Improve HRA

- Development of improved guidance (NUREG-1792, *HRA Good Practices*, and NUREG-1842, *Evaluation of HRA Methods Against the Practices*)
- Interactions with national and international experts to pursue testing and benchmarking of HRA methods
- Support for Halden Reactor Project (HRP) initiative to invite signatory organizations to participate in an effort to “benchmark” HRA methods on the basis of simulator data
- Data collection which includes events and simulator data produced at the Halden simulator

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

Industry Efforts to Improve HRA

- EPRI HRA Users Group founded in 2000
  - Charter is to promote consistency between users, and Promote convergence of methods
  - Developed a software tool with EPRI & NRC methods
- Working with the NRC
  - Comments on HRA guidance documents (NUREG-1792, *HRA Good Practices*, and NUREG-1842, *Evaluation of HRA Methods Against the Practices*)
  - Halden Benchmarking Project – Assessment & HRA Analysis Teams
  - Joint Fire HRA Project (good blueprint for new HRA projects)
    - Will brief ACRS soon
  - Collaborative work on SRM on HRA models
    - Will brief ACRS in June

Recommendations and Directions

- ACRS recommendations to compare the fundamental assumptions of HRA models used by both the NRC and industry
- Commission direction to ACRS to "work with staff and external stakeholders to evaluate the different human reliability models in an effort to propose either a single model for the agency or guidance on which models should be used in specific circumstances," Nov 8, 2006
- Recent Commission direction "to continue to pursue possibly working with EPRI, INPO, and/or international partners to test U.S. nuclear plant operating crews' performance in a variety of situations and keep the Commission informed on the progress in developing a human reliability analysis (HRA) database and benchmarking projects," February 18, 2009

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

Tasks of the HRA Empirical Study

Scenario Definitions
OECD Halden, Assessment Group

HRA Analyses
Analysis teams

Summary of Predictions
Assessment Group

crews – one power plant

Simulator Sessions: Observation, Data Collection
OECD Halden

Data Analysis
OECD Halden

Comparison: HRA Predictions vs. Crew Data
Assessment Group

Development of Insights to Improve HRA Methods and Practices

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

Status of the HRA Empirical Study

- Simulation runs performed at HRP during Nov-Dec 2006
  - Two steam generator tube rupture (SGTR) and two loss of feedwater (LOFW) scenarios
  - Selected four human actions for each SGTR scenario and two human actions for each LOFW scenarios
- Phase 1: Pilot Completed in 2008
  - Used two SGTR human actions to establish the method
  - Established method for crew performance data collection in a format suitable to HRA
  - Established method for evaluating HRA results
  - Compared HRA results to HRP data for the two human actions
  - HRP report (HWR-844), to be published as NUREG/IA-0216, Spring 2009
  - Pilot had external review (Barry Kirwan)

Status of the HRA Empirical Study

- Phase 2: Data collection and analysis of seven SGTR human actions
  - Completed and hold meeting with participants, 3/2-4/2009
    - 14 HRA groups participated (12 domestic and foreign organizations)
    - HRA teams were open to the idea of improving their methods/practices on the basis of evaluation/feedback
    - Discussed the idea of a hybrid method or "tool box approach"
    - External reviews participated in the meeting
      - comments very constructive and positive for the study

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

Status of the HRA Empirical Study

- Phase 2 (cont)
  - ACRS briefing, March 2009
  - Document in draft HRP report, September 2009, and in NUREG/IA, September 2010
- Phase 3: Analysis and documentation of the LOFW human actions
  - Most teams have submitted their analysis of LOFW human actions
  - HRP has completed simulator observations.
  - Now comparing analytical results to HRP data, due Summer 2009
  - International meeting on LOFW results scheduled for Dec 2009
  - Documentation of Phase 3 results expected in Spring 2010
- Complete documentation of the study by Sept 2010

Fatigue Management

Mitchel Taggart
Vice President of PROS

PROFESSIONAL
REACTOR
OPERATOR
SOCIETY

PROS advocates safety first

- We feel, as an industry, we have done that with respect to our outage performance over the last several years
- Rule will make utilities change the way they schedule their man power on the front end and back end of an outage
- Utilities will change a successful past practice and could impact our ability to safely execute an outage

First Issue

- Unit Outage – currently defined as reactor unit disconnected from the grid
- PROS recommended change: up to one week prior to disconnecting the reactor unit from the grid and up to 75% turbine power following reconnection to the grid
- This would require a rule change

Advantages Beginning of Outage

- Crew adjustment to outage schedule and themselves (new crews)
- Preparation/Familiarization in the simulator prior to commencing the shutdown (real time simulator shutdown vice just snap shots at key stages)
- Being prepared helps keep the stress level as low as possible in the Control Room (stress will never be eliminated, but being prepared keeps it in check)

Advantages After the of Outage

- Allow major equipment to be tested/placed in service prior to releasing support personnel
- Sufficient personnel available to handle the given emergent issues that can occur following an outage
- The better a site deals with outage recovery (resolve emergent issues) the fewer challenges experienced during the cycle

PROS is not proposing any change in the work hour allowance of 60 days provided in part 26.205 (d)(4) of the ruling for outages.

Second Issue – Multiunit Sites

- Outage Unit – currently defined as only the reactor that is disconnected from the grid
- PROS recommended change: outage unit would be changed to outage site so multiple unit facilities with combined control rooms would be able to modify all the site personnel schedules to accommodate the outage.
- This would require regulatory guide change

Advantages

- The site can manage work hours using a single set of rules

- More personnel supporting the outage – would need 50% more operators on operating unit to comply with reg guide

- Critical outage activities will have adequate oversight and maintain present outage proficiency – reg guide would result in less people to support the outage

- Communications between work groups will be better if working the same schedule – minimizes stress which is a major contributor to fatigue

Draft Final Regulatory Guide 5.73 "Fatigue Management for Nuclear Power Plant Personnel"

Valerie Barnes, Ph.D.
Senior Technical Advisor for Human Factors
Office of Nuclear Regulatory Research
March 6, 2009

Topics

- History of Regulatory Guide (RG) 5.73
- Overview of Subpart I and draft final RG
- Areas of substantive disagreement with NEI 06-11, Rev. 1
- Implementation and publication schedules

2

History of RG 5.73

- 1982 NRC policy on work hours
- 2002 Rulemaking plan approved
- 2005 Proposed rule published
- 2006 Public meetings on RG began
- 2008 Final rule published March 31
- 2008 DG-5028 published for comment in October
- 2009 Fatigue management provisions must be implemented October 1

3

Subpart I: Managing Fatigue

- Applies only to plants in operational phase
- Requires a fatigue management program for everyone with unescorted access to protected areas
  - Training
  - Self-declaration procedures
  - Fatigue assessments

4

Additional Requirements for "Covered Workers"

- Work hour controls required for
  - Maintenance (risk-significant activities only)
  - Operations (risk-significant activities only)
  - Chemistry (ERO only)
  - Health Physics (ERO only)
  - Fire Brigade (effects of fire suppressants)
  - Security (armed)
  - Individuals who direct risk-significant maintenance and operations

5

Work Hours Controls for "Covered Workers"

- Work hours scheduling
- Work hour limits
- Rest break requirements
- Minimum days off

6

Minimum Days Off Requirements

Cumulative fatigue – the increase in fatigue over consecutive sleep-wake periods resulting from inadequate rest.

7

Minimum Days Off Requirements

Vary according to:

- Plant state (operating or outage)
- Shift duration (8, 10, or 12 hours)
- Job duties
  - maintenance
  - operations, health physics, chemistry, fire brigade
  - security

8

Minimum Days Off Requirements
(Normal Operations)

- In each shift cycle, an average of:
  - 1 day off/week for 8-hour shifts
  - 2 days off/week for 10-hour shifts
  - 2-3 days off/week for 12-hour shifts, by job duties
    - maintenance: 2
    - operations, HP, chemistry, fire brigade: 2.5
    - security: 3
- Days off must be distributed to provide at least 1 day off in any 9-day period

9

Minimum Days Off Requirements
(Unit Outages)

Maintenance	1 day off in any 7 days
Operations, HP, chemistry, & fire brigade	3 days off in each non-overlapping 15-day block
Security	4 days off in each non-overlapping 15-day block

10

Minimum Days Off Requirements
(Unit Outages)

- Covered workers are limited to 60 consecutive days of outage scheduling
- Covered workers are subject to outage controls "while working on outage activities"

11

Draft Final RG 5.73

- Revisions related to two areas of substantive disagreement with NEI 06-11, Rev. 1
- Additions and clarifications to NEI 06-11, Rev. 1, in response to public comments and new guidance

12

Areas of Disagreement

- Both relate to applying the minimum days off (MDO) requirements
  - Concept of periodic overtime
  - Applicability of MDO requirements to operators working on the operating unit(s) at a multi-unit site with one or more units in an outage

13

Periodic Overtime Guidance in NEI 06-11, Rev. 1

- Plan a shift schedule that would include the required MDO
- Permit unscheduled work hours, as needed
- At least quarterly, identify individuals who have averaged >54 work hours/week
- Review the circumstances, determine if a schedule change is needed going forward, document in the corrective action program

14

Overview of Staff Position on Periodic Overtime

- Could lead to violations - guidance in NEI 06-11, Rev. 1, if implemented as written, would
  - Exclude unscheduled work hours when determining the applicable MDO requirements [10 CFR 26.205(d)(3)]
  - Permit covered workers to miss required days off without a waiver [10 CFR 26.207]
- Unnecessary - regulation includes flexibility for periods of unscheduled work hours

15

Scheduled vs. Actual Hours Worked

- Rule addresses shift scheduling requirements separately from work hours controls
- MDO requirements are one set of work hours controls
- Work hours include any duties performed for the licensee, whether the hours are scheduled or unscheduled

16

Flexibility for Emergent Work Provided in Work Hour Limits

- Rule permits covered workers to work as many as
  - 16 hours in any 24-hour period
  - 26 hours in any 48-hour period
  - 72 hours in any 7-day period

17

Flexibility for Applying MDO Requirements

- MDO requirements can be met based on average daily work hours over a period of up to 6 weeks
- Licensees have the flexibility to transition between shift cycle lengths to accommodate emergent work
- Over a shift cycle, a covered worker may work
  - an average of 9 hours/day, 6 days/week = 54/week
  - an average of 11 hours/day, 5 days/week = 55/week
  - for some categories of personnel, 13 hours/day, 5 days/week = 65/week
- Licensees have the flexibility to distribute these extra work hours as necessary to accommodate emergent work

18

Flexibility for Granting Waivers

- Licensees are permitted to grant a waiver of the work hour controls to prevent or mitigate conditions adverse to safety or security
- Licensees are not permitted to allow individuals to miss required days off without a waiver

19

Draft Final RG 5.73

- Retains staff position in DG-5026 that the concept of periodic overtime is unnecessary and, if implemented as written, could cause licensees to violate the regulation
- Includes more detailed guidance on methods to implement the MDO requirements that the staff would find acceptable

20

Operator Work Hours during Outages at Multi-Unit Sites

- Rule permits fewer days off during outages "while working on outage activities"
- Rulemaking intent - any time covered workers perform duties on or for an operating unit, they would be subject to operating MDO requirements

21

Regulatory Guide Relaxation

- In DG-5026, staff agreed that maintenance and some operations personnel working on common systems are eligible for relaxed outage work hour controls
- Disagreed that the licensed operators responsible for an operating unit can be considered "working on outage activities"

22

Policy Bases for Staff Position

- NRC policy and guidance on control room operations
  - 1989 "Policy Statement on the Conduct of Nuclear Power Plant Operations"
  - 2008 Regulatory Guide 1.114, "Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit"

23

Regulatory Basis for Staff Position

- 10 CFR 50.54(m)(2)(i) establishes minimum shift staffing requirements for ROs and SROs on an operating unit based on
  - the number of units
  - control room configurations
- 10 CFR 50.54(k) requires one "operator at the controls" of any operating unit
- 10 CFR 50.54(m)(2)(iii) requires a "senior operator in the control room" of any operating unit

24

NEI 06-11, Rev. 1, and Staff Positions on Operator Outage MDO

- NEI 06-11, Rev. 1 – One RO and one SRO per operating unit should have operating MDO with short-term relief provided by operators who have been working outage hours

- Staff – The "operator at the controls" and the "senior operator in the control room," and the operators who relieve them (depending on number of units and control room configurations) should have operating MDO

25

Minimum Number of Individuals Per Shift Working Nonoutage Schedules for Onsite Staffing of Operating Nuclear Power Units during Outages[1]

Number of operating nuclear power units [2]	Position	Two-unit site		Three-unit site				
				Two control rooms				
		One Control Room	Two Control Rooms	Single Control Room Unit in Outage	Single Control Room Unit and One Unit Served by Dual Control Room in Outage	One of the Two Units Served by Dual Control Room in Outage	Two Units Served by Dual Control Room in Outage	Three Control Rooms
One	Senior Operator	2(2)	2(2)		2(2)		2(2)	2(2)
	Operator	2(3)	2(3)		2(3)		2(4)	2(4)
Two	Senior Operator			2(2)		3(3)		3(3)
	Operator			3(4)		4(5)		4(5)

[1] Numbers in parentheses are minimum shift complement required by 10 CFR 50.54(m)
[2] For the purpose of this table, a nuclear power unit is considered to be operating when it is connected to the grid.

26

Revised Staff Position in Draft Final RG 5.73

- Draft final RG modifies the staff's position in DG-5026 by
  - Relaxing requirements for operators transitioning onto the outage unit to provide long-term relief
  - Permitting operators working outage hours to provide short-term relief for operators on the outage unit under certain circumstances
  - Clarifying that the back-up operators for the operating unit may work on outage activities, except the operator at the controls and the senior operator in the control room required by regulation

27

Implementation and Schedule for Publishing Final RG 5.73

- Fatigue management requirements must be implemented no later than October 1, 2009
- Staff published the draft final RG to support licensee preparations
- Final RG will be published no later than May 31, 2009

28

Discussion

29