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

January 17, 2007

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

Dear Chairman Klein:

SUBJECT: SUMMARY REPORT - 538th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, DECEMBER 7-8, 2006, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 538th meeting, December 7-8, 2006, 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 Graham B. Wallis, Chairman, ACRS:

- Draft Final Regulatory Guide DG-1145, Combined License Applications for Nuclear Power Plants (LWR Edition), dated December 12, 2006

LETTERS

Letters to Luis A. Reyes, Executive Director for Operations, NRC, from Graham B. Wallis, Chairman, ACRS:

- Proposed Revision to Standard Review Plan Section 13.3, "Emergency Planning," dated December 15, 2006
- Draft Final Regulatory Guide 1.207 (DG-1144), "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors," dated December 18, 2006

MEMORANDA

Memoranda to Luis A. Reyes, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS:

- Proposed Revisions to Standard Review Plan Sections in Support of New Reactor Licensing, dated December 15, 2006
- Anonymous Letter Concerning Changes to 10 CFR Part 52 Rulemaking Package (SECY-06-0220), dated December 8, 2006

R/4

HIGHLIGHTS OF KEY ISSUES

1. Draft Final Regulatory Guide, DG-1145, "Combined License Applications for Nuclear Power Plants"

The Committee met with representatives of the NRC staff to discuss DG-1145, "Combined License Applications for Nuclear Power Plants." DG-1145 provides a roadmap to help Combined License (COL) applicants identify the appropriate content of a COL application submitted under 10 CFR Part 52. It is structured to address COL applications that reference an early site permit (ESP), a certified design, neither, or both. DG-1145 is meant to be consistent with proposed final revisions to 10 CFR Part 52 and with the new and revised Regulatory Guides and Standard Review Plan (SRP) Sections being developed in support of new reactor licensing. DG-1145 was based on the guidance previously published in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," and was updated based on operating experience (as reflected in NRC Bulletins and Generic Letters), lessons learned from design certification and ESP reviews, and guidance in SECY Papers and related Staff Requirements Memoranda. There was extensive industry involvement in the development of DG-1145. The members and staff held a detailed discussion on the appropriate content of a COL application related to probabilistic risk assessments (PRAs).

Committee Action

The Committee issued a report to the NRC Chairman on this matter dated December 12, 2006, recommending that the final rule, 10 CFR Part 52, retain the requirements that a design-specific PRA be submitted with the design certification application and that a plant-specific PRA be submitted with the COL application. The Committee also recommended that DG-1145 be issued as a final Regulatory Guide after the staff ensures that it is consistent with the final 10 CFR Part 52 rule and with the Regulatory Guides and SRP Sections/Chapters being revised or developed in support of new reactor licensing.

2. Draft Final Regulatory Guide 1.207 (DG-1144), "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors"

The Committee met with representatives of the NRC staff, the American Society of Mechanical Engineers (ASME), and AREVA to discuss the draft final Regulatory Guide 1.207 (DG-1144), "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors." The staff and representatives from Argonne National Laboratory (ANL) described the objective, technical basis, and regulatory positions of Regulatory Guide 1.207. This Guide uses an environmental correction factor, F_{en}, to account for the effects of the light water reactor coolant environment on fatigue life. The technical basis for this Guide is described in NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials." Regulatory Guide 1.207 also addresses the non-conservatism of the current ASME stainless steel air design curve in the mid-to-high cycle fatigue range by recommending the use of a new stainless steel air design curve in NUREG/CR-6909. Representatives of ASME and AREVA commented that the existing ASME design curves and methodology are adequate, that there is no need for a new regulatory guide, that the new guide will result in more detailed and costly

Supplemental Application. The Committee decided that it was satisfied with the NRR's response.

- The Committee considered the EDO's response of December 6, 2006, to comments and recommendations included in the November 27, 2006 ACRS Report on the Safety Aspects of License Renewal Application for the Palisades Nuclear Plant. The Committee decided that it was satisfied with the EDO's response.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from November 4, 2006 through December 6, 2006, the following Subcommittee meetings were held:

- Future Plant Designs - November 30, 2006

The Subcommittee reviewed DG-1145, "Combined License Applications for Nuclear Power Plants."

- Thermal-Hydraulic Phenomena - December 5, 2006

The Subcommittee discussed the development of the TRACE computer code.

- Materials, Metallurgy, and Reactor Fuels - December 6, 2006

The Subcommittee reviewed draft final Regulatory Guide 1.207 (DG-1144), "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors."

- Planning and Procedures — December 6, 2006

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

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The Committee would like to be informed of any significant changes made to DG-1145 prior to publishing it in final form.

- The Committee would like to be kept informed of any significant changes made to the SRP Sections, prior to issuing them in final form, listed in the December 15, 2006 memorandum from John T. Larkins, Executive Director, ACRS to Luis A. Reyes, Executive Director for Operations, NRC.

- The Committee is awaiting receipt of additional high priority SRP Sections from the staff.

- The Committee would like to be informed of the disposition of the anonymous letter related to changes to 10 CFR Part 52 rulemaking.

- The staff committed to provide the Committee with a description of research activities associated with fatigue degradation mechanisms.

- The Committee plans to continue its review of the State-of-the-Art Reactor Consequence Analysis Project as further progress has been made by the staff.

PROPOSED SCHEDULE FOR THE 539th ACRS MEETING

The Committee agreed to consider the following topics during the 539th ACRS meeting, to be held on February 1-3, 2007:

- Final Review of the Power Uprate Application for the Browns Ferry Nuclear Plant, Unit 1
- Final Review of the License Renewal Application for the Oyster Creek Generating Station
- Development of TRACE Thermal-Hydraulic System Analysis Code
- Proposed Revision to 10 CFR 50.46 LOCA Criteria for Fuel Cladding Materials
- Draft Final Revision 1 to Regulatory Guide 1.189 (DG-1170), "Fire Protection for Nuclear Power Plants," and SRP Section 9.5.1, "Fire Protection Program"
- Wolf Creek Pressurizer Weld Flaws
- Proposed Revisions to Regulatory Guides and SRP Sections in Support of New Reactor Licensing

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

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/RA/

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Chairman

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ADAMS Accession Number: ML070080277

OFFICE	ACRS/ACNW	ACRS/ACNW	ACRS/ACNW
NAME	CSantos	MSnodderly	FGillespie
DATE	01/09/07	01/11/07	01/17/07

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OFFICE	ACRS/ACNW	ACRS/ACNW	ACRS/ACNW
NAME	CSantos CS	MSnodderly MRS	FGillespie
DATE	1/9/07	1/11/2007	

From: Carol Brown
To: Banerjee, Maitri; Barnard, Ethel; Bates, Andrew; Caruso, Ralph; Champ, Billie; Flack, John; Hammer, Charles; Junge, Michael; McKelvin, Sheila; Mike, Linda; Nourbakhsh, Hossein; Perry , Jamila; RidsAslbpMailCenter; RidsEdoMailCenter; RidsNmssOd; RidsNroOd; RidsNrrOd; RidsOcaaMailCenter; RidsOcaMailCenter; RidsOgcMailCenter; RidsOpaMail; RidsRgn1MailCenter; RidsRgn2MailCenter; RidsRgn3MailCenter; RidsRgn4MailCenter; RidsSecyMailCenter; RidsStpMailCenter; Santos, Cayetano; Sosa, Belkys; Thornsbury, Eric
Date: 01/24/2007 5:00:25 PM
Subject: SUMMARY REPORT - 538th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

LETTER TO: The Honorable Dale E. Klein, Chairman

SUBJECT: SUMMARY REPORT - 538th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, DECEMBER 7-8, 2006, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

DATED: January 17, 2007

ADAMS Accession Number: ML070080277

Carol Anne Brown
Administrative Assistant
Advisory Committee on Reactor Safeguards
Operations Support Branch
415-7998, MS T2-E26

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

March 2, 2007

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

FROM: Graham B. Wallis
ACRS Chairman

Graham B. Wallis

SUBJECT: CERTIFIED MINUTES OF THE 538TH MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), December 7-9, 2006

I certify that based on my review of the minutes from the 538th 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.

N/A
Comments

From: John Flack
To: Carol Brown
Date: 03/06/2007 10:40:53 AM
Subject: Re: attached are the Minutes of the 538th Meeting

Carol: The following minutes can be released to the general public:

The certified minutes for the 538th ACRS Full Committee Meeting, dated March 2, 2007.

Thanks,

John

>>> Carol Brown 03/06/2007 10:19 AM >>>
Attached are the Minutes from the 538th Meeting of the ACRS for your SUNSI Review. These Minutes can be found in ADAMS as ML070610606.

Carol Anne Brown
Administrative Assistant
US Nuclear Regulatory Commission
Advisory Committee on Reactor Safeguards
Operations Support Branch
415-7998, MS T2-E26

CERTIFIED

Date Issued: 03/02/07
Date Certified: 03/02/07

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C. Future Meeting Agenda

REPORTS:

The following reports to Dale E. Klein, Chairman, NRC, from Graham B. Wallis, Chairman, ACRS:

1. **Draft Final Regulatory Guide DG-1145, Combined License Applications for Nuclear Power Plants (LWR Edition)**, dated December 12, 2006

LETTERS:

The following letters to Luis A. Reyes, Executive Director for Operations, NRC, from Graham B. Wallis, Chairman, ACRS:

1. **Proposed Revision to Standard Review Plan Section 13.3, "Emergency Planning,"** dated December 15, 2006 (ML063520450).

2. **Draft Final Regulatory Guide 1.207 (DG-1144), "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors,"** dated December 18, 2006 (ML063600095).

MEMORANDA:

The following memoranda to Luis A. Reyes, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS:

1. **Proposed Revisions to Standard Review Plan Sections in Support of New Reactor Licensing**, dated December 15, 2006

2. **Anonymous Letter Concerning Changes to 10 CFR Part 52 Rulemaking Package (SECY-06-0220)**, dated December 8, 2006

APPENDICES

I. *Federal Register Notice*
II. Meeting Schedule and Outline
III. Attendees
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MINUTES OF THE 538th MEETING OF THE
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
December 7-9, 2006
ROCKVILLE, MARYLAND

The **538th** meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on **December 7-9, 2006**. Notice of this meeting was published in the *Federal Register* on **November 15, 2006** (71 FR **66561**) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting schedule and outline (Appendix II). The meeting was open to public attendance.

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

ATTENDEES

ACRS Members: Dr. Graham B. Wallis (Chairman), Dr. William J. Shack (Vice Chairman), Mr. John D. Sieber, (Member-at-Large), Dr. Said Abdel-Khalik (via teleconference), Dr. George E. Apostolakis, Dr. J. Sam Armijo, Dr. Sanjoy Banerjee, Dr. Mario V. Bonaca, Dr. Michael Corradini, Dr. Thomas S. Kress, Mr. Otto L. Maynard, and Dr. Dana A. Powers. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

[Note: Dr. John T. Larkins was the Designated Federal Official for this portion of the meeting.]

Dr. Graham B. Wallis, Committee Chairman, convened the meeting at 8:30 A.M. He announced in his opening remarks that the meeting was being conducted in accordance with the provisions of the Federal Advisory Committee Act. In addition, he reviewed the agenda for the meeting and noted that no written comments or requests for time to make oral statements from members of the public had been received. Dr. Wallis also noted that a transcript of the open portions of the meeting was being kept and speakers were requested to identify themselves and speak with clarity and volume. He discussed the items of current interest and administrative details for consideration by the full Committee.

II. Draft Final Regulatory Guide, DG-1145, "Combined License Applications for Nuclear Power Plants" (Open)

[Note: Mr. David C. Fischer was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the U.S. Nuclear Regulatory Commission staff to discuss DG-1145, "Combined License Applications for Nuclear Power Plants."
Mr. Eric Oesterle, Office of Nuclear Reactor Regulation, said that DG-1145 provides a roadmap to help Combined License (COL) applicants identify the appropriate content of a COL

application submitted under 10 CFR Part 52. He said that while industry initially developed COL application guidance for a "base case" scenario (NEI-04-01), the staff recognized the need for more comprehensive guidance for COL applicants. Consequently, DG-1145 is structured to address COL applications that reference an early site permit (ESP), a certified design (CD), neither, or both. DG-1145 is meant to be consistent with proposed final revisions to 10 CFR Part 52 and with the new and revised Regulatory Guides and Standard Review Plan (SRP) Sections being developed in support of new reactor licensing. DG-1145 was developed based on the guidance previously published in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," but was updated based on operating experience (as reflected in NRC Bulletins and Generic Letters), draft industry guidance in NEI-04-01, lessons learned from Design Certification (DC) and ESP reviews, and guidance in advanced reactor SECY Papers and related staff requirements memoranda. There was extensive industry involvement in the development of DG-1145. There were monthly workshops on specific portions of DG-1145 between March and September 2006. Approximately 500 industry comments were received on early drafts of the document. A "work-in-progress" draft of the entire document was made publicly available on the NRC's website by June 30, 2006. DG-1145 was issued for a formal 45-day public comment period on September 7, 2006 (71 FR 52826).

Mr. Oesterle described the format and structure of DG-1145. Section C provides the guidance on the content of COL applications. Part C.I provides guidance for a COL applicant that references neither a CD nor an ESP (consistent with proposed 10 CFR Part 52.79). This Part is further subdivided, by chapters and similar to the way a Final Safety Analysis Report (FSAR) is organized. However, a new introductory subsection and a new subsection on probabilistic risk assessment (PRA) were added. Dr. Apostolakis said that it was not clear from reading DG-1145 when certain information would be available (e.g., certain PRA related information). Part C.II provides additional technical guidance (consistent with proposed 10 CFR Part 52.80). Part CII is further subdivided to address: PRA; Inspection, Tests, Analyses, and Acceptance Criteria (ITAAC); and Environmental Reports. Mr. Oesterle explained that this section will need to be revised to reflect the fact that the latest Part 52 rule no longer requires the submittal of a PRA. Part C.III provides guidance for a COL applicant that references just a CD as well as those that reference both a CD and an ESP. This section also provides guidance related to ITAAC, design acceptance criteria (DAC), and COL Action Items. Part CI.V provides guidance on miscellaneous topics associated with a COL application [e.g., operational programs, limited work authorizations, generic issues, regulatory treatment of non-safety systems (RTNSS)]. Mr. Oesterle said that the information needed to get a COL will generally consist of information provided for the CD, information provided to get an ESP, and remaining information (e.g., plant-specific design information, information on operational programs). Dr. Wallis questioned how much the ACRS would need to be involved with the review of the remaining information. Dr. Powers questioned whether any of the COL applications would be for a "green field" site. Mr. Colaccino, Office of New Reactors (NRO), said that the vast majority of the proposed COL application sites have operating reactors adjoining the sites.

Mr. Oesterle also provided a brief status of the development of DG-1145. He said the public comment period closed on October 23, 2006, and that approximately 700 individual comments were received. The staff is currently working to resolve the public comments. He emphasized that DG-1145 will be revised to comport with the final revision to 10 CFR Part 52 as approved by the Commission. Mr. Oesterle said that there is a process in place to ensure consistency between DG-1145 and the proposed Regulatory Guide and SRP Section updates. The staff plans to publish DG-1145 as Regulatory Guide 1.206 after incorporation of public comments and final issuance of the Part 52 rule. The staff is considering additional public forums to update external stakeholders on Regulatory Guide 1.206 prior to publication. Dr. Wallis asked if there would be substantive changes to DG-1145 based on public comments. Mr. Oesterle said that the number of substantive changes would be minimal. Mr. Maynard expressed concern that DG-1145 referenced some old NRC generic letters which contained guidance that he said should be more directly incorporated into DG-1145. Dr. Apostolakis said that he wanted to see DG-1145 again before it was issued as a final Regulatory Guide.

Mr. Harrison, NRR, described the guidance contained in DG-1145 related to PRA and severe accident evaluations. He said that the proposed 10 CFR 52 rulemaking included a requirement for COL applicants to submit a plant-specific PRA to the NRC for review. After completion of DG-1145, the NRC position changed to accept the industry comment to delete this requirement. Rather, final 10 CFR Part 52 now requires that the PRA be maintained available for staff inspection at the applicant's office. The requirement to submit the PRA was deleted throughout Part 52, including the existing requirement for design certification applications. Mr. Harrison said that DG-1145 will need to be revised to reflect the change in the NRC position. Specifically, he said that the majority of the guidance currently in Section C.II.1 (PRA) will need to be incorporated into C.I.19 (FSAR Chapter 19). Since FSAR Chapter 19 is a qualitative, summary description of the PRA, results, insights, uses, etc., staff audits will be necessary to fully understand, review, and confirm the bases for the PRA results and insights and adequacy for the PRA uses/applications [e.g., RTNSS, Reliability Assurance Program (RAP)]. Mr. Harrison said that the requirement to submit a design-specific or plant-specific PRA with the DC or COL application is separate and distinct from the requirement to submit PRA updates to the NRC.

Mr. Harrison stated that the basis for the PRA guidance in DG-1145 is taken from the following: NRC Policy Statements, SECY Papers and related SRMs; experience with design certification reviews for CE System 80+, ABWR, AP-600, and AP-1000; and 10 CFR 52.79 PRA/Severe Accident requirements. Dr. Abdel-Khalik asked if the staff could issue a COL without doing an audit of the applicant's PRA. Mr. Rubin, NRR, said that the staff could possibly get the required information via requests for additional information (RAIs). However, Mr. Saltos, NRR, added that staff would have likely already done an audit of the PRA for the referenced certified reactor design. Dr. Kress said that he thought the PRA should be part of the COL applicant's licensing basis.

Mr. Harrison said that the staff intends to use the applicant's PRA and severe accident evaluations to conclude that nine objectives (derived from NRC Policy Statements, SECY Papers, and related SRMs) are met. Several of the objectives are used to identify and assess the balance of preventive and mitigative features (including operator actions) such that the plant design reflects a reduction in risk compared to existing plants (contemporary with the Severe Accident Policy Statement of 1985). Several other objectives are in support of specific uses and applications of the PRA results for programs [e.g., RTNSS, ITAAC, COL and interface requirements]. Mr. Harrison outlined the regulatory guidance provided in DG-1145 to assist COL applicants in the development of Chapter 19 of the FSAR. Dr. Apostolakis asked that the briefing focus more on technical issues, such as using large release frequency (LRF) as a metric as opposed to large early release frequency (LERF) when reviewing a COL applicant's PRA. Dr. Apostolakis asked where LRF was defined and where 10^{-6} per year came from. Mr. Rubin said that LRF and 10^{-6} came from Commission guidance from the 1990's during the staff's review of evolutionary and advanced reactor designs. He added that the severe accident decision metrics of 10^{-6} per year for LRF (i.e., baseline PRA and not delta change criteria) and conditional containment failure probability of 0.1 are only applied for new reactor licensing and are not living metrics. Dr. Kress noted that the conditional containment failure probability is weighted by the core damage frequency.

Mr. Sieber noted that DG-1145 basically embraces a lot of existing regulatory guidance, codes, and standards, rules, and other documents, and provides a roadmap for applicants with respect to what has to be in a COL application. He said that from that standpoint, there is nothing new in DG-1145.

Dr. Apostolakis asked how the uncertainties associated with a COL applicant's PRA would be addressed. Mr. Harrison said that applicants for design certifications have done fairly extensive sensitivity studies and uncertainty analyses to get an idea of the magnitude of the uncertainties in the calculations. Mr. Saltos clarified that for the design certification reviews, the staff identified areas of uncertainty and then did sensitivity studies to see how the uncertainties could impact the results. They took these sensitivity study results into account in their decision making (e.g., to identify design changes or operational requirements).

Mr. Oesterle summarized several of the more significant public comments on DG-1145. The first major comment was that some of the information requested in DG-1145 would not be available at the time of COL application or even during the COL application review phase. For example, battery characteristic curves will not be available until batteries have been procured which will be after submittal of the COL application and could likely be after issuance of the license. A second major comment was that some of the information requested in DG-1145 was not applicable to passive plant designs. For example, the guidance in Chapter 8 did not provide any specific requirements for offsite AC power systems for passive plant designs that rely on Class 1E batteries for emergency power and non-safety related diesel generators for battery charging. A third major comment was that Sections C.II and C.III of DG-1145 requested design information from the COL applicants in some areas that have already been certified.
For example, the guidance in Chapter 9 of Section C.III requests information that should already have been addressed in a certified design, such as DG support systems. Another major issue related to information that was either not available at the time the COL application was submitted or that required an update to verify that as-built or as-procured information conformed with the certified design. Several public comments suggested that construction inspections rather than ITAAC are the more appropriate verification mechanism.

Mr. Oesterle said that, based on the public comments, the staff is considering having applicants identify those areas where information will be provided later, or will be updated, and having them to propose methods for so doing. The staff is also considering putting additional guidance in DG-1145 for plants that incorporate passive safety systems.

Mr. Maynard expressed concern over some apparent inconsistencies in the level-of-detail of the guidance provided in various sections of DG-1145. He also questioned the staff's need for certain information in the COL application (e.g., organization charts, resumes). Dr. Banerjee questioned the meaning of the word "limiting" in Chapter 15 of DG-1145. He also said that the guidance in this chapter is unclear, particularly for cases where there is not a lot of experience.

Mr. Oesterle agreed to consider individual member comments (provided to the staff in advance of the November 30, 2006 Future Plant Design Subcommittee meeting) in revising DG-1145.

Committee Action

The Committee issued a report to the Chairman on this matter, dated December 12, 2006, recommending that the final rule, 10 CFR Part 52, retain the requirements that a design-specific PRA be submitted with the design certification application and that a plant-specific PRA be submitted with the COL application. The Committee also recommended that DG-1145 be issued as a final Regulatory Guide after the staff ensures that it is consistent with the final rule 10 CFR Part 52 and with the Regulatory Guides and SRP Sections/Chapters being revised or developed in support of new reactor licensing. The Committee asked that it be informed of any significant changes made to this Guide prior to publishing it in final form.

III. Draft Final Regulatory Guide 1.207 (DG-1144), "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors" (Open)

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

The Committee met with representatives of the NRC staff, the American Society of Mechanical Engineers (ASME), and AREVA to discuss the draft final Regulatory Guide 1.207 (DG-1144), "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors." The staff from the Office of Nuclear Regulatory Research (RES) and their contractor, Argonne National Laboratory (ANL), presented the objective, technical basis, and regulatory positions related to Regulatory Guide 1.207. Representatives of ASME and AREVA provided comments related to the need for the regulatory guide and its potential impact on the industry.

The staff of RES and ANL developed Regulatory Guide 1.207 based on an NRR User Need Request 2005-004 to develop guidance for determining fatigue life in the light water reactor (LWR) environments in supporting reviews of applications that the agency expects to receive for new reactors. The staff stated that this regulatory guide was categorized as high priority and needed to be completed by March 2007.

Mr. Hipolito Gonzalez, RES, and Mr. Omesh Chopra, ANL, described the development and technical basis for Regulatory Guide 1.207. The ASME Boiler and Pressure Vessel Code Section III fatigue design curves were developed in the late 1960s and early 1970s and are based on tests conducted in laboratory air environments at ambient temperatures. However, the Code does not explicitly account for potential degradation in the fatigue properties attributable to exposure to LWR coolant environments. Recent fatigue test data and analyses have demonstrated conclusively that LWR environments have a significant impact on the fatigue life of reactor structural materials. To address this effect, the staff has selected an environmental correction factor, F_{en}, to account for LWR environments. By definition, F_{en} is the ratio of fatigue life of the material in a room temperature air environment to its fatigue life in a LWR coolant environment at operating temperature. To incorporate environmental effects into the fatigue evaluation, the fatigue usage is calculated using ASME Section III Code procedures, and the fatigue usage is multiplied by the correction factor. In license renewal applications, applicants have used this methodology to evaluate the fatigue usage of materials in Class 1 components.

The F_{en} methodology that the staff considers acceptable is described in Regulatory Guide 1.207. NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," provides the technical basis for this methodology. In developing the underlying models, ANL researchers analyzed existing data to predict fatigue life as a function of temperature, strain rate, dissolved oxygen level in water, and sulfur content of the steel. A Second issue addressed by Regulatory Guide 1.207 is the non-conservatism of the current ASME stainless steel air design curve. Recent evaluations of stainless steel and nickel alloy fatigue test data demonstrate that the ASME design curve is non-conservative in the mid-to-high cycle fatigue range. NUREG/CR-6909 provides a new stainless steel air design curve and the technical basis for the new curve. In addition, the staff evaluated the incorporation of the F_{en} approach methodology in fatigue analyses for Ni-Cr-Fe alloys and concluded that the new fatigue design curve proposed for austenitic stainless steels also adequately represented the fatigue behavior of these alloys.

There were several comments on Regulatory Guide 1.207 provided by Mr. Bryan Erler, ASME, and Mr. Robert Gurdal, AREVA. These comments were that the existing ASME design curves and methodology are adequate, that there is no need for a new regulatory guide, that the new guide will result in more detailed and costly analysis in the design of new plants, and that the use of the new guide will also result in the need for an excessive number of snubbers and pipe whip restraints.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter dated December 18, 2006, recommending that Regulatory Guide 1.207 be issued as final. The Committee suggested that the staff interact with ASME in the development of a Code Case related to reactor coolant environmental effects on fatigue.

IV. Proposed Revisions to Standard Review Plan Section 13.3, "Emergency Planning" (Open)

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

The Committee met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss proposed revisions to NUREG-0800, Standard Review Plan, Section 13.3, "Emergency Planning."

The staff developed the proposed revision in cooperation with the Department of Homeland Security (DHS), and the Federal Emergency Management Agency (FEMA) to ensure up-to-date guidance is available to the staff to review new reactor licensing applications. The staff discussed the rationale behind the proposed changes to the SRP, which was issued for public comments.

Mr. Dan Barss, NSIR, began the presentation by describing the process of new reactor licensing embedded in 10 CFR Part 52 that which was the impetus behind a complete rewrite of the SRP Section. The regulatory standards for the Emergency Preparedness (EP) program remained the same as provisions were made to incorporate the 10 CFR Part 52 process. For the staff to arrive at a reasonable assurance finding before a license could be issued, the staff needs to ensure that adequate measures will be in place following the proposed onsite and offsite EP plans such that upon occurrence of an emergency condition at the reactor site there is reasonable assurance that public will be protected. The staff pointed out that dose reduction --and not complete dose avoidance-- is the goal of the EP regulations and the SRP.

The staff described the elements of the regulatory requirements and guidance contained in Regulatory Guide 1.101 which references the jointly developed NUREG-0654 (FEMA-REP-1) and endorses the industry document NEI 99-01. The SRP Section describes the information that needs to be provided and reviewed by the staff at various stages of the licensing process. The staff described the use of emergency planning inspections, tests, analyses and acceptance criteria (EP-ITAAC) which would address the features of a complete and integrated EP plan that cannot be described in the required detail at the time of the application. The criteria in the EP-ITAAC need to be met before initial fuel loading with a hearing opportunity provided to petitioners contesting it.

The Members questioned how the staff would determine the acceptability of local government's participation in the offsite plan in support of an early site permit application. The staff responded that existing standards and guidance are extended to the new reactor licensing process, even if some local authorities may decline to participate.

Regarding the staff's efforts to learn from other Countries' EP program and activities, the Members noted the benefit of learning from the good practices of Countries with major nuclear programs.

Some Committee ACRS members noted the need to develop guidance on planning for severe external events, like a major earthquake, that wipes out the infrastructure including the transportation and communication network. The staff indicated that planning for such events is not yet considered and it is assumed that the local authorities will use the available infrastructure in protecting people. The staff also mentioned their effort in seeking "lessons learned" from recent major public evacuation events.

Mr. Alan Nelson, NEI, discussed the industry comments on SRP Section 13.3. One of their concerns was that lack of detailed guidance regarding FEMA review of the offsite plan could generate many requests for additional information from the NRC reviewers and delay the application approval process. Mr. Nelson then described the NEI task force on EP of advanced light water reactor designs and the current effort in developing emergency action levels for passive reactors.

<u>Committee Action</u>

The Committee issued a letter to the Executive Director for Operations on this matter dated December 15, 2006, recommending that NUREG-0800, SRP Section 13.3, "Emergency Planning," be issued.

V. State-of-the-Art Reactor Consequence Analysis Project (Open)

[Note: Mr. 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 status of the staff's efforts associated with the state-of-the-art reactor consequence analysis (SOARCA) project. The staff briefed the Committee on a number of topics related to this project including plans for MELCOR and MACCS code improvement, plant grouping, and selection of scenarios to use for consequence analysis. The staff also briefed the Committee on its plan for a site-specific simulation of offsite emergency response for this project.

Mr. Robert Prato, Office of Research (RES), started the presentation by describing the status for MACCS2 code improvements. He stated that only 8 of 10 MACCS 2 code improvements are being implemented. The wet deposition model aerosol size dependency and angular resolution are not being implemented as a part of MACCS 2 code improvements. Mr. Prato continued his presentation by discussing how the staff is evaluating scenarios selection using core damage frequency. He stated that the unavailability of full-scope level-2 PRAs for all plants, limits the staff ability to select scenarios based on release frequency.

Mr. Randolph Sullivan briefed the Committee on site-specific simulation of offsite emergency response for SOARCA project.

The Members had many questions regarding the technical details of this study and how uncertainties will be addressed. The Members agreed that the technical details be discussed in a subcommittee as the process and calculations further develops.

Committee Action

This was an information briefing. The Committee plans to continue its review of this project as further progress is made by the staff.

VI. Proposed Revisions to Regulatory Guides and Standard Review Plan Sections in Support of New Reactor Licensing (Open)

[Note: Mr. David C. Fischer was the Designated Federal Official for this portion of the meeting.]

The Committee discussed "high-priority" Regulatory Guides and SRP Sections that which are being revised or developed in support of new reactor licensing. The Committee identified five SRP Sections that it decided not to review (i.e.;, proposed Revision 3 to SRP Section 2.3.3, "Onsite Meteorological Measurements Program"; proposed Revision 2 to SRP Section 3.2.1, "Seismic Classification"; proposed Revision 2 to SRP Section 3.2.2, "System Quality Group Classification"; proposed new SRP Section 3.13, "Threaded Fasteners - ASME Code Class 1, 2, and 3"; and proposed new SRP Section 17.4, "Reliability Assurance Program"). The Committee's decision is documented in a memorandum dated December 15, 2006, from John T. Larkins, ACRS Executive Director to Luis A. Reyes, NRC Executive Director for Operations.

Dr. Corradini recommended that the Committee not review SRP Section 2.3.1, Regional Climatology. However, Dr. Powers expressed concern that looking solely at historical records may not be adequate to predict extremes of weather. Dr. Powers agreed to take a closer look at the proposed revision to SRP Section 2.3.1 to see if it adequately addressed his concern.

The Committee decided to consider whether the ACRS should review several other non-high-priority SRP Sections [e.g.; , SRP Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment; SRP Section 6.1.2, Protective Coating Systems (Paint) - Organic Materials"; SRP Section 6.2.7, "Fracture Prevention of Containment Pressure Boundary"; and SRP Section 6.5.5, "Pressure Suppression Pool as a Fission Product Cleanup System"]. The Committee noted that it had completed its review and/or consideration of all of the high priority SRP Sections provided by the staff.

Committee Action

The Committee plans to conduct an accelerated review of all Regulatory Guides and SRP Sections that which are determined to warrant ACRS review.

VII. Subcommittee Report on Thermal-Hydraulic Phenomena

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

The Chairman of the Thermal-Hydraulic Phenomena Subcommittee provided a report to the Committee summarizing the results of the December 5, 2006 meeting with the NRC staff and its contractors concerning the development of the TRAC/RELAP5 Analytical Computational Engine (TRACE) computer code. Members expressed concern about the state of the code documentation and noted that the staff's progress in establishing the TRACE code as the standard NRC tool for evaluating light water reactor behavior is slow. The staff described its response to an anonymous letter that had been received by the Committee concerning the numerical solution scheme for the code. Members noted that the staff's efforts to address the underlying technical issues raised in the anonymous letter should be improved

Committee Action

The Committee plans to consider a letter to the Executive Director for Operations on this matter during its February 2007 meeting.

VIII. Election of ACRS Officers for CY 2007

[Note: Mr. John T. Larkins was the Designated Federal Official for this portion of the meeting.]

The Committee elected William J. Shack as ACRS Chairman, John D. Sieber as ACRS Vice Chairman, and Mario V. Bonaca as Member-at-Large for the Planning and Procedures Subcommittee for CY 2007.

IX.. Executive Session (Open)

[Note: Dr. John T. Larkins was the Designated Federal Official for this portion of the meeting.]

A. RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

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

The Committee discussed the response from the NRC Executive Director of Operations (EDO) to ACRS comments and recommendations included in recent ACRS reports:

- The Committee considered the EDO's response of November 27, 2006, to comments and recommendations included in the October 25, 2006 ACRS letter on the draft final NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications." The Committee decided that it was satisfied with the EDO's response.

- The Committee considered NRR's response of December 1, 2006, to the November 6, 2006 memorandum from the ACRS Executive Director regarding the Browns Ferry Nuclear Plant, Unit 1 - Extended Power Uprate Application and Supplemental Application. The Committee decided that it was satisfied with the NRR's response.

- The Committee considered the EDO's response of December 6, 2006, to comments and recommendations included in the November 27, 2006 ACRS Report on the Safety Aspects of License Renewal Application for the Palisades Nuclear Plant. The Committee decided that it was satisfied with the EDO's response.

B. Report on the Meeting of the Planning and Procedures Subcommittee (Open)

The Committee heard a report from the ACRS Chairman and the Executive Director, ACRS, regarding the Planning and Procedures Subcommittee meeting held on December 6, 2006.

The following items were discussed:

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

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

Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through March 2007 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

During this session, the Subcommittee also discussed and developed recommendations on items requiring Committee action.

Staff Requirements Memorandum

The Committee discussed Staff Requirements Memorandum (SRM) dated November 8, 2006, resulting from the ACRS meeting with the NRC Commissioners on October 20, 2006. It this SRM the Commission stated the following:

1. As licensing under Part 52 continues, the Committee should advise the Commission on effectiveness and efficiency of staff's implementation of lessons learned in areas it has reviewed, for example, the development of guidance documents for early site permits.

2. The Committee should provide its views to the Commission on staff's efforts related to digital instrumentation and controls. The Committee should consider potential means for providing reasonable backup, if appropriate.

3. The ACRS should provide its views to the Commission with respect to staff's work on technology neutral licensing framework with a focus on ensuring the value of such an approach versus the development of a licensing framework for specific designs, such as a high temperature gas cooled reactor or a liquid metal cooled reactor.

4. The ACRS should provide the Commission with its recommendations and basis for areas in which NRC should perform additional long term research.

5. The Committee should work with the staff and external stakeholders to evaluate the different Human Reliability models in an effort to propose either a single model for the agency to use or guidance on which model(s) should be used in specific circumstances.

Global Nuclear Energy Partnership

On February 6, 2006, the Secretary of Energy announced a $250 million FY 2007 budget request to launch the Global Nuclear Energy Partnership (GNEP). GNEP has four main goals: (1) reduce America's dependence on foreign sources of fossil fuels and encourage economic growth; (2) recycle nuclear fuel using new proliferation-resistant technologies to recover more energy and reduce waste; (3) encourage prosperity, growth and clean development around the world; and (4) utilize the latest technologies to reduce the risk of nuclear proliferation worldwide. As envisioned, GNEP will require NRC involvement in licensing several new facilities including a reprocessing facility, a fast flux liquid metal burner reactor, a fuel fabrication facility, a waste vitrification facility, and interim storage facility.

In SECY-06-0066 dated March 22, 2006, the staff requested that the Commission approve plans to address the regulatory and resource implications associated with GNEP. In an SRM, dated May 16, 2006, the Commission directed the staff to develop a conceptual licensing process for GNEP facilities, including review of the one-step licensing provisions for enrichment facilities and features of nuclear power plant combined licensing under Part 52 (i.e., construction authorization and operating license hearing process, design certification process, and early site permitting process). The Commission also noted in the SRM that the ACRS and ACNW could help in defining the issues most important to licensing, inspecting, and ultimate decommissioning of reprocessing and related fuel-cycle facilities.

The staff has prepared a SECY (currently in inter-Office concurrence) on its conceptual licensing approach for the GNEP facilities. NMSS staff plans to brief the ACNW on the SECY paper during the December 2006 ACNW Full Committee Meeting. Areas of primary interest include:

- Conceptual licensing approach for the Advanced Burner Reactor (ABR). The ABR is expected to be a 1000MWt sodium cooled fast flux reactor designed to burn transuranic waste (TRUs) in order to reduce the amount of radiological waste entering the geological repository. The staff has developed a conceptual approach to licensing the ABR. The approach and associated regulatory infrastructure needed to implement the approach will be of significant interest to the Commission.

- Conceptual licensing approach for the spent nuclear fuel reprocessing facility. Part 50 still remains the current regulatory framework for licensing reprocessing facilities, although it primarily pertain to licensing light water reactors. The NRC has not licensed a reprocessing facility in for over 30 years. A joint letter by ACRS/ACNW, dated January 14, 2002 raised concerns over the use of integrated safety assessment (instead of PRA) for licensing similar facilities under 10 CFR Part 70. Unless the staff moves to PRA to risk-inform the process, the ISA verses PRA issue will also be concern for reprocessing facilities.

FY2006 ACRS Letter Matrix

As required by the Commission, the ACRS/ACNW Office needs to submit a summary matrix of the FY2006 ACRS reports. This will involve summarizing the recommendations included in the ACRS reports and letters. This summary matrix is included as part of the ACRS/ACNW Operating Plan submitted to the Commission annually. In order to avoid violation of the ACRS Bylaws, the Committee should authorize the ACRS Executive Director or his designee to summarize the recommendations in the ACRS reports and letters.

Nuclear Safety Research Forum-2007

As a followup to the recent Quadripartite Meeting, Dr. Wallis received a letter from

Commissioner Soda, NSC, inviting an ACRS member to give a keynote address at the Nuclear Safety Research Forum-2007, scheduled to be held on Friday, March 9, 2007, in Tokyo, Japan. This is a domestic meeting intended for Japanese audience with two keynote speakers, one from ACRS and another from NEA. The focus of this meeting is on research in the field of aging management and material degradation at nuclear power plants.

Dr. J. Sam Armijo is interested in participating in the meeting and would like to expand the trip to include visiting organizations and laboratories in Japan, whose activities are focused on reactor materials degradation research.

Report by Dr. Powers on the ANS Meeting Session on Sump Blockage and GSI-191

Dr. Powers, who attended the 2006 Winter Meeting of the American Nuclear Society (ANS), prepared the attached report on the Session involving the discussion of sump blockage and GSI-191.

List of Research Topics for ACRS Quality Assessment in FY2007

RES has provided a list of eight topics for the ACRS quality assessment in FY2007. These topics are not consistent with the criteria established in 2004. The Committee needs to revisit the process used by RES in identifying topics.

If the Committee is not satisfied with the topics proposed by RES, we can ask RES to provide another list of topics. The Committee normally selects a list of four topics for assessment. However, only two topics were selected for assessment in 2006. In view of the fact that the ACRS will be preparing its biennial report to the Commission on the overall NRC Safety Research Program in 2007, the Committee should consider selecting only two topics for quality assessment in FY2007.

Election of ACRS Officers for CY 2007

The Committee will elect Chairman and Vice-Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee during the December 7-9, 2006, ACRS meeting. During the November meeting, the members were requested to inform the ACRS Executive Director in writing by November 24, 2006, if they do not wish to be considered for any or all of the Offices. So far, two Members have notified the ACRS Executive Director that they do not wish to be considered for all of the Offices.

Subcommittee Report on TRACE Code

The ACRS Subcommittee on Thermal-Hydraulic Phenomena held a meeting on December 5, 2006 to discuss the activities associated with the development of the TRACE computer code. It would be helpful to the Committee if the Subcommittee Chairman provides a brief report to the Committee summarizing issues and concerns of the Subcommittee and future course of action.

Member Issue

Informal ACRS Meetings with the Staff

In an email dated November 30, 2006, Dr. Powers raised some concerns about the informal meetings between the NRC staff and some ACRS members. The Committee discussed this subject.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the **539th** ACRS Meeting, **February 1-3, 2007.**

The **538th** ACRS meeting was adjourned at on **5:15 PM, December 8, 2006.**

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards: Meeting Notice

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on December 7–9, 2006, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Tuesday, November 22, 2005 (70 FR 70638).

Thursday, December 7, 2006, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

8:35 a.m.–10:30 a.m.: Draft Final Regulatory Guide, DG–1145, "Combined License Applications for Nuclear Power Plants" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding Draft Final Regulatory Guide, DG–1145, "Combined License Applications for Nuclear Power Plants," and resolution of significant public comments.

10:45 a.m.–12:15 p.m.: Draft Final Regulatory Guide, DG–1144, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding Draft Final Regulatory Guide DG–1144 and the resolution of public comments.

1:15 p.m.–3:15 p.m.: Proposed Revisions to Standard Review Plan Section 13.3, "Emergency Planning" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding proposed revisions to Standard Review Plan Section 13.3, "Emergency Planning," and related matters.

3:30 p.m.–5:30 p.m.: State-of-the-Art Reactor Consequence Analysis Project (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding status of the staff's efforts associated with the state-of-the-art reactor consequence analysis project.

5:45 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting.

Friday, December 8, 2006, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

8:35 a.m.–9:30 a.m.: Proposed Revisions to Regulatory Guides and Standard Review Plan Sections in Support of New Reactor Licensing (Open)—The Committee will discuss proposed revisions to Regulatory Guides and Standard Review Plan Sections that are being made in support of new reactor licensing.

9:30 a.m.–10:30 a.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

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

11 a.m.–11:30 a.m.: Election of ACRS Officers for CY 2007 (Open)—The Committee will elect Chairman and Vice-Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee.

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

Saturday, December 9, 2006, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

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

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 2, 2006 (71 FR 58015). 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.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, as well as the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Sam Duraiswamy, Cognizant ACRS staff (301–415–7364), between 7:30 a.m. and 4 p.m., (ET).

ACRS meeting agenda, meeting transcripts, and letter reports are available through the NRC Public Document Room at *pdr@nrc.gov*, or by calling the PDR at 1–800–397–4209, or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS) which is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* or *http://www.nrc.gov/reading-rm/doc-collections/* (ACRS & ACNW Mtg schedules/agendas).

Videoteleconferencing 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 videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

The ACRS meeting dates for Calendar Year 2007 are provided below.

ACRS meeting No.	Meeting dates
—	January 2007.
539	February 1–3, 2007.
540	March 8–10, 2007.
541	April 5–7, 2007.
542	May 3–5, 2007.
543	June 6–8, 2007.
544	July 11–13, 2007.
—	August [1]
545	September 6–8, 2007.
546	October 4–6, 2007.
547	November 1–3, 2007.
548	December 6–8, 2007.

[1] No ACRS Meeting.

Dated: November 8, 2006.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E6–19239 Filed 11–14–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards (ACRS)

Subcommittee Meeting on Materials, Metallurgy, and Reactor Fuels; Notice of Meeting

The ACRS Subcommittee on Materials, Metallurgy, and Reactor Fuels will hold a meeting on December 6, 2006, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Wednesday, December 6, 2006—1:30 p.m. until the conclusion of business.

The Subcommittee will review Draft Regulatory Guide DG–1144, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors." The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Charles G. Hammer (telephone 301/415–7363) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 6:45 a.m. and 3:30 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes to the agenda.

Dated: November 8, 2006.

Antonio F. Dias,

Acting Branch Chief, ACRS/ACNW.

[FR Doc. E6–19241 Filed 11–14–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards (ACRS) Subcommittee Meeting on Thermal-Hydraulic Phenomena; Notice of Meeting

The ACRS Subcommittee on Thermal-Hydraulic Phenomena will hold a meeting on December 5, 2006, 11545 Rockville Pike, Rockville, Maryland in Room T–2B3.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows: *Tuesday, December 5, 2006—8:30 a.m. until the conclusion of business.*

The Subcommittee will hear presentations from the NRC staff, their contractors, and other interested persons concerning the progress they have been making in the development of the TRACE T/H system analysis code. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Ralph Caruso (Telephone: 301–415–8065) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 4:15 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes to the agenda.

Dated: November 8, 2006.

Antonio F. Dias,

Acting Branch Chief, ACRS/ACNW.

[FR Doc. E6–19280 Filed 11–14–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Notice of Sunshine Act Meetings

AGENCY: Nuclear Regulatory Commission.

DATE: Weeks of November 13, 20, 27, December 4, 11, 18, 2006.

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

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of November 13, 2006

There are no meetings scheduled for the Week of November 13, 2006.

Week of November 20, 2006—Tentative

There are no meetings scheduled for the Week of November 20, 2006.

Week of November 27, 2006—Tentative

Thursday, November 30

12:55 p.m. Affirmation Session (Public Meeting) (Tentative). a. Hydro Resources, Inc. (Crownpoint, NM) Intervenors' Petition for Review of LBP–06–19 (Final Partial Initial Decision—NEPA Issues) (Tentative).

Week of December 4, 2006—Tentative

Thursday, December 7, 2006

9:30 a.m. Discussion of Security Issues (Closed—Ex. 2 & 3).

Week of December 11, 2006—Tentative

Monday, December 11, 2006

1:30 p.m. Briefing on Status of Decommissioning Activities (Public Meeting) (Contact: Keith McConnell, 301–415–7295).

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

Tuesday, December 12, 2006

9:30 a.m. Briefing on Threat Environment Assessment (Closed—Ex. 1).

1:30 p.m. Discussion of Security Issues (Closed—Ex. 1 & 3).

Wednesday, December 13, 2006

9:30 a.m. Briefing on Status of Equal Employment Opportunity (EEO) Programs (Public Meeting) (Contact: Barbara Williams, 301–415–7388).

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

Thursday, December 14, 2006

9:30 a.m. Meeting with Advisory Committee on Nuclear Waste (ACNW) (Public Meeting) (Contact: John Larkins, 301–415–7360).

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

November 20, 2006 (REVISED)

SCHEDULE AND OUTLINE FOR DISCUSSION
538th ACRS MEETING
DECEMBER 7-9, 2006

THURSDAY, DECEMBER 7, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - ~~10:30~~ A.M. 10:35 Draft Final Regulatory Guide, DG-1145, "Combined License Applications for Nuclear Power Plants" (Open) (TSK/DCF)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff regarding Draft Final Regulatory Guide, DG-1145, "Combined License Applications for Nuclear Power Plants," and resolution of significant public comments.

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

~~10:30 - 10:45~~ A.M. 10:35 - 10:50 *****BREAK*****

3) ~~10:45 - 12:15~~ P.M. 10:50 - 12:30 P.M. Draft Final Regulatory Guide, DG-1144, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors" (Open) (JSA/CGH/CS)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding Draft Final Regulatory Guide, DG-1144 and the resolution of public comments.

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

~~12:15 - 1:15~~ P.M. 12:30 - 1:30 P.M. *****LUNCH*****

4) ~~1:15 - 3:15~~ P.M. 1:30 - 3:36 Proposed Revisions to Standard Review Plan Section 13.3, "Emergency Planning" (Open) (MLC/DAP/MB)
   4.1) Remarks by the Subcommittee Chairman
   4.2) Briefing by and discussions with representatives of the NRC staff regarding proposed revisions to Standard Review Plant Section 13.3, "Emergency Planning," and related matters.

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

~~3:15 - 3:30~~ **P.M.** *****BREAK*****
3:36 - 4:08 P.M.

5) ~~3:30 - 5:30~~ P.M. 4:08 - 6:15 P.M. — State-of-the-Art Reactor Consequence Analysis Project (Open) (WJS/HPN)

5.1) Remarks by the Subcommittee Chairman
5.2) Briefing by and discussions with representatives of the NRC staff regarding status of the staff's efforts associated with the state-of-the-art reactor consequence analysis project.

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

~~5:30 - 5:45 P.M. ***BREAK***~~ No Break

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

Discussion of proposed ACRS reports on:

6.1) Draft Final Regulatory Guide, DG-1145, "Combined License Applications for Nuclear Power Plants" (TSK/DCF)
6.2) Draft Final Regulatory Guide, DG-1144, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors" (JSA/CGH/CS)
6.3) Proposed Revisions to Standard Review Plan Section 13.3,"Emergency Planning" (MLC/DAP/MB)
6.4) State-of-the-Art Reactor Consequence Analysis Project (Tentative) (WJS/HPN)
6.5) Collaborative Research on Human Reliability Analysis Methods (GEA/EAT)

FRIDAY, DECEMBER 8, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

7) 8:30 - 8:35 A.M. — Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)

8) 8:35 - 9:30 A.M. — Proposed Revisions to Regulatory Guides and Standard Review Plan Sections in Support of New Reactor Licensing (Open) (OLM/DCF)

8.1) Remarks by the Subcommittee Chairman
8.2) Discussion of proposed revisions to Regulatory Guides and Standard Review Plan Sections that are being made in support of new reactor licensing.

9) 9:30 - 10:30 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GBW/JTL/SD)

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

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

~~**10:30 - 10:45 A.M. ***BREAK*****~~ No Break

10) 10:45 - ~~11:00~~ 11:15 A.M. Reconciliation of ACRS Comments and Recommendations (Open) (GBW, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

11) ~~11:00~~ 11:15 - 11:30 A.M. Election of ACRS Officers for CY 2007 (Open) (JTL/SD)
Election of Chairman and Vice-Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee.

11:30 - 1:00 P.M. ***LUNCH***

12) 1:00 - ~~7:00~~ 5:15 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

12.1) Draft Final Regulatory Guide, DG-1145, "Combined License Applications for Nuclear Power Plants" (TSK/DCF)

12.2) Draft Final Regulatory Guide, DG-1144, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors" (JSA/CGH/CS)

12.3) Proposed Revisions to Standard Review Plan Section 13.3, "Emergency Planning" (MLC/DAP/MB)

12.4) State-of-the-Art Reactor Consequence Analysis Project (Tentative) (WJS/HPN)

12.5) Collaborative Research on Human Reliability Analysis Methods (GEA/EAT)

~~**SATURDAY, DECEMBER 9, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND**~~

~~13) 8:30 - 12:00 Noon~~	~~Preparation of ACRS Reports (Open)~~
~~**(10:15-10:30 A.M. BREAK)**~~	~~Continue discussion of proposed ACRS reports listed under Item 12~~
~~14) 12:00 - 12:30 P.M.~~	~~Miscellaneous (Open) (GBW/JTL)~~
	~~Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.~~

NOTE:

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

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

ACRS SD/bjw 11/ /06	ACRS/ACNW MA 11/ /06	ACRS/ACNW MRS 11/ /06	Filed: CM-180

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
538TH FULL COMMITTEE MEETING

December 7-9, 2006
Today's Date: December 7, 2006

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
Donnie Harrison	NRR/DRA
Jerry Wilson	NRO/DNRL
Eric Oesterle	NRO/DNRL
David Matthews	NRO/DNRL
Gareth Parry	NRR/DRA
Eileen McKenna	NRR/DLR
B.P. Jain	NRC/RES
Lynn Mrowca	NRR/DRA
Joe Colaccino	NRO/DNRL
Dan Barss	NSIR/DPR
Steve Tingen	NRR/DE/EQV
Mark Rubin	NRR/DRA
Nick Saltos	NRR/DRA
Charles Ader	NRO/DSRA
Doug Weaver	NRO/DNRL
Andrea Valentin	RES/DFERR
John N. Ridgely	RES/DRASP
Michael Mayfield	NRO/DE
Mark Hartzman	NRR/DE
Don Dube	NRR/DRA
Ching Hng	NRR/DE
Matthew Mitchell	NRR/DCI
Jerome Bettle	NRR/DSS
Kamal Manoly	NRR/DE/EEMB
Bill Cullen	RES/DFERR/ME/CMB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
538TH FULL COMMITTEE MEETING

December 7-9, 2006
Today's Date: December 7, 2006

NRC STAFF SIGN IN FOR ACRS MEETING

Hipolito Gonzales	RES/DFERR/ME/CMB
Kathryn Brock	NSIR/DPR/EPD
Edward H. Roach	NSIR/DPR/EPD
Jonathan T. Johnson	NSIR/DPR/EPD
Bruce Musico	NSIR/DPR
Prosanta Chowdhury	NSIR/DPR
Maitri Banerjee	ACRS
R L Sullivan	NSIR
Robert Prato	RES
Yung Hsien J. Chang	RES/DRASP/PRA
John Monninger	NRR/RES/DRASP
Susan B. Cooper	RES/DRASP
Chris Hunter	RES
Jim Yerokun	RES
Jocelyn Mitchell	RES
Doug Coe	OCM/PBZ
Scott Burnell	OPA
Ata Istar	RES/DFERR

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
538TH FULL COMMITTEE MEETING

December 7-9, 2006
Today's Date: 7, 2006

Outside Attendees SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
Syros Trafores	
Patricia Campbell	General Electric
Russ Bell	NEI
Gary Becker	Southern Nuclear
Robert Gurdal	AREVA NP
Kevin Ennis	ASME
Alan Levin	AREVA
Bryan Erler	ASME
Alan Nelson	NBI
Martin Hug	NEI
R.H. Wessman	self
Biff Bradley	NEI
Jeff Gabon	ERIN
Jim Fulford	ISL

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
538th FULL COMMITTEE MEETING

December 7-9, 2006

TODAY'S DATE: December 7, 2006

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	Donnie Harrison	NRR/DRA
2	JERRY WILSON	NRO / DNRL
3	ERIC OESTERLE	NRO/DNRL
4	DAVID MATTHEWS	NRO/DNRL
5	GARETH PARRY	NRR/DRA
6	Eileen McKenna	NRC
7	B P Jain	NRC/RES
	Lynn Mrowca	NRC/NRR/DRA
9	JOSEPH COLACCINO	NRC/NRO/DNRL
10	DAN BARSS	NRC/NSIR/DPR
11	Steve Tingen	NRR/DE/EQV
12	Mark Rubin	NRR/DRA
13	Nick Saltos	NRR/DRA
14	Charles Ader	NRC/NRO/DSRA
15	Doug Weaver	NRR/DNRL
16	ANDREA VALENTIN	~~RES~~/DFERR
17	John N. Ridgely	RES/DRASP
18	Michael Mayfield	NRO/DE
19	MARK HARTZMAN	NRR/DE
20	DON DUBE	NRR/DRA

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
538th FULL COMMITTEE MEETING

December 7-9, 2006

TODAY'S DATE: December 7, 2006

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	Ching H Ng	NRR/DE
2	Matthew A. Mitchell	NRR/DCI
3	Jerome Bettle	NRR/DSS
4	KAMAL MANOLY	NRR/DE/EEMB
5	Bill Cullen	RES/DFERR/ME/CMB
6	Hipolito Gonzalez	RES/DFERR/ME/CMB
7	Kathryn Brock	NSIR/DPR/EPD
	Edward H. Roach	NSIR/DPR/EPD
9	JONATHAN T. JOHNSON	NSIR/DPR/EPD
10	Bruce Musico	NSIR/DPR
11	PROSANTA CHOWDHURY	NSIR/DPR
12	Maitri Banerjee	ACRS staff
13	R L Sullivan	NSIR
14	R Prato	RES.
15	Yung Hsien J. Chang	RES/DRASP/PRA/HFRB.
16	John Monninger	NRC/RES/DRASP
17	Susan E. Cooper	RES/DRASP
18	Chris Hunter	RES
19	Jimi Yerokun	RES
20	Jocelyn Mitchell	RES

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
538th FULL COMMITTEE MEETING

December 7-9, 2006

TODAY'S DATE: December 7, 2006

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	Doug Coe	OCM/PBL
2	Scott Burnell	OPA
3	ISTAR, ATA	RES/DFERR
4		
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
538th FULL COMMITTEE MEETING
December 7-9, 2006

December 7, 2006
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

	NAME	AFFILIATION
1	SPYROS TRAIFOROS	
2	Patricia Campbell	General Electric
3	Russ Bell	NEI
4	Gary Becker	Southern Nuclear
5	Robert GURDAL	AREVA NP
6	Steve Tingen	
7	KEVIN ENNIS	ASME
8	Alan Levin	AREVA
9	Bryan Erler	ASME
10	Alan Nelson	NEI
11	MARTIN HUG	NEI
12	R. H. Wessman	SLI
13	Biff Bradley	NEI
14	JEFF GABOR	ERIN
15	Jim Fulford	ISL
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Future Agenda

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

December 22, 2006

SCHEDULE AND OUTLINE FOR DISCUSSION
539th ACRS MEETING
FEBRUARY 1-3, 2007

<u>THURSDAY, FEBRUARY 1, 2007, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND</u>

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

2) 8:35 - 11:15 A.M. **(10:00-10:15 BREAK)** <u>Final Review of the Power Uprate Application for the Browns Ferry Nuclear Plant, Unit 1</u> (Open/Closed) (MVB/RC)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and Tennessee Valley Authority (TVA) regarding the 5% power uprate application for the Browns Ferry Nuclear Plant, Unit 1 and the associated NRC staff's final Safety Evaluation.

   Members of public may provide their views, as appropriate.

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

11:15 - 12:45 P.M. *****LUNCH*****

3) 12:45 - 3:30 P.M. **(2:00-2:15 BREAK)** <u>Final Review of the License Renewal Application for the Oyster Creek Generating Station</u> (Open) (OLM/MAJ/MB)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff and AmerGen Energy Company, LLC. regarding the license renewal application for the Oyster Creek Generating Station and the associated NRC staff's final Safety Evaluation Report.

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

3:30 - 3:45 P.M. ***BREAK***

4) 3:45 - 5:15 P.M. Development of TRACE Thermal-Hydraulic Code (Open) (SB/GBW/RC)

4.1) Remarks by the Subcommittee Chairman

4.2) Briefing by and discussions with representatives of the NRC staff regarding the progress made by the staff in developing the TRACE thermal-hydraulic system analysis code and related matters.

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

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

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

5.1) Power Uprate Application for the Browns Ferry Nuclear Plant, Unit 1 (MVB/RC)

5.2) License Renewal Application for the Oyster Creek Generating Station (OLM/MAJ/MB)

5.3) Development of the TRACE Thermal-Hydraulic System Analysis Code (SB/GBW/RC)

FRIDAY, FEBRUARY 2, 2007, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

6) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (WJS/FPG/SD)

7) 8:35 - 10:00 A.M. Proposed Revision to 10 CFR 50.46 LOCA Criteria for Fuel Cladding Materials (Open) (DAP/RC/CGH)

7.1) Remarks by the Subcommittee Chairman

7.2) Briefing by and discussions with representatives of the NRC staff regarding proposed revision to 10 CFR 50.46 loss-of-coolant accident (LOCA) criteria for fuel cladding materials.

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 - 11:15 A.M. Draft Final Revision 1 to Regulatory Guide 1.189 (DG-1170), "Fire Protection for Nuclear Power Plants," and SRP Section 9.5.1, "Fire Protection Program" (Open) (JDS/MAJ)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff regarding draft final revision 1 to Regulatory Guide 1.189 (DG-1170) and Standard Review Plan (SRP) Section 9.5.1, as well as the resolution of public comments.

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

9) 11:15 - 11:30 A.M. Subcommittee Report (Open) (GEA/EAT)

Report by and discussions with the Chairman of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment (PRA) regarding the Economic Simplified Boiling Water Reactor (ESBWR) PRA that was discussed during a meeting on December 14, 2006.

11:30 - 1:00 P.M. ***LUNCH***

10) 1:00 - 2:00 P.M. Wolf Creek Pressurizer Weld Flaws (Open) (JSA/CGH)

10.1) Remarks by the Subcommittee Chairman

10.2) Briefing by and discussions with representatives of the NRC staff regarding the Wolf Creek Pressurizer Weld Flaws, including description, current status, and future actions.

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

11) 2:00 - 2:30 P.M. Proposed Revisions to Regulatory Guides and SRP Sections in Support of New Reactor Licensing (Open) (OLM/DCF)

11.1) Remarks by the Subcommittee Chairman

11.2) Discussion of proposed revisions to Regulatory Guides and SRP Sections that are being made in support of new reactor licensing.

2:30 - 2:45 P.M. ***BREAK***

12) 2:45 - 3:30 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (WJS/FPG/SD)

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

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

13) 3:30 - 3:45 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (WJS, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

3:45 - 4:00 P.M. ***BREAK***

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

14.1) Power Uprate Application for the Browns Ferry Nuclear Plant, Unit 1 (MVB/RC)
14.2) License Renewal Application for the Oyster Creek Generating Station (OLM/MAJ/MB)
14.3) Development of the TRACE Thermal-Hydraulic System Analysis code (SB/GBW/RC)
14.4) Proposed Revision to 10 CFR 50.46 LOCA Criteria for Fuel Cladding Materials (DAP/RC/CGH)
14.5) Draft Final Revision 1 to Regulatory Guide 1.189 and SRP Section 9.5.1 (JDS/MAJ)

SATURDAY, FEBRUARY 3, 2007, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

15) 8:30 - 12:30 P.M. **(10:15-10:30 A.M. BREAK)** Preparation of ACRS Reports (Open)
Continue discussion of proposed ACRS reports listed under Item 14.

16) 12:30 - 1:00 P.M. Miscellaneous (Open) (WJS/JTL)
Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

- Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.
- Thirty-Five (35) hard copies and (1) electronic copy of the presentation materials should be provided to the ACRS.

APPENDIX V

LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
538TH ACRS MEETING
December 7-9, 2006

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

MEETING HANDOUTS

AGENDA ITEM #	DOCUMENTS
1.	Opening Remarks by the ACRS Chairman 1. Items of Interest, dated December 7-9, 2006
2.	Draft Final Regulatory Guide, DG-1145, "Combined License Applications for Nuclear Power Plants" 2. Proposed Agenda for DG-1145 Discussion 3. Presentation on DG-1145, "Combined License (COL) Applications for Nuclear Power Plants (LWR Edition)," Slides prepared by NRC staff, Eric Oesterley, NRO/DNRL/NGIF
3.	Draft Final Regulatory Guide, DG-1144, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactos" 4. Proposed Schedule 5. RG 1.207 - Guidelines for Evaluating Fatigue Analysis Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors (Slides presented by Hipolito Gonzalez and Omesh Chopra on behalf of NRC staff).
4.	Proposed Revisions to Standard Review Plan 13.3, "Emergency Planning" 6. Proposed Revision to Section 13.3, "Emergency Planning" (EP) of the Standard Review Plan (SRP) & Combined License Application Guidance (DG-1145) [Slides presented by NRC/NSIR, Daniel M. Barss] 7. NUREG-0800 Section 13.3, "Emergency Planning" [Slides presented by Alan Nelson of the Nuclear Energy Institute]
5.	State-of-the-Art Reactor Consequence Analysis Project 8. State-of-the-Art Reactor Consequence Analyses [Slides presented by Robert J. Prato of NRC/RES] 9. Site-Specific Simulation of Offsite Emergency Response for SOARCA [Slides presented by Randolph L. Sullivan, NRC/NSIR] 10. State of the Art Reactor Consequence Analysis Information Request 11. Plant Containment Matrices 12. Mark I BWRs Internal Events Screening 13. Westinghouse 4-Loop, Large Dry PWRs Internal Events Screening

6. Preparation of ACRS Reports

7. Opening Remarks, Friday, December 8, 2006

8. Proposed Revisions to Regulatory Guides and Standard Review Plan Sections in Support of New Reactor Licensing

9. Future ACRS Activities/Report of the Planning and Procedures Subcommittee
   14. P&P Subcommittee Minutes, December 6, 2006
   15. ACRS Review of High Priority Standard Review Plan Sections (Status Report)

10. Reconciliation of ACRS Comments and Recommendations
   ·16. Handout Containing Prior Letters/Memos

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2. Draft Regulatory Guide DG-1145, "Combined License Applications for Nuclear Power Plants (LWR Edition)"
   17. Table of Contents
   18. Proposed Agenda
   19. Status Report

3. Review of Regulatory Guide 1.207 (DG-1144), "Guidelines for Evaluating Fatigue Analyis Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors"
   20. Proposed Schedule
   21. Status Report

4. Proposed Revisions to Emergency Planning Regulatory Guide and Review Standards
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5. State-of-the-Art Consequences Analysis Project
   25. Table of Contents
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   28. Table of Contents
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ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
538th	9	9.1

Title: PLANNING & PROCEDURES/ FUTURE ACTIVITIES

Authors
John Larkins

Document Attached

P&P Subcommittee Minutes, December 6, 2006

9

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

From Staff Person:
Sam Duraiswamy

CERTIFIED:

SUMMARY/MINUTES OF THE
ACRS PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
December 6, 2006

The ACRS Subcommittee on Planning and Procedures held a meeting on December 6, 2006, in Room T-2B3, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 10:00 am and adjourned at 11:15 am.

ATTENDEES
G. Wallis
W. Shack
J. Sieber

ACRS STAFF
J. T. Larkins
F. Gillespie
S. Duraiswamy
H. Nourbakhsh
R. Caruso
J. Flack
E. Thornsbury
M. Junge
D. Fischer
M. Snodderly
J. Gallo
T. Santos

1) Review of the Member Assignments and Priorities for ACRS Reports and Letters for the December ACRS meeting

Member assignments and priorities for ACRS reports and letters for the December ACRS meeting are attached (pp. 7). Reports and letters that would benefit from additional consideration at a future ACRS meeting were discussed.

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the December ACRS meeting be as shown in the attachment (pp. 7) with the exception of the ACRS report on Human Reliability Models. Dr. Apostolakis should hold a Subcommittee meeting to discuss the Commission request on this matter and submit a revised report for Committee consideration at a future ACRS meeting.

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through March 2007 is attached (pp. 8-9). 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

During this session, the Subcommittee also discussed and developed recommendations on items requiring Committee action (pp. 10-13).

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the anticipated workload. Changes will be made, as appropriate.

3) Staff Requirements Memorandum

In a Staff Requirements Memorandum (SRM) dated November 8, 2006, resulting from the ACRS meeting with the NRC Commissioners on October 20, 2006, the Commission stated the following (pp. 14):

1. As licensing under Part 52 continues, the Committee should advise the Commission on effectiveness and efficiency of staff's implementation of lessons learned in areas it has reviewed, for example, the development of guidance documents for early site permits.

2. The Committee should provide its views to the Commission on staff's efforts related to digital instrumentation and controls. The Committee should consider potential means for providing reasonable backup, if appropriate.

3. The ACRS should provide its views to the Commission with respect to staff's work on technology neutral licensing framework with a focus on ensuring the value of such an approach versus the development of a licensing framework for specific designs, such as a high temperature gas cooled reactor or a liquid metal cooled reactor.

4. The ACRS should provide the Commission with its recommendations and basis for areas in which NRC should perform additional long term research.

5. The Committee should work with the staff and external stakeholders to evaluate the different Human Reliability models in an effort to propose either a single model for the agency to use or guidance on which model(s) should be used in specific circumstances.

RECOMMENDATION

The Subcommittee recommends that Dr. Corradini (Item 1), Dr. Powers (Item 4), Mr. Sieber (Item 2), Dr. Kress (Item 3), and Dr. Apostolakis (Item 5) propose a course of action for responding to the issues raised by the Commission.

4) Global Nuclear Energy Partnership

On February 6, 2006, the Secretary of Energy announced a $250 million FY 2007 budget request to launch the Global Nuclear Energy Partnership (GNEP). GNEP has four main goals: (1) reduce America's dependence on foreign sources of fossil fuels and encourage economic growth; (2) recycle nuclear fuel using new proliferation-resistant technologies to recover more energy and reduce waste; (3) encourage prosperity growth and clean development around the world; and (4) utilize the latest technologies to reduce the risk of nuclear proliferation worldwide. As envisioned, GNEP will require NRC involvement in licensing several new facilities including a reprocessing facility, a fast flux liquid metal burner reactor, a fuel fabrication facility, a waste vitrification facility, and interim storage facility.

In SECY-06-0066 dated March 22, 2006, the staff requested that the Commission approve plans to address the regulatory and resource implications associated with GNEP. In an SRM, dated May 16, 2006, the Commission directed the staff to develop a conceptual licensing process for GNEP facilities, including review of the one-step licensing provisions for enrichment facilities and features of nuclear power plant combined licensing under Part 52 (i.e., construction authorization and operating license hearing process, design certification process, and early site permitting process). The Commission also noted in the SRM that the ACRS and ACNW could help in defining the issues most important to licensing, inspecting, and ultimate decommissioning of reprocessing and related fuel-cycle facilities.

The staff has prepared a SECY (currently in inter-Office concurrence) on its conceptual licensing approach for the GNEP facilities. NMSS staff plans to brief the ACNW on the SECY paper during the December 2006 ACNW Full Committee Meeting. Areas of primary interest include:

- Conceptual licensing approach for the Advanced Burner Reactor (ABR). The ABR is expected to be a 1000MWt sodium cooled fast flux reactor designed to burn transuranic waste (TRUs) in order to reduce the amount of radiological waste entering the geological repository. The staff has developed a conceptual approach to licensing the ABR. The approach and associated regulatory infrastructure needed to implement the approach will be of significant interest to the Commission.

- Conceptual licensing approach for the spent nuclear fuel reprocessing facility. Part 50 still remains the current regulatory framework for licensing reprocessing facilities, although it primarily pertain to licensing light water reactors. The NRC has not licensed a reprocessing facility in for over 30 years. A joint letter by ACRS/ACNW, dated January 14, 2002 raised concerns over the use of integrated safety assessment (instead of PRA) for licensing similar facilities under 10 CFR Part 70. Unless the staff moves to PRA to risk-inform the process, the ISA verses PRA issue will also be concern for reprocessing facilities.

RECOMMENDATION

The Subcommittee recommends that interested members of the ACRS attend the 175th ACNW Full Committee session on GNEP. The ACRS staff should keep the Committee informed as this topic develops further. At the appropriate time the ACRS should discuss with the ACNW how the review of this regulatory activity should be shared between Committees.

5) FY2006 ACRS Letter Matrix

As required by the Commission, the ACRS/ACNW Office needs to submit a summary matrix of the FY2006 ACRS reports. This will involve summarizing the recommendations included in the ACRS reports and letters. This summary matrix is included as part of the ACRS/ACNW Operating Plan submitted to the Commission annually. In order to avoid violation of the ACRS Bylaws, the Committee should authorize the ACRs Executive Director or his designee to summarize the recommendations in the ACRS reports and letters.

RECOMMENDATION

The Subcommittee recommends that the Committee authorize the ACRS Executive Director or his designee to summarize the recommendations in the FY2006 ACRS reports.

6) Nuclear Safety Research Forum-2007

As a followup to the recent Quadripartite Meeting, Dr. Wallis received a letter from Commissioner Soda, NSC, inviting an ACRS member to give a keynote address at the Nuclear Safety Research Forum-2007, scheduled to be held on Friday, March 9, 2007, in Tokyo, Japan (pp. 15-18). This is a domestic meeting intended for Japanese audience with two keynote speakers, one from ACRS and another from NEA. The focus of this meeting is on research in the field of aging management and material degradation at nuclear power plants.

Dr. J. Sam Armijo is interested in participating in the meeting and would like to expand the trip to include visiting organizations and laboratories in Japan, whose activities are focused on reactor materials degradation research.

RECOMMENDATION

The Subcommittee recommends that the Committee send Dr. Armijo to provide a keynote speech at this meeting and visit facilities in Japan which carry out materials research related to nuclear power plants.

7) Report by Dr. Powers on the ANS Meeting Session on Sump Blockage and GSI-191

Dr. Powers, who attended the 2006 Winter Meeting of the American Nuclear Society (ANS), prepared the attached report (pp. 19-21) on the Session involving the discussion of sump blockage and GSI-191.

8) List of Research Topics for ACRS Quality Assessment in FY2007

RES has provided a list of eight topics (pp. 22) for the ACRS quality assessment in FY2007. These topics are not consistent with the criteria established in 2004. The Committee needs to revisit the process used by RES in identifying topics.

If the Committee is not satisfied with the topics proposed by RES, we can ask RES to provide another list of topics. The Committee normally selects a list of four topics for assessment. However, only two topics were selected for assessment in 2006. In view of the fact that the ACRS will be preparing its biennial report to the Commission on the overall NRC Safety Research Program in 2007, the Committee should consider selecting only two topics for quality assessment in FY2007.

RECOMMENDATION

The Subcommittee recommends that Dr. Powers provide his views on the list of topics proposed by RES as well as on the number of topics to be assessed in FY2007. Also, Dr. Powers should propose the topics for quality assessment by the ACRS in FY2007 along with assignments.

9) Election of ACRS Officers for CY 2007

The Committee will elect Chairman and Vice-Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee during the December 7-9, 2006, ACRS meeting. During the November meeting, the members were requested to inform the ACRS Executive Director in writing by November 24, 2006, if they do not wish to be considered for any or all of the Offices. So far, two Members have notified the ACRS Executive Director that they do not wish to be considered for all of the Offices.

10) Holiday Party

The Holiday Party sponsored by the members is scheduled between 11:30 and 1:00 pm on Friday, December 8, 2006. Some Commissioners and the EDO are expected to attend this party.

11) Subcommittee Report on TRACE Code

The ACRS Subcommittee on Thermal-Hydraulic Phenomena held a meeting on December 5, 2006 to discuss the activities associated with the development of the TRACE computer code. It would be helpful to the Committee if the Subcommittee Chairman provides a brief report to the Committee summarizing issues and concerns of the Subcommittee and future course of action.

RECOMMENDATION

The Subcommittee recommends that Dr. Banerjee, the Thermal-Hydraulic Phenomena Subcommittee Chairman, provide a report to the Committee at the December 2006 ACRS meeting.

12) Member Issue

Informal ACRS Meetings with the Staff

In an email dated November 30, 2006, (pp. 23) Dr. Powers raised some concerns about the informal meetings between the NRC staff and some ACRS members.

RECOMMENDATION

The Subcommittee recommends that informal meetings between the staff and some ACRS members be held, as warranted. The Subcommittee's views on informal meetings are as follows:

- The objective of the informal meetings are to obtain information on the status of staff activities associated with some significant regulatory matters for use in scheduling Subcommittee and full Committee meetings, as needed.
- They are information gathering meetings as allowed by FACA and are intended to identify significant issues of concern to the ACRS members but not to discuss those issues in detail.
- Such meetings will help the new ACRS members to obtain background information on certain topics, which in turn will minimize raising fundamental questions during ACRS meetings.
- These meetings will help the staff to prepare focused presentations for ACRS meetings.
- Any issues discussed at these meetings are always scheduled for detailed discussion during a Subcommittee and/or a full Committee meeting, thereby providing a forum to the public to express their views consistent with FACA requirements.

ANTICIPATED WORKLOAD
December 7-9, 2006

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	—	Thornsbury	Collaborative Research on Human Reliability Analysis Methods	—	Committee needs to decide whether to send this report	**Draft**
Armijo	—	Santos/Hammer	Proposed Reg. Guide, DG-1144, "Guidelines for Evaluating Fatigue Analysis Incorporating Life Reduction of Metal Components Due to the Effects of the Light Water Reactor Environment for New Reactors"	A+	To support Agency schedule	—
Corradini	Powers	Banerjee	Proposed Revisions to SRP Section 13.3, "Emergency Planning"	A+	To support Agency schedule	**Draft**
Kress	—	Fischer	Proposed Reg. Guide, DG-1145, Combined License Applications for Nuclear Power Plants	A+	To support Agency schedule	**Draft**
Maynard	—	Fischer	Proposed Revisions to Reg. Guides and SRP Sections in Support of New Reactor Licensing	—	—	—
Shack	—	Nourbakhsh	State-of-the-Art Reactor Consequence Analysis **[STATUS REPORT]**	—	—	—

ANTICIPATED WORKLOAD
FEBRUARY 1-3, 2007

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Bonaca	—	Caruso	5% Power Uprate Application for Browns Ferry Unit 1	A	To support staff schedule	—
Maynard	—	Fischer	Proposed Revisions to Reg. Guides and SRP Sections in support of New Reactor Licensing	—	—	—
		Junge/Banerjee	Final Review of the License Renewal Application and the Final SER for the Oyster Creek Generating Station	A	To support staff schedule	—
Powers	Armijo	Caruso	Revised LOCA Criteria for Fuel Cladding Materials	A	To support staff schedule	—
Shack	Armijo	Banerjee/Hammer	Proposed Revision to 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events"	A	To support staff schedule	—
Sieber	—	Junge	Draft Final Revision 1 to Reg. Guide 1.189 (DG-1170), Fire Protection for Nuclear Power Plants and SRP Section 9.5.1, Fire Protection Program	A+	To support agency schedule	—

ANTICIPATED WORKLOAD
MARCH 8-10, 2007

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	—	Thornsbury	Response to Commission SRM on Human Reliability Models	A	To respond to Commission SRM	—
	Sieber	Junge	Risk Management Technical Specification Initiative 4b. Risk-Informed Completion Times	A	To support staff schedule	—
Bonaca	—	Thornsbury	Research on Mitigating Strategies for New Reactor Designs **[CLOSED]**	A	To support staff schedule	—
Kress	Corradini	Fischer	Response to the Commission SRM regarding the Development of Framework Document for Future Plant Licensing	A	To respond to Commission SRM	—
Maynard	—	Fischer	Proposed Revisions to Reg. Guides and SRP Sections in Support of Future Plant Licensing	—	—	—
Sieber	—	Junge	Response to Commission SRM Related to Digital I&C Matters **[TENTATIVE]**	A	To respond to Commission SRM	—
Shack	—	Nourbakhsh	State-of-the-Art Reactor Consequence Analysis	Report as needed	—	—
Wallis	Banerjee	Caruso	Final Results of Chemical Effects Head Loss Tests in a Simulated PWR Sump Pool Environment	A	To support the staff schedule	—

ACRS Items Requiring Committee Action

1 **SRP 3.12, ASME Code Class 1, 2, and 3 Piping Systems and Associated Supports Design** (Open)

Member: Sam Armijo **Engineer:** Cayetano Santos

Estimated Time:

Purpose: Determine a Course of Action

Priority: High

Requested by: NRR

The NRC staff has identified this standard review plan (SRP) section as needing revision in support of new reactor licensing. The Commission directed the staff to complete the development/revision of "high priority" RGs and SRP sections by March 2007.

The Planning and Procedures Subcommittee agrees with Dr. Armijo's recommendation not to review SRP Section 3.12.

2 **SRP Section 17.4,, Rev. 0, "Reliability Assurance Program"** (Open)

Member: George Apostolakis **Engineer:** Eric Thornsbury

Estimated Time:

Purpose: Determine a Course of Action

Priority: High

Requested by: NRR P. Prescott, -3026

The NRC staff has identified this standard review plan (SRP) section as needing development in support of new reactor licensing. The Commission directed the staff to complete the development/revision of "high priority" RGs and SRP sections by March 2007. While this SRP is not on the "high priority" list, it is a new SRP section relevant to new reactors.

This new SRP Section was forwarded to the ACRS for possible review by memo from P. Hiland, NRR to J. Larkins, ACRS on 10/31/06.

The Planning and Procedures Subcommittee agrees with Dr. Apostolakis' recommendation not to review SRP Section 17.4

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3 **Proposed Revision 2 to Standard Review Plan Section 3.2.1, "Seismic Classification"** (Open)

Member: George Apostolakis **Engineer:** Hossein Nourbakhsh

Estimated Time:

Purpose: Determine a Course of Action

Priority: High

Requested by: NRR

The NRC staff has identified this standard review plan (SRP) section as needing revision in support of new reactor licensing. The Commission directed the staff to complete the development/revision of "high priority" RGs and SRP sections by March 2007.

The Planning and Procedures Subcommittee requests that Dr. Apostolakis be prepared to make a recommendation at the December Full Committee Meeting on whether or not the Committee should review this revision.

4 **Proposed New Standard Review Plan Section 3.13, "Threaded Fasteners - ASME Code Class 1, 2, and 3"** (Open)

Member: William Shack **Engineer:** Maitri Banerjee

Estimated Time:

Purpose: Determine a Course of Action

Priority: High

Requested by: NRR

The NRC staff has identified this standard review plan (SRP) section as needing revision in support of new reactor licensing. The Commission directed the staff to complete the development/revision of "high priority" RGs and SRP sections by March 2007.

The Planning and Procedures Subcommittee agrees with Dr. Shack's recommendation not to review SRP Section 3.13.

5 **Proposed Revision 2 to Standard Review Plan Section 3.2.2, "System Quality Group Classification"** (Open)

Member: Sam Armijo **Engineer:** Maitri Banerjee

Estimated Time:

Purpose: Determine a Course of Action

Priority: High

Requested by: NRR

The NRC staff has identified this standard review plan (SRP) section as needing revision in support of new reactor licensing. The Commission directed the staff to complete the development/revision of "high priority" RGs and SRP sections by March 2007.

The Planning and Procedures Subcommittee agrees with Dr. Armijo's recommendation not to review SRP Section 3.2.2.

6 **SRP 5.4.8, Reactor Water Cleanup System (BWR)** (Open)

Member: John Sieber **Engineer:** Michael Junge

Estimated Time:

Purpose: Determine a Course of Action

Priority: High

Requested by: NRR

The NRC staff has identified this standard review plan (SRP) section as needing revision in support of new reactor licensing. The Commission directed the staff to complete the development/revision of "high priority" RGs and SRP sections by March 2007.

The Planning and Procedures Subcommittee agrees with Mr. Sieber's recommendation that the Committee not review this SRP Section.

7 **GSI-191 PWR Sump Performance - Regulatory Information Conference March 2007**

Member: Graham Wallis **Engineer:** Ralph Caruso

Estimated Time:

Purpose:

Priority:

Requested by: NRR M. Scott

The NRR Staff has invited the Committee to make a presentation on GSI-191 "PWR Sump Performance" at the 2007 Regulatory Information Conference on March 13, 2007. The P&P requests that Dr. Wallis recommend a course of action, regarding whether a Committee member should participate, who that member should be, and what should be presented.

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IN RESPONSE, PLEASE
REFER TO: M061020

November 8, 2006

MEMORANDUM TO: John T. Larkins
Executive Director, ACRS

FROM: Annette L. Vietti-Cook, Secretary /RA/

SUBJECT: STAFF REQUIREMENTS - MEETING WITH ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, 2:30 P.M., FRIDAY, OCTOBER 20, 2006, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission met with the Advisory Committee on Reactor Safeguards (ACRS) to discuss the Committee's activities and current focus.

As licensing under Part 52 continues the Committee should advise the Commission on effectiveness and efficiency of staff's implementation of lessons learned in areas it has reviewed, for example, the development of guidance documents for early site permits.

The Committee should provide its views to the Commission on staff's effort related to digital instrumentation and controls. The Committee should consider potential means for providing reasonable backup, if appropriate.

The ACRS should provide its views to the Commission with respect to staff's work on technology neutral licensing framework with a focus on ensuring the value of such an approach versus the development of a licensing framework for specific designs, such as a high temperature gas cooled reactor or a liquid metal cooled reactor.

The ACRS should provide the Commission with its recommendations and basis for areas in which NRC should perform additional long term research.

The Committee should work with the staff and external stakeholders to evaluate the different Human Reliability models in an effort to propose either a single model for the agency to use or guidance on which model(s) should to be used in specific circumstances.

cc: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
OGC

14

Nuclear Safety Commission
3-1-1 Kasumigaseki.Chiyoda-ku. Tokyo, Japan 100-8970
Phone: +81-3-3581-3470. FAX: +81-3-3581-3475.
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NSC

Dr. G. B. Wallis
Chairman, Advisory Committee on Reactor Safeguards
US Nuclear Regulatory Commission
Washington, DC 20555
U.S.A.

November 17, 2006

Dear Dr. Wallis,

I would like to thank you again for all arrangement made for us at the 2006 Quadripartite meeting in Washington, D.C. The meeting was well organized and successful. It was also very informative and valuable in exchanging information and view on issues of safety significance among the participating advisory groups. I truly appreciate all members of ACRS and their staff on behalf of NSC.

Regarding the summary of the meeting, I made a short comment on the slides you presented at the meeting. It was well prepared and effort of your staff was appreciated. I did mention briefly some comments and corrections to be made in the slide at my summary speech. Since I received a request from Ms. Mugeh for us to send comments to the slides, comments have been prepared as attached and sent to Ms. Mugeh. I appreciate if comments from us are taken into consideration for the summary.

As for the future meeting of the Quadripartite meeting, all participating advisory groups indicated that correspondence and information exchange should be continued and meeting should be held more frequently. Suggested are such as an annual meeting or working group meeting on specific topic selected. I do support this idea and I would like to keep in touch with you to further discuss this matter. NSC is ready to host such meeting in Tokyo if requested.

I would like to inform you that NSC jointly with NISA and MEXT plans to hold "Nuclear Safety Research Forum -2007" with focus on research in the field of ageing management and materials degradation at nuclear power plants. The Forum scheduled on March 9, Friday, 2007, in Tokyo, is a domestic meeting intended for Japanese audience with two key note speakers, one from ACRS and one from NEA. I would like to know if you as the Chairman of ACRS or other member of ACRS are available and can accept our invitation. If acceptable, NSC will formally invite you or other designated member to the meeting. Please note that trip expense is borne by NSC for one person and Mr. Echavarri of NEA has already accepted our invitation.

Sincerely,

Kunihisa Soda

Kunihisa Soda
Commissioner

Attachment: Comments to the Summary, Text and PPT

15

NSC Comments on Summary

Please note that NSC comments are indicated by ***the Italic*** in below.

Slide 3

NOTE: Wording is corrected to take into consideration of the current practice as below:

Safety Trends in Member Countries
•Growing use of PRA/PSA in regulation in all countries
•Use of a risk-informed, rather than a risk-based, approach in all countries
•Developing standards and/or guidance for acceptable PRA/PSA approaches in all countries
•Two countries have quantitative safety goals
•Two countries believe safety goals are unnecessary, but use PRA results to draw insights
•Regulator should not try to regulate safety culture.
•Regulator should be able to perform assessments of safety culture
–Regulator should do audits, interview plant personnel, attend meetings to get a feel for the safety culture
•Top management is responsible for safety management of the plant and fostering safety culture
•The corrective action program and backlog can be examined as indicators of safety culture

Slide 9

NOTE: wording is corrected by taking the current practice as of October 19, 2006, into writing as below:

Seismic Design Guidelines
•Japan has a new examination guideline for seismic safety design which is applied to all new nuclear facilities.
–Licensees have been requested by the regulators to re-evaluate seismic safety of existing nuclear facilities based on the new guideline.
•RSK is developing a new guideline which may be implemented in 2-3 years.
•The US has developed a performance-based seismic hazard analysis methodology that may be used in future early site permits and combined license applications

Slide 14

NOTE: wording is corrected to appropriate terminology as below.

Response to Significant Operating Events
•The number of significant events in the past several years is decreasing

•Human performance, organizational factors, and loss of electrical power identified as contributors to significant events

•Material degradation issues are difficult to predict and can result in serious hazards

–Flow accelerated corrosion/***erosion*** in stainless steel

–Boric acid corrosion in low alloy steel

Slide 15

NOTE: wording is corrected to take into consideration of the existing practice in Japan as below:

Plant Aging, Life Extension, and Periodic Safety Reviews

•In Japan a long term maintenance program ***along with ageing evaluation*** should be executed as part of the periodic safety review prior to 30 years of operation

•US regulations for extending the 40-year license are focused on aging management of long-lived passive components

•GPR reviewed EDF's aging management programs and concluded that it is well adapted to the goal of maintaining safety provisions (defense in depth and confinement barriers)

•RSK considers that systematic aging management at plants is necessary and assumes that utilities will establish and follow effective aging management plans

Safety Trends in Member Countries

- Growing use of PRA/PSA in regulation in all countries
- Use of a risk-informed, rather than a risk-based, approach in all countries
- Developing standards and/or guidance for acceptable PRA/PSA approaches in all countries
- Two countries have quantitative safety goals
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- Regulator should not try to regulate safety culture.
- Regulator should be able to perform assessments of safety culture
  - Regulator should do audits, interview plant personnel, attend meetings to get a feel for the safety culture
- *Top management is responsible for safety management of the plant and fostering safety culture*
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Seismic Design Guidelines

- *Japan has a new examination guideline for seismic safety design which is applied to all new nuclear facilities.*
  - *Licensees have been requested by the regulators to re-evaluate seismic safety of existing nuclear facilities based on the new guideline.*
- RSK is developing a new guideline which may be implemented in 2-3 years.
- The US has developed a performance-based seismic hazard analysis methodology that may be used in future early site permits and combined license applications

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- The number of significant events in the past several years is decreasing
- Human performance, organizational factors, and loss of electrical power identified as contributors to significant events
- Material degradation issues are difficult to predict and can result in serious hazards
  - Flow accelerated corrosion/*erosion* in stainless steel
  - Boric acid corrosion in low alloy steel

Plant Aging, Life Extension, and Periodic Safety Reviews

- In Japan a long term maintenance program *along with ageing evaluation* should be executed as part of the periodic safety review prior to 30 years of operation
- US regulations for extending the 40-year license are focused on aging management of long-lived passive components
- GPR reviewed EDF's aging management programs and concluded that it is well adapted to the goal of maintaining safety provisions (defense in depth and confinement barriers)
- RSK considers that systematic aging management at plants is necessary and assumes that utilities will establish and follow effective aging management plans

8

To: Planning and Procedures Subcommittee

From: Dana A. Powers

Subject: **ANS Meeting Session on Sump Blockage and GSI-191**

At the 2006 Winter Meeting of the American Nuclear Society, the Nuclear Installations Safety Division sponsored a terrific session on GSI-191 and the sump blockage issue. NRC unveiled the substantial body of work they were doing and had completed. Technical reports on this work will soon be on the NRC website.

The session began with a very nice introductory and overview presentation by Mike Scott. The presentation provided the history of the issue which for pressurized water reactors began in the 1970's. Scott indicated that there were 9 people in his organization working full time on the resolution of the issue. He noted the problematic materials: fibrous and particulate insulation, coatings, latent debris including tags, labels and trash, aluminum corrosion products and the buffering agent used to control pH in the sump. He noted the importance of chemical effects and that this issue had been raised by the ACRS. He discussed the issues of downstream effects should debris penetrate strainers and enter the reactor coolant system. Scott noted that several licensees were struggling to meet the December 31, 2007 deadline for providing reasonable assurance of strainer operability. A variety of more or less dramatic measures are being taken by licensees to meet their obligations including materials changeouts. Extensions to March 2008 are being granted with cause. NRC is auditing progress. Three such audits have been done and 9 more are planned or underway.

Industry is arguing that visual inspections of coatings is adequate for assessing their potential contribution to the sump burden. NRC has questioned this and experiments are being planned.

Industry remains concerned that new issues may arise and undo resolutions that previously appeared acceptable.

Rob Tregoning presented 4 papers on experimental studies sponsored by NRC. The first dealt with coating transport in a 30 foot flume at the Naval Surface Warfare Center. They have looked at paint chips 1/64" to 2" in size. Paints examined include:

- single coat alkyd (least dense)
- zinc primer/epoxy topcoat (most dense)
- two coat epoxy
- 6 coat epoxy
- three part epoxy used on concrete

The flume was monitored with a laser system so that they could track individual paint chips. They monitor the water velocities where chips first begin to move and water velocities where 80% of the chips are entrained in the flow. The range of velocities is appropriate for the current strainers (nominally 0.05 to 1.0 ft/s). The velocities may not be appropriate for the much larger

(A)

strainers that licensees are now considering. Incipient velocities are lowest for the least dense paint chips. There is not much of a chip size effect. More important is whether the chip is curled so that it is easier to entrain. All the tests are done in pure water so that any surface effects that augment or inhibit entrainment of paint chips are not addressed. Tregoning thinks, however, that the more important omission in the testing is paint chip disintegration.

Tregoning's second presentation dealt with the chemical effects testing that they have done. A test matrix is shown below:

Temp (°C)	pH control agent	pH	boron (ppm)	debris
60	NaOH	10	2800	100% fiberglass
60	Trisodium phosphate	7	2800	100% fiberglass
60	Trisodium phosphate	7	2800	20% fiberglass 80% Cal-sil
60	NaOH	10	2800	80% Cal-Sil
60	Sodium Tetraborate	8-8.5	2400	100% fiberglass

In some of the tests they have included coupons of corroding metals like aluminum so they also get the corrosion products. They distinctly do get chemical effects. The effects are somewhat complicated and involve occlusion of surfaces that reduce problems with strainers. They, unlike the folks working on license renewal, do get calcium phosphate to precipitate from solution.

Tregonings third presentation dealt with smaller scale chemical effects tests they are doing with fiberglass and Cal-Sil insulations. They find with aluminum present rather small effects (small swings in temperature or pH) can trigger very large changes in head loss through the filter system they use. All is evidence of reaching solubility limits for various things in complex mixtures.

Tregonings final presentation dealt with the use of aqueous chemistry codes to model systems. The codes they use are basically thermodynamic speciation codes. They use corrosion kinetics to get time dependence in the computed results. They have examined several codes but seem to have concentrated on EQ3/6, PHREEQC and OLI Stream Analyzer. Tregonings presentation was focused on results obtained with PHREEQC. The codes do something well. The code predictions are very dependent on the thermodynamic data used in the code. It appears to me that the phosphate data base in PHREEQC may be in error. CODATA has blessed data for most of the simple species of interest for the tests.

Some testing is underway also at the Pacific Northwest National Laboratory. They are working with fiber glass and paint particles (Dimetcote 6 zinc primer, Americoat 90 epoxy topcoat, and Amercoat 5450 alkyd). They have a sophisticated way to monitor bed height. They find that the least pressure drop occurs when particles and fibers are premixed. The worst case is to have particles circulating and then add fibers. They do see some bed restructuring with particles moving down among the fibers. They can form complete beds with fibers alone. They cannot form complete beds with Cal-Sil particles alone. Particle beds always have some holes that allow water passage.

NRC is attempting to model the bed formation using a modified version of the Ergun equation. They find guidance from the LANL research on modeling the bed not very useful. The new model can consider the bed a single volume or multiple volumes. The model does account for flow compression of the debris bed on the strainers. There is a non-recoverable height loss and an elastic height loss term in the model. The single volume model seems to underpredict headloss. The two volume model bounds head loss in all but three of the about 40 experiments that have been analyzed. The NRC does plan further work on the modeling.

Altogether, the session was very informative. The work is being documented and the reports will soon be on NRC's website. People who are doing the work have labeled themselves as "sumpologists". Apparently, Shack and I are considered "sumpologists". I don't know whether Graham has earned his sumpology degree yet or not. I bet he does once we start reviewing this issue again.

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From: Hossein Nourbakhsh
To: DanaPowers@msn.com
Date: 12/04/2006 12:08:33 PM
Subject: Candidate Projects for ACRS Quality Asessment-FY 2007

Dana,

RES has proposed the following projects as candidates for the ACRS Quality Assessment for FY 2007.

1. NUREG-1842, "Evaluation of Human Reliability Analysis Methods Against Good Practices"

2. Development of probabilistic risk assessment (PRA) Quality Standards and Incorporation into Reg. Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities

3. Fire Model verification and validation project, National Institute of Standards and Technology (NIST) contact under JCN 6309, "Program to Evaluate and Improve Computer Fire Models."

4. "Associated Circuits Bin 2 Fire Testing," JCN N6125 at Sandia National Laboratories (SNL), --- also known as Able Response tO Live FIRE, or "CAROLFIRE"

5. Proactive Materials Degradation Assessment

6. Pressurized Thermal Shock Reevaluation Technical Basis

7. Fatigue Crack Flaw Tolerance in Nuclear Power Plant Piping - A Basis for Improvements to ASME Code Section XI Appendix L

8. NUREG/CR - 6895 "Technical Review of On-Line Monitoring Techniques for Performance Assessment: Volume 1: State-of-the-Art," published in January 2006

Many of these projects are the same as those RES proposed last year. As you may recall, the Committee raised some concerns last year about the candiadate projects since many of them were either among projects that the Committee had previously reviewed or the project was more about a process rather than a specific research.

Hossein

CC: Cayetano Santos; Frank Gillespie; John Larkins; Michael Snodderly; Sam Duraiswamy

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#11

From: "Dana Powers" <DanaPowers@msn.com>
To: "sxd1" <sxd1@nrc.gov>
Date: 11/30/2006 9:29:37 PM
Subject: Informal Meetings with the Staff on Various Subjects

To: Planning and Procedures Subcommittee
From: Dana Powers

Subject: Proliferation of "informal" Meetings with Staff

I see a proliferation of "informal" meetings with the NRC staff. It is not apparent to me that this is all consistent with ACRS conducting its business in full view of the public. I also worry that not all members will have access to the information passed on in these meeting scheduled when travel arrangements can not be altered to attend.

ITEMS OF INTEREST

538th ACRS MEETING

DECEMBER 7-9, 2006

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ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
538th MEETING
December 7-19, 2006

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- Article entitled, "Cracking Indications at Wolf Creek Lead to Wider PWR Weld Questions, Volume 28/ Number 24/ November 27, 2006 53-56
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NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

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No. S-06-031

"Achieving Improved Nuclear Plant Safety Through Digital Technologies - The Regulator's Perspective"

Prepared Remarks
by Commissioner Peter B. Lyons

at the

Nuclear Plant Instrumentation and Controls and Human-Machine Interface Technology Embedded Topical Meeting

2006 American Nuclear Society Winter Meeting
Albuquerque, New Mexico

November 13, 2006

Introduction

Good morning. I want to thank the American Nuclear Society for inviting me to speak at this topical meeting, and also to thank my fellow panelists for their participation and the contribution of their perspectives today. I must note, as always, that I am speaking today as only one Commissioner.

As I've visited many nuclear plants in the U.S., I've been struck by the predominance of generally very old analog instrumentation. The age of these analog instruments and their increasing obsolescence over many years has clearly motivated industry's interest in using more modern digital replacements.

During my time as an NRC Commissioner, I have also had the opportunity to begin - and I really mean begin - learning about the potential safety benefits and the unique challenges associated with the application of digital technologies to nuclear power plant instrument and control (I&C) systems and the improvements that these technologies make possible for control room designs and operator interface. I've visited several facilities that incorporate such applications, from the plants at Palo Verde, San Onofre, and Waterford that use relatively simple core protection calculators designed in the 1970s, to the Advanced BWR Kashiwazaki Kariwa Units 6 and 7 in Japan using fully computerized control rooms. I've also seen the advanced control room digital retrofit at Oskarshamn Unit 1 in Sweden, the computerized control room of the Civeaux N4 reactor in France with its impressive human-machine interface (HMI), and the fully modern digital systems of the research

In addition, the NRC continues to support the digital I&C and HMI research that I have already mentioned is being done at the OECD Halden Reactor Project. This work is aimed at addressing challenges that include the impact of rapidly changing technology, increasing complexity, new failure modes, system and human reliability metrics, new concepts of operation, and the need for updating acceptance criteria and review procedures. Halden is helping to provide us with a growing technical basis for more realistic safety decisions related to the software and hardware of digital systems, as well as the humans that operate and maintain them. This work includes developing surveillance and monitoring techniques based on advanced decision algorithms, particularly in the areas of on-line monitoring and diagnostics.

Also, I'm very pleased that Halden is working with the OECD's NEA to develop a new database, named Computer Systems Important to Safety, or COMPSIS, to collect digital system failure information to support improved operation and regulation of digital systems. The NRC encourages this effort and expects that it will improve our understanding of digital system failure modes and frequencies based on a worldwide data gathering effort. Halden also cosponsored a workshop in May with the NEA's Working Group on Human and Organizational Factors on "Future Control Station Designs and Human Performance Issues in Nuclear Power Plants," which will help focus human factors work at Halden and elsewhere.

Our international work is part of an NRC Digital System Research Plan that aims to address many related technical regulatory needs. This publically available Plan organizes our digital system safety research into categories of: system characteristics, software quality assurance, risk assessment, cyber-security, emerging new technologies, and advanced reactor I&C and control room designs. In its recent periodic review of the NRC safety research program, the Advisory Committee on Reactor Safeguards (ACRS) gave this Plan good marks. I was also pleased that they recommended further enhancements involving exploring the acceptability of international standards for meeting regulatory requirements as an element of an MDEP.

The NRC's research in this area has also sought to take advantage of the application of digital technology to safety-critical systems in industries beyond nuclear power. Specifically, we have been seeking insights from industries such as aerospace (including the International Space Station), medical devices, military, and foreign accreditation agencies such as TÜV in Germany. In seeking to utilize these insights, we are careful to ensure we fully understand the differences in their safety functions and the degree to which they are relied upon to control the hazards.

Cyber-Security Issues

Cyber-security is another major consideration for digital systems. Through my own work at our national labs, I am very familiar with the need to provide for cyber-security as part of any digital system. For example, the digital systems that provide highly useful plant parameter and status information to the NRC Incident Response Center and other authorized recipients during exercises and real events, like the soon to be upgraded Emergency Response Data System (ERDS), must be designed to absolutely ensure they do not provide any possible mechanism for an outsider to gain access or interfere with internal plant systems. In addition, viruses, trojan horses, and other malware remain a concern for any software-based system, and software used in safety or security applications must be protected through multiple strategies including effective configuration and access controls. The NRC is actively engaged in these issues, having revised regulatory guidance in 2005, reviewed industry cyber-security program guidelines, and proposed new cyber-security requirements to 10CFR73.55. Through these efforts I believe NRC will be prepared to meet the evolving challenge of cyber-security.

Near-Term Retrofitting and Licensing Challenges

Moving now to the immediate needs and applications, I believe that continued and expanded NRC-industry dialog is imperative to maintain the focus of both NRC and industry efforts on the most important challenges for the retrofitting of existing plants and in potential licensing of new plants. Specific examples of the types of digital I&C system regulatory issues that must be addressed are:

- What is acceptable independence for inter-channel communication, for one-way and two-way communication, and between safety and non-safety channels?
- What are acceptable diversity and defense-in-depth?
- What is acceptable digital system reliability, and can it be estimated with any confidence?
- Will advances in digital technology create new failure modes that affect the reliability and maintainability of safety systems?
- How do we reasonably ensure that emergency preparedness, security, and safety of nuclear power plants are protected from cyber-threats?

For new plant designs, overall safety in a plant's design must be considered at every step, from initial concepts through high-level design certification to the final engineering design details, giving special consideration to the role and failure modes of the digital components. The first design certifications under 10 CFR Part 52, the ABWR, CE System 80+, and the AP600 and AP 1000 all required intensive industry/NRC dialog on the high-level architecture of the I&C systems and control room designs. However, much of the details within this high-level architecture remained purposefully undefined and open to new technological advances. The current design certification, in progress for the ESBWR, and for those that follow will all require this same dialog. As we delve further into actual design details, the level and extent of this dialog will need to expand.

For both retrofit and new licensing, the NRC is working to clarify digital-based safety system regulatory standards and acceptance criteria in updates to regulatory guides and the Standard Review Plan. At its most fundamental level, this dialog must lead to regulatory requirements that address:

- The taxonomy of possible digital system failure modes,
- How each failure mode can be mitigated, and
- For a specific plant design, how overall plant safety will be maintained in the event of a digital system failure.

I personally will continue to value the advice of our Advisory Committee on Reactor Safeguards, which should stay very active in these matters.

A significant challenge moving forward into the future will be to keep regulatory guidance current with the pace of digital technology progress. Rulemaking cannot always keep pace - so we need to rely on guidance documents that can. I see no other answer than for the staff, nuclear research community, and the nuclear industry to maintain a joint and active engagement with the larger multi-industry technical community for this rapidly evolving technology.

Advanced Control Room Designs

Until now I've primarily discussed issues associated with digital system safety. But today, building on a wealth of experience from other industries as well as the nuclear power industry, human operators and their information gathering and cognitive processes are being considered to a greater extent than ever before in the design of NPP information displays and controls, aided by ongoing and extensive research. Once again, and in no way to minimize other research work, the example with which I am most familiar is the work at Halden.

Halden experiments include those related to human error, human performance, teamwork and the effects of computer-driven interfaces on human performance. We in the U.S. don't have a reconfigurable simulator for research use, so access to Halden's facilities is invaluable. The Halden simulator can be driven by either a PWR or BWR model, and offers a prototype reconfigurable advanced control room with an integrated surveillance and control system, data collection facilities, and capabilities in virtual and augmented environments. This is a unique resource operated by a staff of knowledgeable and dedicated I&C and Human Performance researchers.

We have used the results of Halden human factors research as an input to our technical bases for regulatory guidance in areas such as alarm systems, control room design, display navigation, and development of human performance measures. The results have also been used as part of the basis for our Standard Review Plan. These guidance documents are for use in reviewing changes to control stations at current reactors, for licensing reviews of new reactors, for license amendment requests, and for plant inspections.

Halden researchers are also investigating the effects of context, task complexity factors, sustained workload and work practices in computer-based control rooms and team cooperation in new operational settings. Future plans include investigating human system interfaces that

- deliver relevant data and information in comprehensible and understandable formats,
- present the data and information in a manner that does not cause cognitive overload or confusion, and
- will be useful for developing guidance for new advanced control rooms.

In addition, the Halden research in virtual environments is an application of exciting new technologies to support human-factors-design input into control room configurations, into radiation (and possibly fire) visualization methods, and into virtual reality-based team training.

Human Resources and Technical Expertise

One of my continuing areas of concern since becoming a Commissioner has been the overall need for the NRC and industry as a whole to attract new people to reemerging work in nuclear power in order to build and maintain the necessary pool of talent to successfully accomplish growth without compromising the safety performance of existing plants. One of the most significant of these challenges is that we are competing for digital system technical expertise with many other industries in a very competitive job market. At the NRC, I believe the solution will be a balance of attracting and building in-house expertise combined with close links to the expertise at our national laboratories and with programs and facilities that are part of the larger technical infrastructure and communities-of-practice for digital systems across all the industries that use these systems for safety or critical functions. By maintaining our connection with this larger infrastructure and utilizing organizations with broad expertise among many industries, we would expect to efficiently be able to take advantage of the most applicable and relevant national and global work being done on safety-critical digital systems.

Another perspective on this same point is that the move toward state-of-the-art I&C systems and HMI in our power reactors and away from antiquated and obsolete technology will certainly enhance the interest and recruitment of the next generation of students to the nuclear industry. But unfortunately, as I visit research reactors throughout the U.S., I am struck by our national failure to upgrade the instrumentation and controls at our research reactor facilities to state-of-the-art capabilities and the negative impact this must have on our ability to attract new students.

A final perspective on this topic is the need for NRC to stay current in training its own staff on digital system technology and regulatory requirements. We use self-study courses in programmable controllers and early next year will launch a new course for our inspectors and other staff on the fundamentals of digital system design, licensing, and operations as used in the nuclear power industry.

Closing

In closing, I will reemphasize my key point: Digital I&C and safety systems offer the potential for improved HMI and safety performance provided that the failure vulnerabilities are thoroughly identified, understood, and mitigated. Achieving this potential will require industry, the research community, and the NRC to work through new and complex technical issues systematically and thoroughly, with the constant mutual goal of justifying the adequacy of overall plant safety. Further, to accomplish this efficiently, we must all seek to fully leverage the experience of others in the international community who have moved ahead in applying digital systems to nuclear power plants.

Lastly, based on my personal experiences, I have long been concerned that we temper our enthusiasm in creating complex computer models with the recognition that our models must be verified against experimental data wherever possible and practicable. We must then always remember the level of validation when judging the extent to which such models can be relied upon for decisions. An extension of this concern is that, although we have an ever-expanding set of new tools to create digital I&C systems that function in more and more complex ways, like the 'brain and nervous system' of a nuclear plant, I also believe that we must constantly remind ourselves that increasing complexity will exponentially increase the cost of demonstrating and maintaining safety and also the difficulty in detecting and correcting problems.

I am encouraged by the ongoing dialogue between NRC staff and the industry to tackle topics such as inter-channel communication, improving the methods to achieve defense-in-depth and diversity where necessary, cyber-security, and advanced control room design. As we continue this dialogue and move forward, I think it is useful to remind ourselves that the greatest difficulties reside in the multitude of details that must be considered. Therefore success will require a constant discipline to master the complexity to ensure it serves only the cause of safety.

I believe a brief quote ascribed to G.F. McCormick says this best, taken from Dr. Nancy Leveson's book, Safeware - System Safety and Computers, A Guide to Preventing Accidents and Losses Caused by Technology:

> *Software temptations are virtually irresistible. The apparent ease of creating arbitrary behavior makes us arrogant. We become sorcerer's apprentices, foolishly believing that we can control any amount of complexity. Our systems will dance for us in ever more complicated ways. We don't know when to stop. . . . We would be better off if we learned how and when to say no.*

As I noted earlier, I'm an optimist with respect to my confidence that this industry, the research community, and the NRC together will systematically and thoroughly address the safety aspects of applying digital systems to nuclear power plants. This embedded topical meeting is an excellent forum for the necessary information exchanges in support of focused and constructive dialog. I very much look forward to meeting the challenges ahead.

Thank you.

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NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

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Web Site: http://www.nrc.gov

No. S-06-032

REMARKS PREPARED FOR DELIVERY BY
DR. DALE E. KLEIN,
CHAIRMAN, NUCLEAR REGULATORY COMMISSION

BEFORE THE
AMERICAN NUCLEAR SOCIETY WINTER MEETING
ALBUQUERQUE, N.M.

NOVEMBER 13, 2006

Good morning. I'm very happy to be here with you. This is quite an audience, it is nice to see a "nuclear renaissance" at an ANS meeting.

I would like to thank the ANS organizing committee for providing me this opportunity. My fellow Commissioner, Dr. Pete Lyons, is also here today. You will hear from him later.

The theme for this meeting is "Ensuring the Future in Times of Change: Non-Proliferation and Security." What I want to discuss with you today is how the NRC can contribute to this mission.

In 2004, President Bush announced, and the Department of Energy has subsequently begun implementing, several major non-proliferation initiatives. You are going to hear in detail about the Global Nuclear Energy Partnership this afternoon, so I won't go into any details.

I can sum up the regulatory aspects of non-proliferation and security very quickly. Essentially, we are not going to have the worldwide nuclear renaissance without addressing safety and non-proliferation concerns. The question is how to do this.

The Commission believes that a strong and fully independent regulator, who communicates and exchanges best practices with strong and independent regulators from other countries, is the best guarantee of an orderly and safe deployment of nuclear plants to meet the world's growing energy demands. U.S. safety and non-proliferation goals can only be achieved in this context by working closely with NRC's international regulatory partners to create in those countries a strong governance framework that ensures that these goals are achieved. The NRC will be working with our domestic and international partners to create that framework.

On the home front, much of the perspective I have gained in my brief time at NRC has to do with the future of nuclear energy here in the United States. It is not surprising that you – and the Commission – are thinking about the future. We are hearing predictions that the U.S. could build 50 nuclear plants in the next 20 years.

Furthermore, half of the 104 nuclear plants in the U.S. have either had their operating licenses extended for 20 years, or have applied for NRC approval. Most of the rest are expected to apply in the future.

I assumed the NRC chairmanship knowing that I would face a different set of challenges than my recent predecessors. We are talking more today about construction than decommissioning in an era that has been described as a "nuclear renaissance." That said, I don't want to talk to you today solely about new reactors. Instead, I want to focus on the things the NRC and the industry must do to insure the safety and reliability of the current operating reactor fleet, and why we must do them.

In my first months at NRC, I have given a fair amount of thought to my vision for the NRC. My vision is pretty basic. First and foremost, I believe that the NRC must continue to be a strong regulator.

I recently returned from Europe, where I conferred at length with my counterparts from other countries with established commercial nuclear programs. I can tell you from those conversations that the world is watching us. I was gratified by the high regard in which the U.S. regulatory regime is held.

I can assure you that the Commission intends to maintain and enhance that reputation for regulatory credibility.

I have simple criteria for achieving that, and here they are:

1. The NRC will hold our licensees accountable;
2. We will articulate our requirements clearly;
3. We will be demanding; and
4. We will be responsive to legitimate needs and concerns.

The NRC needs to show the industry, the financial community – and above all, the public – regulatory stability.

In turn, the industry needs to show the NRC the attention to detail and the focus on quality necessary to protect the public health and safety.

As you know, the elephant in every room in which nuclear people gather is that an accident anywhere would have a drastic impact on the industry everywhere. The Commission's primary responsibility is to protect the public health and safety. The NRC has many programs in place to ensure that no such accident occurs. I invite you as individual members of the "nuclear family" to support this whenever and wherever you can.

As you may have read in the press, low levels of tritium contamination have been discovered at a number of the nation's nuclear power plants. That, in effect, was my welcome to the NRC. The Commission's most basic regulatory obligation is to determine whether there is adequate protection of public health and safety. Addressing public concerns and perceived risk as a result of unplanned and unmonitored releases has been a big part of that job. The NRC has not found any public health impacts from these tritium leaks.

We have put out an extensive report on tritium, and I am encouraged by the industry's response this year. I hope that you, the technical community, will follow up with a solid, long-term public education program to get ahead of the curve of "perceived risk." I believe that an educated public will be an invaluable ally in the efforts to achieve our safety and non-proliferation goals.

In a recent speech to industry executives, I shared some of what I like to call my most basic insights, "Thoughts While Shaving." I got one the other day. What does a regulator want most?

No surprises. If the NRC identifies a problem, especially if it is a surprise, that means the industry is not doing its job and INPO is not doing its job. There should be no surprises.

I know that the industry's response to significant surprises is far-reaching and effective. But the key word here is "response." Where is the next surprise going to be found? Neither the industry, DOE, nor NRC has in my view put enough money in the last decade into research issues associated with operating power plants.

We need to get ahead of the unknowns and the only way to do that is we, including the NRC, DOE and industry, must bring focus and funding to our research efforts.

Accountability and hard work are what is required to get the nation's nuclear industry from the here and now of possibilities to the future that is envisioned – hard work to maintain and improve safety performance for all operating reactors while at the same time preparing for the construction of new reactors.

The Commission is gearing up to meet these challenges, adding personnel and reorganizing. We will increase our staff by a net of about 200 positions a year through 2008, and the Commission is also battling for a greater share of the finite resources of government to get our expanded staff adequate office space and resources to do their jobs.

While we're talking about staffing, let me share a concern. As the entire industry begins to staff up for this "nuclear renaissance," it needs to evaluate its ranks not only in terms of succession planning, but also the expertise of its personnel. I would venture to say that the nuclear operating organizations today are very much different than they were 10 years ago. This is because the demands of the nuclear industry changed. The demands of the industry are changing again and the industry needs to plan for that change instead of responding to it.

The NRC is preparing for the future, training new executives and making organizational changes. The Commission has created an Office of New Reactors, separate from the Office of Nuclear Reactor Regulation. The new construction office in Atlanta will be headed by a Deputy Regional Administrator for Construction.

We will also look at some possible procedural changes in the review process in the future. I would like to see the review time required for early site permits and combined operating licenses reduced, with no compromise on safety. That is not an unrealistic goal, if the industry does its job on the front end.

We will set out our requirements and let the industry know where it stands at all times. The NRC will not be a bottleneck. The NRC will conduct comprehensive safety, environmental and legal reviews.

If the industry provides the NRC with high quality submittals, the NRC will show the industry timeliness.

The key to success in these endeavors is an open, continuous line of communication. There can be differences of opinion, but there *must* be continuous communication.

Another initiative that will help to achieve an orderly and safe expansion of nuclear energy worldwide is the Multinational Design Evaluation Program (MDEP).

Unlike the previous generation of nuclear power plants, the majority of plants to be built around the world in the next five to 15 years will likely be limited to a small number of relatively standardized designs, purchased from a limited number of multinational corporations. This standardization creates an opportunity to leverage the resources and knowledge of the national regulatory authorities who will review these designs. This international regulatory transparency is fundamental in achieving safety and non-proliferation objectives.

In September 2005, the NRC approved Stage One of the MDEP and some preparatory work for Stage Two. Stage One is under way, and is currently focused on the planned design reviews associated with the AREVA EPR reactor. A reactor of this design is now being built in Finland, has been proposed for construction in France and is undergoing pre-application reviews in the U.S. Several U.S. license applications over the next few years are expected to utilize the design.

Stage Two is intended to be more extensive. Its early activities are beginning and will proceed in parallel with Stage One. The primary objective of Stage Two is convergence of codes, standards and safety goals for designs across international borders. Stage Three of the MDEP will depend to a large extent on the results of the prior stages. The implementation and expansion stage would use the products of the Stage Two effort to review the advanced reactor designs of Generation IV reactors.

I believe that the MDEP will initially encourage development of standardized reactor designs, which will allow for more meaningful exchanges of reactor experience. The MDEP should foster the safety of reactors in those countries with less experienced and extensive regulatory regimes, and enhance the safety of advanced reactor designs by encouraging a comprehensive safety review. And eventually, international regulatory partners will become accustomed to sharing insights on licensing that will improve licensing processes in general around the world.

The premise of your meeting theme is undeniably true: This is a time of change, and it is during unsettled times that we must take particular care to ensure the future. There is a lot of hard work to do.

You can be assured that the NRC will do everything in its power to ensure the future in these changing times.

Thank you, and I look forward to your questions during the Q&A period.

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

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Web Site: http://www.nrc.gov

No. S-06-030

The Meaning and Proper Use of "Risk"

Prepared Remarks by
The Honorable Gregory B. Jaczko
Commissioner

at the Quadripartite Meeting
Washington, DC

October 19, 2006

Good morning. I would like to thank the Advisory Committee on Reactor Safeguards, specifically Graham Wallis and John Larkins, for the opportunity to be here today. This is an opportunity to speak to a diverse and distinguished group of U.S. and international nuclear advisory panels that Commissioners do not often have.

You may be wondering what exactly a Nuclear Regulatory Commission (NRC) Commissioner actually does on a daily basis. Well, we do a lot of things focused on developing the high-level policy of the agency. Sometimes we act as lawyers, and practice law without a license, but what concerns me more is when we practice risk analysis without a license.

An acceptable level of risk to the public is ultimately not a technical determination. Risk is simply a tool we use to make decisions. Too often at the Commission level we are not dealing with tangible risk information, but with the perceptions of risk. This idea lays the framework for what I want to speak about this morning.

The country and the world are at a crossroads on the potential deployment of new electrical power sources. Nuclear power contributes 20 percent of this nation's power supply. We would need scores of new plants by 2030 and 2050 just to maintain that 20 percent figure.

Should the markets and energy policy makers choose to maintain the nuclear option, which looks likely, regulators will have more sophisticated risk tools with which to assess the impact of these facilities on public health and safety. You will most likely be using these tools to advise the regulators in your respective countries on important safety issues.

I would like to take a few minutes to discuss one of those tools. Recently the staff decided to update the 1982 consequence analysis study, also known as NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," which indicated quite severe consequences from low probability events. Many people look to this study to evaluate how accepting they are of nuclear power. It is often quoted in the press and by the public interest groups.

The 1982 study did not do a good job, however, of explaining the concept of risk. The document did convey consequences but did not put those consequences and the low probability of them occurring into the proper context. So, the conclusions of this study are often misunderstood.

Because of the lack of clarity provided by the 1982 study, and because of the advances that we have made in our understanding of the behavior of radioactive materials in the intervening decades, the agency is now undertaking an update. The agency is, however, attempting to move away from addressing higher consequence events by arguing they are of such low probability that they are no longer worthy of consideration. This staff proposal, which the Commission endorsed, involves only analyzing the consequences of events whose large early release frequency is 1x10E-6 or greater.

I argued unsuccessfully that this was not the proper approach to updating the consequence analysis study. All this proposal does is to define a certain narrower range of events and analyze the consequences of that predefined and somewhat arbitrary frequency of occurrence.

I believe that we should analyze the full spectrum of events that is physically reasonable to occur at a nuclear power plant. The only way to comprehensively address the consequences of accidents is to focus on those consequences regardless of the probability that they will occur. If we only focus on what is most likely to occur, we will always have doubts and gaps in our knowledge of events which *could* occur.

I believe that as the agency has learned to work with risk tools and become comfortable with them, we have developed a tendency to overly rely on them. I am concerned that the staff and the Commission have tended not to assess risk, but rather to use probability as a surrogate for risk. As we all know, Risk equals Probability times Consequences, but we seem to want to focus on the probability and not the consequences.

Safety is a policy decision. It involves many variables other than just risk. But we have developed an aversion to true and complete consequence analysis. Because we avoid thorough consideration of those events that have a low probability of occurring, we send the wrong message to the public.

One event that piqued my interest was the idea of a steam explosion or alpha-mode failure. I asked the staff to brief me on this event and the specific question I conveyed to them in preparation for the briefing was the following: Is this an event that is of low probability of occurrence or is it an event that is physically not possible to occur?

Back in the 1990s the research community revisited this event and the response I received from the staff after a very good briefing was that this event is of very low probability and is also not physically reasonable to occur.

After the staff briefed me on the alpha-mode failure concept, they provided me with NUREG-1524, which is a July 1996 manuscript on the reassessment of the alpha-mode failure. At the end of this document was what I would consider a dissenting opinion by Dr. Bal Raj Sehgal of the Royal Institute of Technology, Sweden. His paper's first section was titled, "What is the meaning of all those probability estimates?" He noted that he did not fully understand the 10E-6 and10E-5 values advanced at the meeting. What were they based on? What was the level of confidence in the numbers?

Dr. Sehgal argued that the best numerical estimate of alpha-mode containment failure probability that he could actually calculate with a high degree of confidence was 10E-2. In other words, he felt comfortable stating that he was 99.9% certain that the chance of an in-vessel steam explosion causing a PWR containment to fail given a core melt is less than one in 100. That was the most precise probability he could calculate with confidence given the information at his disposal.

He goes on to argue, however, that even though he cannot calculate this event is of a lower probability, he believes it is physically unrealistic to assume that such a failure will occur.

I believe this explanation does a lot to explain the distinction between our ability to calculate probability and a clear look at whether something is physically reasonable. Here we have a relatively high probability event that is not physically reasonable and therefore does not require much concern on the part of a regulator.

I use this example to caution against ignoring the consequences of the opposite of those types of events, events that have a low probability of occurring and high consequences, and are not physically unreasonable.

Why is this concept so relevant today? It goes to the heart of how we regulate.

We are in a new era. We not only have to assess the risk associated with random accidents, now we have to assess the risks of terrorism.

The Commission has done a great deal of work through the national labs to assess the impacts of large aircraft hitting a nuclear power plant. This research was expensive and supported the conclusion that in the unlikely event of a radiological release due to a terrorist attack, there would be time to implement the required offsite planning strategies already in place to protect public health and safety. That is about as far as I will delve into the matter today.

However, I believe the Commission should require that any new plants are designed to withstand an attack by aircraft. For the current fleet of plants, we have assessed the damage that may occur and required licensees to develop mitigating strategies to protect the core, containment, and spent fuel pool in the event of damage from large fires and explosions. I am comfortable with this approach with the current fleet of reactors.

We should not, however, miss this opportunity to design new facilities in a way that would not require such mitigating strategies. Significantly improved separation and protection of systems necessary to maintain core, containment, and spent fuel pool integrity must be a requirement for the next generation of nuclear power plants.

I will leave you with two thoughts:

1) Probability cannot be a surrogate for risk. We must get the consequences of low probability events out to the public in a properly conveyed context. If we do not, the public will always default to the 1982 study as the real consequences of an accident.

2) Risk is only one input. Safety is a policy judgement.

Thank you again for this opportunity and I look forward to answering any questions you may have.

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Speeches are available through a free list serve subscription at the following Web address: http://www.nrc.gov/public-involve/listserver.html . The NRC homepage at www.nrc.gov also offers a SUBSCRIBE link. E-mail notifications are sent to subscribers when speeches are posted to NRC's Web site.

IN RESPONSE, PLEASE
REFER TO: M061020

November 8, 2006

MEMORANDUM TO: John T. Larkins
Executive Director, ACRS

FROM: Annette L. Vietti-Cook, Secretary ***/RA/***

SUBJECT: STAFF REQUIREMENTS - MEETING WITH ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, 2:30 P.M., FRIDAY, OCTOBER 20, 2006, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission met with the Advisory Committee on Reactor Safeguards (ACRS) to discuss the Committee's activities and current focus.

As licensing under Part 52 continues the Committee should advise the Commission on effectiveness and efficiency of staff's implementation of lessons learned in areas it has reviewed, for example, the development of guidance documents for early site permits.

The Committee should provide its views to the Commission on staff's effort related to digital instrumentation and contro
The Committee should consider potential means for providing reasonable backup, if appropriate.

The ACRS should provide its views to the Commission with respect to staff's work on technology neutral licensing framework with a focus on ensuring the value of such an approach versus the development of a licensing framework for specific designs, such as a high temperature gas cooled reactor or a liquid metal cooled reactor.

The ACRS should provide the Commission with its recommendations and basis for areas in which NRC should perform additional long term research.

The Committee should work with the staff and external stakeholders to evaluate the different Human Reliability models in an effort to propose either a single model for the agency to use or guidance on which model(s) should to be used in specific circumstances.

cc: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
OGC
CFO
OCA
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OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR

IN RESPONSE, PLEASE
REFER TO: M061109B

November 13, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

FROM: Annette Vietti-Cook, Secretary /RA/

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON DRAFT FINAL RULE - PART 52 (EARLY SITE PERMITS/STANDARD DESIGN CERTIFICATIONS/COMBINED LICENSES), 9:30 A.M., THURSDAY, NOVEMBER 09, 2006, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff and industry representatives on the draft final rule to update 10 CFR Part 52 (Early Site Permits/Standard Design Certifications/Combined Licenses).

The Commission supports the staff holding a public meeting as soon as reasonably possible to discuss and possibly resolve comments raised on the draft final Part 52 rule, including section 52.99, inspection during construction. The staff should reach out to interested stakeholders to ensure they are informed of the public meeting. The staff should report the results of the meeting shortly following the meeting, so that the Commission can complete voting on the final rule in an expeditious manner.

The staff should brief the Commission Technical Assistants on the intent of the draft rule language regarding applicant or licensee responsibilities for inclusion of international operating experience insights into license applications and plant design.

The staff should provide regular Commission briefings on preparation for combined license applications.

cc: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
OGC
CFO
OCA
OIG
OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR

November 15, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

FROM: Annette L. Vietti-Cook, Secretary /RA/

SUBJECT: STAFF REQUIREMENTS - COMSECY-06-0052 - STATUS OF BROWNS FERRY UNIT 1 RECOVERY PROJECT

The Commission believes that it is premature to authorize the Region II Administrator to allow restart of the Browns Ferry Nuclear Plant (BFN) Unit 1 and, therefore, chooses not to approve the staff request at this time. The Commission is willing to reconsider this decision after the Commission briefing scheduled in January 2007. The Commission will provide further direction, as appropriate, in a Staff Requirements Memorandum (SRM) following the January meeting.

cc: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
OGC
CFO
OCA
OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR

IN RESPONSE, PLEASE
REFER TO: M061025C

November 16, 2006

MEMORANDUM FOR: Luis A. Reyes
Executive Director for Operations

FROM: Andrew L. Bates, Acting Secretary **/RA/**

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON RESOLUTION OF GSI-191, ASSESSMENT OF DEBRIS ACCUMULATION ON PWR SUMP PERFORMANCE, 1:30 P.M., 2006, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by representatives of industry and the staff on the progress on resolution of Generic Safety Issue (GSI) - 191, Assessment of Debris Accumulation on PWR Sump Performance.

The staff should work with industry to develop a matrix for all plants, listing their schedule for installing larger strainers and other major plant modifications that are part of their plan to resolve the sump issue and provide it to the Commission. This matrix should include plants that have requested and been granted extensions and the reasons for the extensions.

The staff should work with industry to develop a systematic approach to buffer evaluation and encourage licensees to replace buffers, when indicated, during scheduled outages.

The staff should inform the Commission through a Commission TA briefing on results of the zone-of-influence (ZOI) containment coatings testing done by industry, including the ongoing industry-sponsored visual coating condition assessment demonstration projects, and the NRC sponsored coating transport tests and whether or not a joint coating condition assessment program will be developed with EPRI.

The staff and industry should make a concerted effort to look at resolution of this issue holistically. Such an approach should include understanding the interdependence of changes in water chemistry on debris accumulation and sump performance.

cc: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
ASLBP
OGC
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Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR

November 16, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

Karen D. Cyr
General Counsel

Jesse L. Funches
Chief Financial Officer

FROM: Andrew L. Bates, Acting Secretary /RA/

SUBJECT: STAFF REQUIREMENTS - SECY-06-0187 - SEMIANNUAL UPDATE OF THE STATUS OF NEW REACTOR LICENSING ACTIVITIES AND FUTURE PLANNING FOR NEW REACTORS

The Commission supports the staff's design-centered review approach (DCRA) described in Regulatory Issue Summary 2006-06 (and any subsequent related guidance documents) for reviewing Combined License (COL) and Design Certification (DC) Applications.

The staff should consider the following set of factors when making resource allocations and schedule decisions if and when actual licensing work exceeds the new reactor budget. These factors apply when allocating resources during budget execution only and should not be applied in preparing budget requests. The staff should continue to plan and budget for all low and medium uncertainty new plant licensing applications.

For COLs:

- for any one of multiple COL applications referencing the same design certification, the extent of the applicant's commitment to the design-centered review approach described in Regulatory Issue Summary 2006-06 and any subsequent related guidance documents (this factor should not, however, disadvantage a COL applicant referencing a design that is not referenced in other COL applications)

- the extent to which an application references a completed early site permit (ESP) and a certified design;

- for applications referencing designs not yet certified or for which significant changes in the current Certificate are being sought by the vendor, the degree to which the staff's design review is in advanced stages and the vendor is providing the necessary support for timely completion;

- the quality and the completeness of the application itself;

- the extent to which an application references an ESP application submitted well in advance of the COL and which demonstrates the likelihood that environmental and emergency planning issues will be resolved prior to the COL hearing;

- the extent to which an applicant has coordinated with applicable state permitting authorities;

- the extent to which an applicant has coordinated toward meeting other applicable

- the schedule of the Department of Homeland Security (DHS) review of an applicant's EP plan, and the schedule for the DHS security consultation consistent with Section 657 of the Energy Policy Act of 2005;

- evidence of the applicant's financial commitment to build a reactor in the near term, such as the extent of procurement and orders for long lead time reactor components that can facilitate the NRC scheduling of vendor and construction inspections and other related financial information;

- the degree of an applicant's adherence to schedules and meeting of milestones that could impact the staff's review;

- the extent to which prioritization of the application could enhance efficiencies in the conduct of the adjudicatory process; and

For ESPs:

- the quality and the completeness of the application itself;

- the extent to which an application is likely to be followed up in the near term by a COL at the designated site; and

- the degree of an applicant's adherence to schedules and meeting of milestones that could impact the staff's review;

For Design Certifications:

- the quality and the completeness of the application itself;

- the extent to which a certification is likely to be followed up in the near term by a COL application that would reference the designated design; and

The staff should continue to keep the Commission fully and currently informed regarding new reactor activities.

The OGC and CFO staff should develop a paper on the feasability and appropriateness under existing law of charging prospective applicants a fee (a payment upfront to be submitted with the applicant's letter of intent), including various options for putting such a fee in place and the pros and cons associated with each option, within 90 days of the date of the SRM on this paper.

cc: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
OCA
OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR

IN RESPONSE, PLEASE
REFER TO: M061025A

November 21, 2006

MEMORANDUM FOR: Luis A. Reyes
Executive Director for Operations

Jesse L. Funches
Chief Financial Officer

FROM: Annette L. Vietti-Cook, Secretary **/RA/**

SUBJECT: SUBJECT: STAFF REQUIREMENTS - BRIEFING ON INSTITUTIONALIZATION AND INTEGRATION OF AGENCY LESSONS LEARNED, 9:30 A.M., 2006, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff on accomplishments and challenges concerning implementation of the Agency's Lessons Learned Program (LLP) and the role of the Lessons Learned Oversight Board.

The Commission emphasized making sure lessons are implemented so that they do not recur, e.g., loss of communicati during an event. The staff should consider whether there are additional legacy effectiveness reviews that would be worthwhile to examine such as construction problems at South Texas, and issues that arose that warranted an Incident Investigation Team. The staff should consider the performance measures contained in Exhibit 3, "Examples of Performance Measures," of Management Directive 6.8, "Lessons-Learned Program," when conducting the annual review of the LLP to evaluate the effectiveness of the program, disposition of recommendations for improving the program, and recommend process changes. The staff also should consider providing a status of the LLP and discussing lessons-learned items as well as the corrective action plans (CAPs) at each Agency's Senior Leadership Meeting.

The staff should continue to develop effective means to communicate the results of the LLP to the NRC staff, the public, and Congressional Oversight Committees. The staff should request stakeholder comments and compare its LLP to those in other federal agencies and industry.

In the interests of continuing reinforcement to the staff of the importance of this program, the staff should either display the actual corrosion evidence of the Davis-Besse event or procure a model that shows the magnitude of this problem. The staff should inform the Commission of the cost to accomplish this task prior to expending funds.

The staff should provide the Commission options for funding continuous review of the effectiveness of the LLP and individual CAPs and for funding the information technology that tracks and supports the program and dissemination of the results.

cc: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
ASLBP
OGC
OCA
OIG

IN RESPONSE, PLEASE
REFER TO: M061016AB

November 21, 2006

MEMORANDUM FOR: Luis A. Reyes
Executive Director for Operations

Jesse L. Funches
Chief Financial Officer

FROM: Annette L. Vietti-Cook, Secretary **/RA/**

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON STATUS OF NEW REACTOR ISSUES–COLS, 9:30 A.M. and 1:30 P.M., MONDAY, OCTOBER 16, 2006, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by representatives of industry, the States of North Carolina and Florida, the Union of Concerned Scientists, the NRC staff, and the Atomic Safety Licensing Board Panel on the status of current activities related to new reactor design certification, Early Site Permits (ESPs) and anticipated Combined License (COL) Applications. The Commission emphasized standardization, high quality applications, and timely, high quality responses to requests for additional information as the way for industry to receive timely decisions from the NRC.

The staff's next update to the Commission on the progress of developing an inspection and oversight program for new reactor construction should discuss how the various components of construction oversight (e.g. vendor inspection, construction site inspection, inspection resource targeting, assessment of inspection finding significance, oversight program responses to findings, enforcement policy aspects, public accessibility and timeliness of inspection and assessment results) all work together in an integrated and coherent manner.

The Commission requests that the CFO and EDO prepare an impact analysis identifying what would be deferred if the NRC is under a continuing resolution at the FY 06 level beyond December 2006.

With regard to the performance-based rule, 10 CFR 20.1406, the staff should appropriately engage stakeholders to consider various improvement approaches such as rulemaking, issuing or modifying regulatory guidance, or endorsing stakeholder developed guidance.

The Commission provided further guidance to the staff on new reactor issues in its Staff Requirements Memorandum on SECY-06-0187, dated November 16, 2006.

cc: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
ASLBP
OGC
OCA
OIG
OPA

November 21, 2006

The Honorable Bart Gordon
Ranking Member, Committee on Science
United States House of Representatives
Washington, D.C. 20515

Dear Congressman Gordon:

I am writing in response to your letter dated October 27, 2006, wherein you requested information on the U.S. Nuclear Regulatory Commission's (NRC's) handling of sensitive unclassified information following the events of September 11, 2001. Specifically, you inquired about the availability of sensitive unclassified information in the Local Public Document Rooms at public libraries near the Nation's commercial nuclear power reactors. In response to your letter, I directed the Executive Director for Operations to review your concerns and respond to me. The enclosed memorandum contains the results of that review. Responses to your questions are also enclosed.

Please contact me should you have any further questions.

Sincerely,

/RA/

Dale E. Klein

Enclosure:
As stated

November 21, 2006

MEMORANDUM TO: Dale E. Klein
Chairman

FROM: Luis A. Reyes ***/RA William F. Kane Acting For/***
Executive Director for Operations

SUBJECT: REVIEW OF NRC'S HANDLING OF SENSITIVE UNCLASSIFIED INFORMATION FOLLOWING SEPTEMBER 11, 2001

In accordance with your direction, the NRC staff has reviewed the agency's handling of sensitive unclassified information following September 11, 2001. Specifically, the staff focused its review on the concerns identified in Congressman Bart Gordon's October 27, 2006, letter to you. The following is a summary of the NRC's review and actions taken on this matter.

The NRC has been aware since shortly after September 11 that a limited amount of sensitive information regarding commercial nuclear power plants exists in a variety of public and private collections. The information that remains publicly available was considered "nonsensitive" prior to September 11 and, in accordance with our strategic goal of openness, was released to the public. Today, in light of the need for increased vigilance, the NRC designates some of this information as "sensitive unclassified non-safeguards information" (SUNSI) and, therefore, withholds it from the public. It should be noted that information directly related to the security programs and protection for nuclear power plants is designated as Safeguards Information, is controlled similar to Classified Information, and is not among the records at public libraries or elsewhere in the public arena.

The NRC acknowledges that a limited quantity of documents currently within the former[1] Local Public Document Room (LPDR) collections meets the revised withholding criteria for SUNSI information. However, the NRC believes that the amount of such information is small and that its utility is minimal given the fact that the level of sensitivity is below that of classified or safeguards information and because of its age and post-September 11 security enhancements

[1]Prior to the development and implementation of the Agencywide Documents Access and Management System (ADAMS), the NRC maintained (funded and provided documents) licensing and regulatory document collections in more than 80 "Local Public Document Rooms" (LPDRs) in local libraries (who volunteered, and were paid, to house and maintain the document collections) in the vicinity of power reactors and some materials licensees. When the NRC implemented ADAMS in 1999, the Commission decided to discontinue funding the LPDR program beyond FY 1999. See 64 Fed. Reg. 48942 (September 9, 1999). In ending the LPDR program, the NRC offered each of the LPDR libraries the opportunity to keep their LPDR document collections. Most of the libraries accepted the NRC's offer to transfer ownership of the collections and those libraries now own and control the collections of pre-ADAMS documents. 64 Fed. Reg. 48942-44.

and physical modifications to NRC-regulated facilities. Therefore, the NRC decided not to attempt to retrieve or restrict access to the previously released information and instead focused our efforts on more recent and relevant public information available in our electronic record-keeping systems. In the past, the NRC declined to accept the collections from former LPDRs that wished to return them. However, the NRC has changed its position on this matter and in a July 12, 2006 letter to the former LPDRs, we indicated that should a former LPDR, that is not part of the Federal Depository Library Program, request to return its collection to the NRC, we will accept the collection. On the same day, the NRC sent a similar letter to former LPDRs that are part of the Federal Depository Library Program instructing them to follow U.S. Government Printing Office policies if they desired to dispose of their collections.

Following the September 11, 2001, terrorist attacks, the NRC took prompt action to enhance the control of information that potentially could be used by an adversary. The NRC immediately advised nuclear facilities to review their information collections (e.g., web sites) to decide if information determined to be security-related in the wake of September 11, 2001, not previously considered sensitive, was publicly available. The NRC conducted a similar review of our web site and public record-keeping systems. This resulted in the NRC and our licensees removing some information previously publicly available. Subsequently, the NRC issued guidance to our staff and licensees on how to recognize sensitive information as well as methods to protect such information from being used by an adversary. The NRC continues to review documents to ensure that information which could be of interest to terrorists is not contained in the documents we place on our web site or in our publicly accessible record-keeping systems, while striving to provide the public with appropriate material on our regulatory activities and policies.

The staff is aware that the Office of the Inspector General (OIG) has been reviewing the NRC's handling of SUNSI information following September 11, 2001. However, it is our understanding that the review is not complete at this time. Upon receipt of OIG's report on this matter, the NRC will review any recommendations and take appropriate actions.

Responses to the specific questions raised in the Congressman's letter are provided as an enclosure to this memorandum.

Enclosure:
As stated

cc: Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
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Responses to the specific questions raised in the Congressman's letter are provided as an enclosure to this memorandum .

Enclosure:
As stated

cc: Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
SECY
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Distribution:
EDO R/F
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Non-Publicly Available, Non-Sensitive

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RTaylor:paa		JSilber	LReyes
11/06/2006	11/07/2006	11/07/2006	11/21/2006

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Question 1: Was there a decision made by the [U.S. Nuclear Regulatory Commission] NRC not to remove information from the local public document rooms, and, if so, who made that decision and why?

Answer:

Following the September 2001 terrorist attacks, the NRC took prompt action to enhance the control of information that could potentially be used by an adversary. The information that remains publicly available in Local Public Document Rooms (LPDRs) was considered "nonsensitive" by the NRC prior to September 11 and was released to the public. In light of the need for increased vigilance, the NRC would now designate some of this information as "sensitive unclassified nonsafeguards information (SUNSI)."

The NRC understood that, upon establishing our criteria for designating information as SUNSI, limited quantities of information now considered sensitive would remain in the public realm. On April 4, 2002, the NRC staff informed the Commission, in COMSECY-02-0015 (at p.2), that "because NRC does not control archival collections external to the agency, documents may continue to be made publically available through other sources." (See attached copy of COMSECY-02-0015 dated April 4, 2002, and associated SRM dated May 28, 2002.) The NRC determined that the usefulness of the information that remained publicly available was minimal given its age and subsequent improvements in security programs and measures. In addition, the anticipated cost and effectiveness of efforts to retrieve this small amount of information did

Enclosure

not support an NRC decision to pursue that course of action.

Question 2: What is the current NRC policy regarding the removal or control of access to sensitive documents from the NRC's local public document rooms?

Answer:

Currently, the NRC's policy is not to remove or restrict access to potentially sensitive documents in the former LPDRs. Since September 11, the NRC has required, and licensees have implemented, substantial security enhancements, including physical modifications to commercial nuclear power plants. Information directly related to these security programs and the protection for nuclear power plants is designated as Safeguards Information, is controlled similar to Classified Information, and is not among the records at public libraries or elsewhere in the public arena. The NRC has determined that the usefulness of the limited quantities of sensitive information available in the LPDRs is minimal given the fact that the level of sensitivity is below that of Classified or Safeguards Information and because of its age, and subsequent improvements in security programs and measures. We continue to work with licensees to ensure that the most recent and relevant information related to the security of nuclear power plants is protected.

Question 3: Has the NRC ever removed documents from its local public document rooms due to security concerns since September 11, 2001? Please provide specific details of any instances of removal that may have occurred and why the NRC believed this was necessary.

Answer:

Other than one isolated incident detailed below, the NRC has not removed, and has no plans to remove on our own initiative, the collections maintained at any of the former LPDRs. The LPDR program was discontinued in September 1999 and ownership of the document collections transferred to the individual libraries. Following the transfer of the collection of NRC documents maintained at the Greenfield Community College library in Greenfield, Massachusetts, which were maintained for the decommissioned Yankee Rowe Nuclear Plant, the NRC regional offices performed a survey of the remaining LPDRs throughout the nation to ascertain the status of their collections. During that survey, a regional staff member removed the collection of documents maintained at the Pottstown Public Library near the Limerick Generating Station in Pennsylvania. The NRC promptly returned the collection to the library because its removal was not in accordance with NRC policy and would restrict public access to legitimate nonsensitive information.

Question 4: Does the NRC have any plans in place to remove sensitive documents removed from ADAMS from the local public document rooms? Please explain how the NRC intends to accomplish this and the scope of documents the NRC believes should be removed – if any.

Answer:

Currently, the NRC has no plans to remove any documents from the former LPDRs on our own initiative. However, on July 12, 2006, the NRC's Deputy Chief Information Officer sent letters to the former LPDRs explaining that if any former LPDR no longer wished to maintain its collection, the NRC would accept an offer to return the collection, provided the former LPDR is not part of the Federal Depository Library Program. Should a former LPDR choose to return its collection, the NRC will assist in making arrangements to properly dispose of the collection.

For former LPDRs that are part of the Federal Depository Library Program, the NRC recognizes that the disposal of documents at these libraries must be in accordance with the U.S. Government Printing Office (GPO) Information Dissemination Policy Statement 72, "Withdrawal of Federal Information Products from Information Dissemination Collection and Distribution Programs." Therefore, if a Federal Depository Library no longer wishes to maintain its collection, the library would need to dispose of the materials following GPO procedures for withdrawing material from the depository collection, as prescribed in the Instructions to Depository Libraries.

Question 5: If the NRC does not plan to remove sensitive documents currently available in local public document rooms, your evaluation of their sensitivity must have shifted since the time when they were removed from ADAMS. Please explain how that reevaluation occurred and when. Provide any documentation necessary to understand this shift in views. Please explain why the materials have not been returned to ADAMS if this has occurred.

Answer:

Since September 11, the NRC screens its documents prior to making them publicly available to ensure that sensitive information that could potentially aid terrorists or adversaries of the United States is appropriately withheld. The NRC continues to work diligently to balance its commitment of openness with the public with the need to prevent releases of sensitive information.

After September 11, the NRC revised its criteria for balancing its goal of releasing as much information as possible with the need to withhold information that might be useful to terrorists. The NRC developed criteria that resulted in a relatively small amount of information being withheld that was previously released to the public. The NRC recognized that there would be limitations on its ability to remove some information deemed sensitive, using the revised criteria, from the public realm after the information had been in the public domain for decades. The NRC decided to implement the policy change and focus its efforts and resources on keeping out of the public domain recent, relevant and easily accessible information and information available in its electronic record-keeping systems. In determining this policy change, the NRC

Answer:

weighed the benefit of withholding information from public access versus its ability to remove certain documents that had already been in the public domain for decades which were, for all practical purposes, out of NRC's control. As stated previously, the NRC believes that the amount of such information is small and that its utility is limited by its age and post-September 11 security enhancements and physical modifications to NRC-regulated facilities.

November 27, 2006

The Honorable Tom Davis
Chairman, Committee on Government Reform
United States House of Representatives
Washington, D.C. 20515

Dear Mr. Chairman:

On behalf of the U.S. Nuclear Regulatory Commission (NRC) and in accordance with 31 U.S.C. 720, enclosed is the NRC's response to the recommendations made by the U.S. Government Accountability Office (GAO) in its report entitled "Nuclear Regulatory Commission: Oversight of Nuclear Power Plant Safety Has Improved, but Refinements Are Needed" (GAO-06-1029). If you have any questions or comments, please contact me.

Sincerely,

/RA/

Dale E. Klein

Enclosure:
NRC Response to GAO Recommendations

cc: Representative Henry Waxman

Identical letter sent to:

The Honorable Tom Davis
Chairman, Committee on Government Reform
United States House of Representatives
Washington, D.C. 20515
cc: Representative Henry Waxman

The Honorable Susan Collins
Chair, Committee on Homeland Security
and Governmental Affairs
United States Senate
Washington, D.C. 20510
cc: Senator Joseph I. Lieberman

The Honorable George V. Voinovich
Chairman, Subcommittee on Clean Air,
Climate Change, and Nuclear Safety
Committee on Environment and Public Works
United States Senate
Washington, D.C. 20510
cc: Senator Thomas Carper

The Honorable Ralph M. Hall
Chairman, Subcommittee on Energy
and Air Quality
Committee on Energy and Commerce
United States House of Representatives
Washington, D.C. 20515
cc: Representative Rick Boucher

The Honorable Joe Barton
Chairman, Committee on Energy and Commerce
United States House of Representatives
Washington, D.C. 20515
cc: Representative John D. Dingell

The Honorable James M. Inhofe
Chairman, Committee on Environment
and Public Works
United States Senate
Washington, D.C. 20510
cc: Senator James M. Jeffords

The Honorable David M. Walker
Comptroller General of the United States
U.S. Government Accountability Office
Washington, D.C. 20548
cc: James E. Wells, GAO

The Honorable Rob Portman
Director, Office of Management and Budget
Washington, D.C. 20503

NRC RESPONSE TO GAO RECOMMENDATIONS

In its report, "Nuclear Regulatory Commission: Oversight of Nuclear Power Plant Safety Has Improved, but Refinements Are Needed" (GAO-06-1029), the U.S. Government Accountability Office (GAO) made recommendations for improving the U.S. Nuclear Regulatory Commission's (NRC's) ability to identify declining safety performance at nuclear power plants before significant safety problems develop. Specifically, GAO recommended that the NRC:

1. Aggressively monitor; evaluate; and, if needed, implement additional methods or processes to increase the effectiveness of its efforts under the reactor oversight process (ROP) to assess safety culture at plants.

2. In addition to periodically evaluating the effectiveness of its safety culture efforts, NRC may also be able, through its performance indicator program, to develop specific indicators to measure important aspects of plants' safety culture. Trends in these performance indicators could be useful feedback to NRC on its safety culture activities. The indicators could also provide useful information to the public and other NRC stakeholders on the safety culture at plants.

3. In the absence of performance indicators or other performance metrics for plants' safety culture, make publicly available, through the ROP Web site, consolidated and comprehensive data on the plants that have substantive, open cross-cutting issues to provide a more comprehensive picture of plant performance and provide insights into aspects of the plants' safety culture that otherwise are not readily available on the Web site.

NRC Response:

As noted in the GAO's report, the staff has taken significant actions to incorporate safety culture into the ROP. These efforts included: (1) implementing a multi-office ROP safety culture focus team to resolve implementation issues, to interface with internal and external stakeholders, and to evaluate and act on lessons learned; (2) revising guidance documents and inspection procedures to further define key safety culture aspects and prescribe when an independent assessment of a licensee's safety culture is warranted based on licensee performance; (3) conducting training for inspectors on the safety culture ROP changes; and (4) interacting with external stakeholders during the development phase, including the opportunity to provide comments on the draft ROP documents that incorporated the safety culture changes.

The staff is monitoring the implementation of the safety culture enhancements through the NRC safety culture focus team. An 18-month initial implementation period is under way, during which time the NRC staff will monitor and evaluate the effectiveness of the enhancements using performance metrics through its self-assessment process. We will determine the need to implement additional methods or processes to increase the effectiveness of the ROP based on the lessons learned during this initial implementation phase.

The NRC is revising Inspection Manual Chapter (IMC) 0307, "Reactor Oversight Process Self-Assessment Program," to add specific measures to determine the effectiveness of this important initiative. In support of this effort, we have added specific questions to the internal and external ROP surveys, which are currently being administered, in order to solicit feedback on the safety culture effort. The survey responses will be consolidated and analyzed, and the results will be presented in the annual performance metric report and discussed in the annual ROP self-assessment provided to the Commission for review.

NRC believes the annual ROP self-assessment process and performance metric report, versus the ROP performance indicator program, are better tools to gather and assess feedback on the safety culture enhancements. We will use these feedback processes to provide useful information to internal and external stakeholders, and make the ROP more efficient and effective at identifying declining licensee performance.

As a first step in the process, and as recommended by the GAO, we have added a Web page that presents consolidated and comprehensive data on the plants that have substantive, open cross-cutting issues. We plan further refinements to the ROP Web pages to more prominently highlight plants that have substantive cross-cutting issues to provide a more comprehensive picture of plant performance.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

November 22, 2006

EA-06-199

Duke Power Company, LLC d/b/a
Duke Energy Carolinas, LLC (Duke)
ATTN: Mr. B. H.Hamilton
Site Vice President
Oconee Nuclear Station
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (OCONEE NUCLEAR STATION - NRC INSPECTION REPORT NOS. 05000269/2006017, 05000270/2006017, AND 05000287/2006017)

Dear Mr. Hamilton:

The purpose of this letter is to provide you with the Nuclear Regulatory Commission's (NRC) final significance determination for a finding involving the failure of Duke's Oconee Nuclear Station to effectively control maintenance activities, and also the failure to assess and manage the risk, associated with removing an access cover in the south wall of the standby shutdown facility (SSF) to facilitate installation of temporary electrical power cables.

The finding was initially documented in NRC Integrated Inspection Report Nos. 05000269,270,287/2006002, which was issued on April 28, 2006. NRC Inspection Report Nos. 05000269,20,287/2006016, dated August 31, 2006, documented the NRC's assessment of the finding under the significance determination process, and concluded that the finding was a preliminary White issue (i.e., an issue of low to moderate safety significance which may require additional NRC inspection). The cover letter to our inspection report of August 31st provided Duke an opportunity to request a regulatory conference on this matter. In lieu of a regulatory conference, Duke chose to provide a written response, dated October 5, 2006.

Duke's written response documented its conclusion that a performance deficiency did not exist as described in the NRC's inspection report of August 31, 2006. Additionally, based on its review, Duke concluded that the NRC's SDP Phase 3 evaluation, the conclusions of which are inappropriately based on qualitative factors resulting in the preliminary White finding, support a conclusion that the resulting safety significance was actually very low (Green).

After carefully considering the information developed during the inspection and the information provided in Duke's written response, the NRC has concluded that the final inspection finding is appropriately characterized as White in the Mitigating Systems cornerstone. In response to Duke's contention that the matter does not represent a performance deficiency, the NRC notes that although the access penetration may have been constructed in 1992 to the best estimate of flood height (4.71 feet above grade) at the time, by 1993 Duke was aware that this estimate was flawed and non-conservative. By 1993, Duke was aware that predicted flood heights were much higher than 4.71 feet. In fact, the percentage of floods that were assumed to overtop the effective five foot flood barrier/wall and fail the SSF was estimated to be 20 percent. As such, removal of the access cover would directly impact the accredited effectiveness of the five foot flood barrier/wall. In 1999, a Duke maintenance rule expert panel re-assessed the safety significance of the SSF flood barrier and concluded that the safety significance was low. The panel considered the SSF wall as a passive feature with the only likely failure being a watertight doorway. However, the panel failed to recognize that the access cover was susceptible to maintenance which could create a bypass around the wall and degrade the function of the wall. Accordingly, since Duke's expert panel failed to act in consideration of the best available information at the time, the NRC considers the 1999 re-assessment of the SSF flood barrier to be non-conservative. Therefore, the staff concluded that Duke's subsequent failure to assess and manage the increased risk due to potential flooding that resulted from the SSF maintenance activity was a performance deficiency.

The NRC reviewed the information provided in Duke's written response, and concluded that changes to our preliminary analysis were warranted. In particular, the physical location of the SSF cover was revised to reflect the information provided by Duke. The NRC also agreed with Duke that an additional SSF success term (used in the staff's preliminary seismic evaluation) was redundant, and as such this term was deleted from the final analysis. The staff did not agree, however, with Duke's recommended change to the initial fragility term (for seismic considerations), as no new information was provided by Duke to support such a change. Applying these modifications to the final significance determination resulted in a change in core damage frequency that was consistent with the licensee's assessment.

Duke also stated that, given the uncertainty associated with the results of an external events analysis, a qualitative assessment was more appropriate. The NRC recognized that such consideration was appropriate in this situation, and considered other attributes that would have a bearing on safety significance. These included defense in depth and the ability to protect the public given an accident. The initiating events associated with this performance deficiency fall into the rare occurrence category. However, for these postulated accident sequences, there was an exclusive reliance upon the SSF to prevent core damage (no redundancy or diversity of mitigation). Any functional degradation of the SSF flood barrier from these initiating events directly increased the failure probability of the SSF and therefore, increased the likelihood of core damage. With a loss of core cooling, the fuel cladding and the Reactor Coolant System would eventually fail, causing a loss of multiple barriers that protect the public. The significance of the SSF flood barrier was clearly understood and delineated in Duke's Probabilistic Risk Assessment for External Events, dated December 1996. Given the nature of the initiating events associated with this performance deficiency, the emergency plan response would also be impaired. Consequently, from a blended qualitative and quantitative perspective the NRC's final Significance Determination remains low to moderate (White).

You have 30 calendar days from the date of this letter to appeal the staff's determination of significance for the identified finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

The NRC also has determined that this finding resulted in a violation of regulatory requirements. In this case, the failure to use adequate procedures to control maintenance activities that could affect safety-related equipment was determined to be a violation of Technical Specification 5.4.1. As a result, Duke failed to assess and manage the increase in risk from external floods for this maintenance activity, as required by 10 CFR 50.65(a)(4). The violation is cited in the enclosed Notice of Violation (Notice), and the circumstances surrounding the violation are described in detail in NRC Inspection Report Nos. 05000269,270,287/2006016, dated August 31, 2006, and NRC Integrated Inspection Report Nos. 05000269,270,287/2006002, dated April 28, 2006. In accordance with the NRC Enforcement Policy, the Notice of Violation is considered escalated enforcement action because it is associated with a White finding.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

For administrative purposes, this letter is issued as a separate NRC Inspection Report, No. 05000269,270,287/2006017, and the above violation is identified as VIO 05000269,270,287/2006017-01, White Finding - Inadequate Procedural Controls and Risk Management Associated with Breach in SSF Flood Protection Barrier. Accordingly, apparent violations AV 05000269,270,287/2006016-01 and AV 05000269,270,287/2006016-02 are closed.

Because plant performance for this issue has been determined to be in the regulatory response band, we will use the NRC Action Matrix to determine the most appropriate NRC response for this event. We will notify you by separate correspondence of that determination.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS) which is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, any response should not include any personal privacy, proprietary, classified, or safeguards information so that it can be made available to the public without redaction. The NRC also includes significant enforcement actions on its Web site at www.nrc.gov; select **What We Do**, **Enforcement**, then **Significant Enforcement Actions**.

Should you have any questions regarding this letter, please contact Mr. James Moorman, Chief, Branch 1, Division of Reactor Projects, at (404) 562-4647.

Sincerely,

/RA/ by Victor McCree Acting for/

William D. Travers
Regional Administrator

Docket Nos.: 50-269, 50-270, 50-287
License Nos.: DPR-38, DPR-47, DPR-55

Enclosure: Notice of Violation

cc w/encl: (See page 4)

cc w/encl:
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Compliance Manager (ONS)
Duke Energy Corporation
Electronic Mail Distribution

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*See Previous Concurrence

X PUBLICLY AVAILABLE X NON-SENSITIVE

ADAMS: X Yes ACCESSION NUMBER: ML063260154

OFFICE	RII:EICS	RII:DRP	RII:DRS	RII:ORA	NRR	
SIGNATURE	CFE	CAC	WGR	VMM		
NAME	EVANS	CCASTO	WROGERS	VMcCree		
DATE	11/02/2006	11/02/2006	11/03/2006	11/22/2006		
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO		

OFFICIAL RECORD COPY DOCUMENT NAME: C:\FileNet\ML063260282.wpd

NOTICE OF VIOLATION

Duke Power Company
Oconee Nuclear Station
Units 1, 2 and 3

Docket Nos. 50-269, 50-270, 50-287
License Nos. DPR-38, DPR-47, DPR-55
EA-06-199

During an NRC inspection completed on August 31, 2006, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> Technical Specification 5.4.1 requires that written procedures shall be established implemented and maintained as recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Section 9, Procedures for Performing Maintenance, requires that maintenance which can affect the performance of safety-related equipment should be properly preplanned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances.
>
> 10 CFR 50.65 (a)(4), "Requirements for monitoring the Effectiveness of Maintenance at Nuclear Power Plants" requires in part, that prior to performing maintenance activities, the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities.
>
> Contrary to the above, on August 13, 2003, while performing planned maintenance involving the opening of a penetration in the Standby Shutdown Facility (SSF) exterior wall to route temporary electrical power cables, the licensee failed to use an adequate procedure to open and control a penetration through a passive flood protection barrier and route temporary power cables. Specifically, the procedure used, IP/0/A/3010/006, Cable Pulling Procedure, Revision 16, did not address the installation of temporary power cables, and did not address breaching and restoring a flood barrier. As a result, the licensee failed to assess and manage the increase in risk associated with the degradation of the flood protection capability of the SSF's exterior wall from August 13, 2003 to August 3, 2005.

This violation is associated with a White significance determination process finding for Units 1, 2 and 3 in the Mitigating Systems cornerstone.

Pursuant to the provisions of 10 CFR 2.201, Duke Power Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator, Region II, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice of Violation (Notice) within 30 days of the date of the letter transmitting this Notice. This reply should be clearly marked as a "Reply to a Notice of Violation; EA-06-199" and should include: (1) the reason for the violation or, if contested, the basis for disputing the violation or severity level; (2) the corrective steps that have been taken and the results achieved; (3) the corrective steps that will be taken to avoid further violations; and (4) the date when full compliance will be achieved. Your response may reference or include previously docketed correspondence if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS) accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within 2 working days.

Dated this 22nd day of November 2006

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

November 20, 2006

NRC INFORMATION NOTICE 2006-26: FAILURE OF MAGNESIUM ROTORS IN MOTOR-OPERATED VALVE ACTUATORS

ADDRESSEES

All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to inform addressees of recent failures of motor-operated valve (MOV) actuators that were attributed to the oxidation and corrosion of the magnesium motor rotor fan blades and shorting ring resulting from exposure to high humidity and temperatures. This IN serves to reaffirm the necessity of adequate inspection and/or preventive maintenance on MOV actuators manufactured with magnesium rotors to ensure the safe operation of nuclear power facilities. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this IN are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

A recent NRC staff review of MOV actuator failures at certain plants identified the following examples:

1. Failure of a Main Feedwater Isolation Block Valve to operate automatically (Crystal River 3; October 28, 2005; Licensee Event Report (LER) 50-302/2005-004-00). The licensee attributed this failure to the corrosion and oxidation of the magnesium fan blades and shorting ring of the motor rotor as a result of exposure to high humidity and temperatures.

2. Failure of the Residual Heat Removal (RHR) Cold Leg Injection Valve to open when placing RHR in operation for cooldown (Turkey Point 3; March 6, 2006; LER 50-250/2006-003-00). The licensee attributed this failure to the corrosion and oxidation of the magnesium fan blades and shorting ring of the motor rotor as a result of exposure to high humidity and temperatures.

ML062070124

3. Failure of a Recirculation Pump Suction Valve to operate (Browns Ferry 3; January 15, 2006; documented in the licensee's corrective action program). The licensee attributed this failure to the corrosion of the motor rotor fan blades and shorting ring.

BACKGROUND

Many safety- and non-safety-related MOVs utilize Limitorque actuators with Reliance motors or a similarly styled design by a different manufacturer. Based on torque requirements, aluminum and magnesium alloy cast-squirrel-cage rotors are utilized in MOV actuators. Valve actuators with a motor maximum torque of 40 foot-pounds force (54 Newton-meters) are typically aluminum, and magnesium actuators are used for applications requiring greater than 60 foot-pounds force (81 Newton-Meters).

The typical magnesium rotor is made of stacked, steel punched core plates with AM100A magnesium alloy (approximately 90% magnesium, 10% aluminum, 0.1% manganese) components—the conductor bars, end rings, and cooling fan blades—cast to complete the rotor. While magnesium provides higher torque through its higher resistivity, this relatively brittle cast alloy is susceptible to shrinkage cracking and gas porosity. Specifically, magnesium rotors are susceptible to three main failure mechanisms: galvanic corrosion, general corrosion, and thermally induced stress.

The first failure mechanism is galvanic corrosion. Following manufacture, the electrical potential difference between the magnesium and the steel core is 1.9 volts creating the conditions for galvanic corrosion, with the most vulnerable area being the interface between the steel core and the magnesium end ring. Most manufacturers alleviate this by protecting the magnesium end rings with a paint and/or lacquer coating. Though the rotor might be initially protected, even the smallest scratch or chip in this exterior coating will cause localized, accelerated corrosion in the form of magnesium hydroxide (MgOH) powder. The formation of MgOH powder leads to rotor cracks that add to the existing problems of shrinkage cracking, gas porosity, and MgOH volume difference. Motor overheating events (typically due to locked rotor conditions) accelerate this coating degradation. A propagating crack at the interface between the stacked core and the end ring causes a high resistance connection with the end ring, which in turn causes a high current density (due to current redistribution) on the opposite side of the rotor. This increased current density increases the temperature on that side of the rotor resulting in thermal stress. At the steel-magnesium interface, the higher temperature may melt the magnesium into small beads. These thermally-stressed rotor areas and the melted magnesium beads then provide new opportunities for coating degradation and cracking resulting in new areas of high resistance between the stacked core and end ring and new areas of the rotor with a higher current density. This cycle of events can then repeat around the rotor.

The second major failure mechanism affecting magnesium rotors is general corrosion. Most actuator motors for safety-related MOVs that are located in potentially harsh environments have T-drain pipe plugs to allow moisture to escape. These same plugs allow moisture to enter and condense inside the motor. This moisture leads to the formation of MgOH and magnesium oxide (MgO_2). The white MgOH powder can form a light haze on the inside of the motor without impacting its operation. However, MgOH and MgO_2 can form beads between core plates (from the magnesium conductor bars) and at the interface between the stacked core and the end ring causing high resistance points and the high current density phenomena stated above and even further cracking. The rate of general corrosion increases in a higher humidity operating environment.

The third major failure mechanism affecting magnesium rotors is thermally induced stress which reveals itself in different ways. First, because galvanic corrosion is thermally catalyzed, the corrosion rate increases with temperature, with a significant increase in the corrosion rate occurring at temperatures above approximately 93 °C (200 °F). The rate of galvanic corrosion increases when the motor is located in a higher temperature environment, as well as during general motor high-current conditions and/or within the high current density regions mentioned earlier. Secondly, magnesium has twice the thermal expansion coefficient of steel. This produces uneven axial and radial forces across the rotor causing further cracks in the magnesium and its paint and/or lacquer coating. Finally, many rotors experience significantly higher temperatures because their thermal overloads are set higher than the recommended 10 to 15 seconds for locked rotor current conditions (in order to ensure safety-related function as given in NRC Regulatory Guide 1.106, Revision 1, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves"). For example, some rotors reach 700 °F (371 °C) in 15 seconds, and temperatures of 700 °F to 850 °F (371 °C to 454 °C) cause a significant loss of magnesium yield strength.

Various laboratory tests have been conducted to better understand magnesium rotors. General Electric (GE) tested-to-failure 3 motors in varying aged and environmental conditions, with the most limiting failure being a new motor which failed after 43 days in a high temperature environment under a maximum temperature of 223 °F (106 °C). The Institute of Electrical and Electronics Engineers (IEEE) inspected 14 magnesium rotors and discovered 5 showing varying levels of degradation. Finally, IEEE reviewed plant motor failure rates and found magnesium rotors failing at three times the rate of aluminum rotors.

The following documents address similar MOV failures with related technical details:

- NRC Information Notice 86-02, "Failure of Valve Operator Motor during Environmental Qualification Testing," January 6, 1986: this IN reported on the results of the previously discussed GE laboratory test on three motors in response to issues at the River Bend and Nine Mile Point 2 nuclear power stations. In addition to the technical details stated earlier, the NRC within this IN suggested that licensees review the qualification of these motors in their Design Basis Event applications.

- NUREG/CR-5404, ORNL-6566/V1, "Auxiliary Feed Water Aging Study," July 1993: while this report is extensive and covers many wide-ranging aspects, Section 4.5 (Alternate Methods of Valve Actuator Motor Testing) reviews two methods for the preventive maintenance of magnesium rotors.

- NUREG/CR-6205, ORNL-6796, "Valve Actuator Motor Degradation," December 1994: this NUREG provides a detailed review of the technical phenomena citing all of the failure mechanisms with insights from the GE test and the IEEE report.

- IEEE Transactions on Energy Conversion, Vol. 3, No. 1, "An Investigation of Magnesium Rotors in Motor Operated Valve Actuators," March 1988: this IEEE report provides a detailed, technical analysis of the failure mechanisms and material impact of magnesium rotors. This analysis includes the review of various laboratory tests and licensee database reviews. This report includes a detailed inspection procedure for user guidance.

The IEEE report, NUREG/CR-5404, Crystal River LER 50-302/2005-004-00, Turkey Point LER 50-250/2006-003-00, and the operating experience from Browns Ferry provide some specific methods for preventive maintenance:

1. The IEEE report and the LER's from Crystal River and Turkey Point provide detailed inspection procedures with acceptance criteria. They specifically discussed boroscopic inspections of MOV actuators through the T-drain pipe as a preventive maintenance method.

2. The Crystal River LER also provides detail on performing electrical Polarization Index inspections from measurements of the motor winding insulation resistance.

3. NUREG/CR-5404 reviews motor current signature analysis as a method for revealing broken or distorted rotor bars.

4. The IEEE report reviews ideal thermal overload setpoints in order to avoid the thermally induced stresses discussed earlier but also proposes graduated inspection criteria if these setpoints are not met.

5. Operating experience from Browns Ferry describes their consideration of duty cycle limitations to ensure the motors are not actuated without a proper cooldown interval in order to avoid or not exacerbate thermally induced stresses.

DISCUSSION

Recent failures of MOV actuators as a result of galvanic corrosion, general corrosion, and/or thermally induced stress highlight the particular vulnerabilities of motor actuators with magnesium rotors, particularly when the motor is located in a high humidity and/or high temperature environment. These MOV failures illustrate the necessity of adequate inspection and/or preventive maintenance on actuators manufactured with magnesium rotors.

CONTACTS

This information notice requires no specific action or written response. Please direct any questions about this matter to the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation project manager.

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Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under the Electronic Reading Room/Document Collections.

A Public Servant to the Last

Before Stepping Down, He Calls a New Generation to Serve

By Elizabeth Williamson
Washington Post Staff Writer
Wednesday, December 6, 2006; A23

In 1961, Ed McGaffigan Jr. was a seventh-grader from Boston watching the inauguration on television when a president told him to ask what he could do for his country.

The son of an Irish immigrant laborer, McGaffigan had a ready answer, as did many of his generation: He could work for his country.

That inspired a three-decade career that included stints at the State Department, the White House, the U.S. Embassy in Moscow, Capitol Hill and the Nuclear Regulatory Commission, where last month McGaffigan became the agency's longest-serving commissioner.

McGaffigan, who turns 58 on Friday, would normally have many more years of service ahead. Instead, he is fighting an illness that threatens to vanquish him, and he is taking on one last assignment: closing his career in a way that inspires others to consider public service.

"I do what I do out of a deep sense of appreciation for the opportunities that this country gives people like my father and me," he said. "I'm proud to have been there, and proud to serve with a bunch of people as dedicated as I am.

"I hope there's another generation."

One recent rainy morning, McGaffigan, a physicist with thick gray hair and a runner's build, took his place at the front of the NRC's conference room for a celebration of his career. A line of masking tape on the carpet showed him where to stand, and he placed the toes of his shoes precisely on it.

There were jokes, gifts and tributes. "He can quote the most obscure regulations and give exact details on how they were written," said fellow commissioner Pete Lyons. He presented McGaffigan, a serial marathoner, with a specially produced audiobook to listen to while he runs: a recitation of 10 CFR 3240, "tests required for tritium-powered auto lock illuminators," a reg so obscure it stumped even McGaffigan.

Then it was McGaffigan's turn. "As long as I'm here," he said, "I'm going to be dedicated to making you all improve."

He wept a little. But he did not take his shoes off the tape.

McGaffigan was the first in his family to attend college, earning a physics degree, with honors, from Harvard, and master's degrees from the California Institute of Technology and Harvard's Kennedy School of Government. Since his education was underwritten by taxpayers, he decided, he said, to give them the benefit of the technical expertise they paid for.

While at Cal Tech, he took the foreign service exam, and in 1976 he joined the State Department. During that time, he served in the White House Office of Science and Technology Policy, overseeing international scientific programs, worked for two years in the U.S. Embassy in Moscow, reporting on science, technology and atomic energy.

"I'm sort of an omnivore when it comes to scientific knowledge," he said. "That gave me an advantage."

In 1982, he married Peggy Weeks, whom he met through a friend. Through his former boss at State, Kennedy School professor Joseph Nye, he met Sen. Jeff Bingaman (D-N.M.), and for the next 14 years he advised the senator in his work on the Senate Armed Services Committee.

The McGaffigans had a son, Edward, in 1984, and a daughter, Margaret, two years later. Soon after, Peggy McGaffigan developed Huntington's disease, a degenerative disease of the nervous system. Bingaman reduced McGaffigan's workload, giving him time to care for his wife and children.

"He was always very organized, always making lists," said Margaret, who goes by Meggy. Grocery lists, family appointments, the annotated Christmas shopping list his children found -- for a decade, his reams of routine have kept his household on task.

Meggy McGaffigan, a physics student, has begun an internship at the NRC, where, at least for a time, she'll work alongside her father.

"He's always told me and my brother that we can do what we want" as a career, she said. But it has been her father's dedication to public service "that kind of makes you wonder if that's something that we want to do."

Her father hopes other young idealists follow his path.

"With Kennedy, serving government was a noble cause," he said. "Now Republicans and Democrats alike bash government when it serves their purpose.

"We're going to need a massive influx of young people. The answer is not contracts." Outsourcing the work of government technicians, he said, means "there are agencies where they can't do the calculations themselves any more."

When McGaffigan was appointed to the NRC in 1996, his government experience helped the commission unravel a tangle of poor policies "one by one."

Today, he says, the NRC is "much more efficient, much more timely, more fair I believe and more transparent." At the height of this effort, in 1999, McGaffigan was diagnosed with melanoma and underwent extensive surgery. The following year, Peggy McGaffigan died. The day after her funeral, Ed McGaffigan was testifying on the Hill. "What we do is we go on," he said. "It's part of the duty thing."

Throughout his illness, McGaffigan has rarely spent more than a few days off the job.

"The odds are very, very stacked against me," he said. "But I'm going to remain commissioner as long as I'm of use."

Jeff Sharkey, McGaffigan's chief of staff, calls his boss "one of my heroes."

"He invests his entire character in this mission," he said, pausing to compose himself. "You want to do your very best for him."

A few weeks ago, McGaffigan asked his staff to help him write his obituary.

Filled with detailed instructions and proud anecdotes, "it shows what's important to him," Sharkey said. "He continues to be someone who shows others the way, . . . how to deal with things honorably, with grace and with dignity."

Today, Ed McGaffigan and his daughter will drive together to work. He's showing her the ropes, because too soon, Meggy McGaffigan will be working there alone.

"It's one of those things that goes unsaid," she said. Anyway, she knows what her father would tell her.

"Just go on to work like any other day," she said. "Just go on."

Inside NRC

Volume 28 / Number 24 / November 27, 2006

Cracking indications at Wolf Creek lead to wider PWR weld questions

The discovery of unusual indications of cracking in Wolf Creek pressurizer welds has prompted NRC and the industry to comb through data gathered from Wolf Creek and other PWRs and might lead to a reassessment of assumptions that are the basis of an inspection regime for such welds. The Wolf Creek indications do not represent an "immediate safety concern," industry officials emphasized at a November 16 meeting with NRC staff at agency headquarters in Rockville, Maryland. John Grobe, the associate director of engineering and safety systems in NRC's Office of Nuclear Reactor Regulation, did not dispute the point but he said he wanted to decide around mid-December whether there was a "need for NRC to take timely regulatory action."

Five flaws were discovered in October, during Wolf Creek's fall refueling outage, according to presentations at the meeting by Terry Garrett, Wolf Creek Nuclear Operating Co.'s vice president for engineering, and Mike Robinson of Duke Energy and the Electric Power Research Institute's Materials Reliability Program, or MRP.

The cracks were 1, 2.75, 3.75, 5 and 11.5 inches long, the presentations said. All five were circumferential, they said.

In the NRC presentation, NRR's Ted Sullivan said the length of the flaws and the fact that they were circumferential were "what got our attention," since both features were unexpected. Long flaws tend to leak, and to rupture, in less time than short ones, Sullivan said. Circumferential cracking is potentially more safety significant than axial cracking.

The indications of cracking were in alloy 82/182, variants of alloy 600 used in welds. Cracking of alloy 600 and its variants has been a focus of industry and NRC attention for the past six years.

The "most probable mechanism" for the Wolf Creek flaws is primary water stress corrosion cracking, or PWSCC, Garrett said. But he said certain features did not seem to support that hypothesis. Industry and NRC participants at the meeting said the few recent examples of circumferential indications at similar reactor locations — that is, at "butt welds" between dissimilar metals — were significantly shorter.

The NRC and the industry officials agreed that there were important gaps in the information on the Wolf Creek flaws, and they said they would exchange data as they sought to determine whether the indications represented an anomaly or an unexplained degradation mechanism. There is a "whole variety of variables," making analysis "extremely challenging," Grobe said. There is "a lot of data that could have been obtained that was not," because Wolf Creek repaired the flaws without first taking samples of the material, he said. While that would have been helpful, he said, he acknowledged that it was not required under NRC regulations.

In an interview after the meeting, he said that there are "reasons not to [take the material sample] unless you really need to do it." It is time-consuming and could involve a significantly increased radiation dose, he said.

Role of weld stresses
Also hindering the Wolf Creek analysis, he said, is the lack of records on the weld repairs that might have been made there. Decades ago, he said, there was less understanding than there is today of the effects of residual stresses on welds. Such stresses may be a "driver" of the Wolf Creek flaws, he said.

Time and temperature — which are generally seen as the main factors in PWSCC — do not appear to be the main causes at Wolf Creek, he said. Temperatures in pressurizers are fairly uniform from one reactor to another, he said. (In contrast, reactor vessel heads — where fairly widespread PWSCC was discovered several years ago — have significant variations in temperature.)

According to data presented by the MRP at the meeting, Wolf Creek falls squarely in the middle, in terms of age, among reactors whose pressurizers have been inspected under a new program the MRP is in the process of implementing. Alex Marion, the Nuclear Energy Institute's senior director for engineering, said in an interview that the industry is "going through somewhat of a learning curve" as it completes the initial rounds of butt-weld inspections. The regime for those inspections is spelled out in a document known as MRP-139.

Under MRP-139, initial inspections of butt welds are to be completed by 2010. Pressurizers are a priority because of their high temperature; utilities are supposed to do those inspections by the end of 2007.

Anomaly?
In the course of carrying out the inspections, the industry expected to find some indications of cracking, Marion said. But Wolf Creek was a "surprise," he said, because PWSCC typically has "a certain morphology," and the Wolf Creek flaws did not conform to that pattern.

Wolf Creek could turn out to be an "anomaly," or it could be "of significance." Marion said. "We really don't know," he added.

NRC staffers are "doing the right thing" to "assimilate and gather as much information as they possibly can," and to "compare assumptions" with the industry, Marion said.

However, the situation is not "an emergency," he said. "We are in agreement this needs to be investigated" but do not necessarily agree that NRC has to make a decision "immediately," Marion said.

At the meeting, NRC officials said that if they were going to take some regulatory action, they would have to do so fairly soon if they wanted to have an impact on the spring 2007 refueling outages. But NRC already has "missed that window," Marion said.

He pointed to actions the industry already is taking in response to the Wolf Creek findings. During the meeting, Robinson said information gleaned from the analysis of those findings will be "plowed back" into an updated version of MRP-139. Also, he said, utilities that are thinking of requesting a "deviation" from the MRP-139 will be asked to check the "technical basis" for their request. MRP-139 is a part of a larger industry initiative on materials degradation. One aspect of that initiative is that the industry is to police itself, with input from NRC but no direct regulatory involvement. NRC has balked at the idea for a self-policed inspection regime (INRC, 8 August '05, 1) but has narrowed its differences with the industry over the specifics of the butt-weld inspection plan.
—Daniel Horner, Washington

NRC says no, for now, on Browns Ferry-1 restart

NRC commissioners told the staff it is too soon to authorize the Region II administrator to allow restart of Browns Ferry-1, which the Tennessee Valley Authority now thinks could occur as early as February.

In a November 15 staff requirements memorandum (SRM), the commissioners said they would be willing to reconsider the decision after a briefing scheduled in January. Following that meeting, the commissioners said they would provide further direction in an SRM. In an October 20 paper (Comsecy 06-52), the staff asked the commissioners to authorize the administrator to allow the restart. The paper also provided a status report on the project.

All three Browns Ferry units were shut in 1985 in a management collapse. Unit 2 restarted in 1991, and unit 3 in 1995. In its status report, the staff said TVA plans to load fuel into unit 1 in December and place the reactor in mode 2 (startup) in May 2007. But TVA also has advised the staff it is "attempting to improve on" that date to as early as February, the report said. The staff said it has completed a flexible inspection plan that could provide the necessary oversight for a restart date as early as February.

The staff said it is expediting its activities to ensure that regulatory activities do not result in any significant delays for the restart.

The staff's master schedule outlines agency actions and processes that must be completed before the regional administrator, in consultation with the director of the Office of Nuclear Reactor Regulation, would be prepared to authorize restart.

The staff said TVA has completed "a substantial portion" of its planned construction activities, including piping and component replacement. As construction activities near completion, TVA is focusing on returning rebuilt and refurbished systems to service and transferring control of those systems from the construction organization to the Browns Ferry operations staff, it said.

TVA has also completed many of the licensing activities associated with restart, it said. Of the 21 license amendments identified by TVA as required to revise the technical specifications prior to restart, 17 have been issued, the staff said.

The staff said it is reviewing TVA's completion of about 49 generic communications, including generic letters and bulletins, which had not been previously implemented at unit 1.

Hold off

The commissioners were unanimous in telling the staff to hold off on restart approval.

Chairman Dale Klein said he approved the concept of authorizing the Region II administrator to allow restart of unit 1 "at the appropriate time." He said, "Now, however, is not the appropriate time."

Klein said the staff should ensure completion of all necessary actions and brief the commissioners prior to making a determination regarding restart.

Commissioner Edward McGaffigan, like other commissioners, congratulated the staff on developing and implementing a restart program. However, he said, given the scope and extent of the recovery process and regulatory oversight effort, "It should thus come as no surprise that a great many novel technical issues, programmatic concerns, and policy questions need to be assessed by the Commission itself prior to gaining confidence that BFN Unit 1 is again ready to operate."

McGaffigan said the magnitude of the recovery program "rendered BFN Unit 1 a *de facto* construction site, with significant piping and component replacement, electrical recabling, and overall refurbishment."

Commissioner Jeffrey Merrifield noted that, for the previous restart decisions at Browns Ferry-2 and -3, the commissioners received information, including papers and briefings on actions taken by the licensee, before they delegated the restart decision to the regional administrator. "The staff has not provided any new information that would support a change in the process previously used by the Commission to establish the readiness of a plant to restart," he said. Commissioner Gregory Jaczko outlined steps he said he believed were necessary before he could make an informed decision regarding restart, including holding a public meeting. He also raised the issue of noncompliance with fire protection regulations. "Licensees, including TVA, should consider a comprehensive solution by implementing the riskinformed fire protection regulations," he said.

The staff said TVA's current approach includes use of manual operator actions to address individual instances of noncompliance with fire protection regulations. The staff said it has inspected the program and is reviewing those results, and will evaluate the need for additional fire protection inspections.

Jaczko also said he has concerns with the current planned review of the extended power uprate application for unit 1.

Browns Ferry-1 is rated at 1,098 MW and units 2 and 3 at 1,155 MW. TVA plans to uprate all three units to 1,280 MW. The staff noted that TVA has revised its unit 1 application, which originally sought a 20% uprate. The revised submittal includes analyses for a 20% uprate, but will initially limit the power increase to 5%, it said. The change is in response to NRC concerns about TVA's earlier analyses related to steam dryer structural adequacy and credit for containment overpressure for emergency core cooling system analyses. To resolve the steam dryer concerns, TVA plans to collect plant-specific steam dryer performance data at 105% power. TVA will then submit a revised steam dryer performance analysis to seek approval for operation at 120% power, the staff said.

The staff said it does not believe a full Advisory Committee on Reactor Safeguards review is needed for restart at 105% power. Browns Ferry-2 and -3 were uprated by 5% power in 1998, and unit 1 is similar in design and operation to those units, it said. Assuming the ACRS agrees, the staff said it expects to complete its review of the uprate to 105% by next month, which would allow restart as early as February. Full ACRS review of the 105% level would delay the issuance of the uprate amendment by three to four months, it said. The ACRS would review the staff's safety evaluation for the 120% uprate level in its entirety, it said.

Jaczko questioned whether the staff's anticipated review of the uprate at 105% power is actually a review of the TVA analysis that supports a 20% uprate, but simply restricts unit 1 to a 5% power uprate. If that is the case, the ACRS should review the TVA uprate analysis prior to a staff decision on the issue, he said.

Referring to issues raised by other commissioners, Commissioner Peter Lyons said he believed the January commission meeting "will provide an appropriate opportunity for the staff to present the results of its oversight and the remaining activities necessary to provide an adequate regulatory basis for authorizing restart and to address Commissioner questions on matters of particular interest. The Commission may then provide direction if deemed necessary," he said.—***Tom Harrison, Washington***

Staff to review the effectiveness of NRC's risk-informed regulation

NRC staff plans to perform an assessment of the future of risk-informed regulation and will establish an "effectiveness review process" for such activities.

In an update of NRC's risk-informed regulation implementation plan, known as the RIRIP (Secy 06-217, released November 9), NRC Executive Director for Operations Luis Reyes said that the staff would "perform an assessment of where the agency should take risk-informed regulation in the short term (i.e., 1-5 years) and, if possible, the long term (i.e., 5-10 years). This assessment will factor in commission guidance and input received from stakeholders."

Reyes said the results of these assessments would guide the staff in determining which risk-informed activities should be closed out and what new activities were needed. Reyes also said the staff believes that there should be "a risk-informed vision" developed for each of the agency's three core areas — reactors, materials, and waste — and specific goals for each of the three NRC functional areas — oversight, licensing/certification, and rulemaking and guidance development.

The staff concluded that "it may be necessary to separate the reactor arena" into three categories — operating reactors, new reactors, and advanced reactors — in order "to facilitate development of a clear vision and specific goals. This is due to the large difference in the extent to which these reactor categories could feasibly be risk-informed and performancebased," Reyes said.

The staff will "restructure the RIRIP around these arenas to facilitate a clear understanding of the agency's plan. For each activity that is determined to be necessary to meet the vision and goals, the staff will develop a program plan that contains specific milestones and deliverables," he said.

Reyes said that the staff anticipates "significant differences in the visions and goals established for the various arenas because of such factors as (1) the inherent major differences in the complexities and risk associated with NRC-regulated license activities (e.g., a nuclear power plant versus a sealed radiation source), (2) the state-of-the-art with regard to PRA [probabilistic risk assessment] technology and methods (i.e., PRA methods are relatively well developed for the reactor arena versus the materials and waste arenas), (3) the level of commitment of stakeholders in the various arenas interested in pursuing risk-informed activities, and (4) the potential costs and benefits associated with adoption of risk informed approaches."

Effectiveness review

The staff said in Secy 06-217 that "the new effectiveness review process would focus on determining whether completed RIRIP activities have achieved the desired outcomes and, if not, why not. In addition, the effectiveness review process would identify any needed corrective actions and lessons to be adopted as best practices for future activities." The staff said it will "evaluate the assessment feedback mechanisms" in NRC's reactor oversight process and operating experience program for insights to develop the effectiveness review process.

"As the environment evolves and using the results of the RIRIP effectiveness reviews, NRC senior management will modify and update the goals consistent with the established vision," the staff said in its paper.

Reyes said in the paper that the proposals constituted part of the staff's response to commission direction in a June 1 staff requirements memorandum, which followed up on a May 3 commission briefing on risk-informed regulation (INRC, 15 May, 5). In the June SRM, the commission said that the staff should improve the RIRIP "so that it is an integrated master plan for activities designed to help the agency achieve the Commission's goal of a holistic, risk-informed and performance-based regulatory structure. The plan should continue to give priority to risk-informed activities underway and incorporate lessons learned from earlier activities as appropriate."

Biff Bradley, risk assessment manager at the Nuclear Energy Institute, said in a November 21 interview that it was a good idea for the NRC to have "a more tangible set of objectives" and "more tangible goals" in the area of riskinformed regulation. "We've had some prior issues that have taken years and years" to be resolved, he said. The ultimate measure of effectiveness will be whether and how the new process impacts NRC review of implementation of industry's risk-informed initiatives by specific licensees at their power reactors, he said.

"How NRC can get these initiatives implemented [at] a large number of plants" represents "a significant challenge" for both industry and the agency, Bradley said. Industry representatives have said in the past that management at some plants has been wary about moving forward with risk informed initiatives absent demonstration of tangible benefits (INRC, 27 June '05, 5).

"The staff should give priority to the development of such regulations and processes most likely to be utilized," and "should ensure that processes are in place to resolve issues in a timely manner, including raising issues to senior management and/or the Commission, as appropriate," the commission said in its SRM.

Secy 06-217, including the updated RIRIP, is available on NRC's web site (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2006/secy2006-0217/2006-0217scy.pdf).—***Steven Dolley, Washington***

Inside NRC

Volume 28 / Number 23 / November 13, 2006

New reactors add sense of urgency to resolving digital I&C issues

With only a year to go before the first new plant license requests are filed, industry officials say it is urgent that NRC address the requirements for digital technology that will be installed in new reactors, or used as upgrades in existing units.

The current fleet of reactors relied on analog technology, which is gradually becoming obsolete. Reactor control rooms full of switches, annunciators, and panel-mounted meters will be replaced in the future with flat-screen displays and keyboards that fit into a fraction of the space.

Amir Shahkarami, who chairs the industry's Digital Instrumentation & Control (I&C) and Human Factors Working Group, told NRC commissioners at a November 8 meeting that digital controls would be crucial for the future of the industry. He said the technology would improve reliability and performance, and ultimately enhance safety. Another benefit of the technology is that it would reduce the burden on operators, he said.

He stressed that digital technology is not new and that it has been adopted by other industries and countries as "superior to analog technology." He said digital technology has been used by the airlines, military, aerospace, energy and petroleum industries, and it has been used in nuclear power plants built abroad. In the US, the applications have been limited to secondary systems and some primary systems, though none of the primary systems have been fully integrated, he said. Commissioner Peter Lyons agreed that digital technology has been successfully used by other industries, but he noted that the transition has not been without some "hiccups."

Shahkarami said the industry wants to follow the license renewal model and have the agency establish a digital I&C/human factors steering committee, with senior managers involved from four major offices — Office of Nuclear Reactor Regulation, Office of Nuclear Regulatory Research, or RES, Office of New Reactors and Office of Nuclear Safety and Incident Response. He asked the commissioners to maintain close watch by holding periodic briefings, and he suggested there be frequent meetings with NRC staff and other stakeholders. In addition, Shahkarami urged there be an industry-NRC meeting set up soon to discuss research activities relating to new and existing I&C issues. The results of the research studies are expected to support changes in existing guidance and used in developing new guidance. "From a new plant perspective," Shahkarami said, "this is not good." That's because design plans are nearly complete, he said. But Jack Bailey of the Tennessee Valley Authority, who heads the industry's New Plant Working Group, said that because work is well under way for the "first wave" of plant applicants, the fully digital plants likely would come in the following wave of new plants.

Even for existing plants, there needs to be clarification on the requirements, Bailey said. He said TVA initially wanted to upgrade the emergency core cooling system and another system in refurbishing Browns Ferry-1. But the staff cautioned that the review schedule would be extended, so TVA decided to stick to analog technology, he said. Bailey said it was not the NRC staff's fault that the industry is now pressuring for an acceleration of I&C work.

"The industry could have started two years ago," he said. But at this point, it will have to be "dealt with urgency on both sides," he said.

Shahkarami said the staff should allow industry representatives to be involved in revising guidance on diversity and defense in depth, which the staff has dubbed D3. The guidance will address digital control rooms. Although the staff began working on the guidance changes in June, the industry has been shut out of the process, he said. He urged the commissioners not to endorse any of the recommendations until the industry has had an opportunity to provide input.

Defense in depth

Having a backup system in place — defense in depth — is central to the debate on implementing digital I&C. Commissioner Edward McGaffigan noted that Finnish regulators decided that analog backup was needed for some digital components.

A representative from the Electric Power Research Institute said that digital software could be used just like other components as backup.

NRC Office of Nuclear Reactor Regulation Director Jim Dyer told the commissioners that the staff recognizes that digital I&C is a "rapidly evolving" issue. But he said he did not believe there needed to be commission oversight. McGaffigan said later during the briefing, however, that staff should not be surprised to get some direction from the commission.

Dyer said the staff and industry only recently sat down to begin discussing a plan for resolving digital technology issues. Both sides were able to identify key issues for follow-up during an October 19 meeting, he said.

Dyer said there is recognition in revisions to the Standard Review Plan and DG-1145, the draft regulatory guide for combined construction permit-operating license applications for LWRs, that a shift is under way to digital technology, particularly in four areas: the control systems, protective systems, mitigation systems, and monitoring systems.

Although digital technology "offers improved safety and diagnostic capability," it also can lead to more frequent common cause failure, Dyer said.

Bill Kemper of RES said the D3 effort is focused on addressing common cause failures. That work is expected to be completed in August. Also, he said, the RES staff is working on developing a risk-informed review method. In his presentation to the commission, he said other industries have used reliability and risk methods. A draft work product is targeted for issuance in late 2007, Kemper said.
There are several other areas where research has begun, including work on an integrated control room, cyber security, on-line monitoring, and alternatives to micro-processor technology, Kemper said.

Dyer said the agency will need additional technical staffers to review digital I&C applications. He said there are 13 such qualified staffers now, and that management estimates twice as many will be needed by the end of 20008.
But he said that management was planning its own "defense in depth" in case hiring that many employees proves too difficult. In that case, management would turn to contractors or DOE laboratories to find the staff, he said.
—Jenny Weil, Washington

Jaczko critiques NRC approach to reactor consequences study

A new NRC project to assess the consequences of reactor accidents should consider "the full spectrum of events that is reasonable to occur at a nuclear power plant," including scenarios with low probabilities that might be screened out by the project's current evaluation criteria, Commissioner Gregory Jaczko said in a speech last month.

Agency staff and contractor Sandia National Laboratories have begun a three-year "state of the art reactor consequences analysis," known as Soarca, to develop quantitative estimates of radiation doses and fatalities from a severe accident for each of the nation's operating power reactors. The Soarca analyses will update a 1982 Sandia report on reactor consequences which has been criticized by some, including Commissioner Edward McGaffigan, as being far too conservative in its assumptions.

"Because of the lack of clarity provided by the 1982 study, and because of the advances that we have made in our understanding of the behavior of radioactive materials in the intervening decades, the agency is now undertaking an update. The agency is, however, attempting to move away from addressing higher consequence events by arguing they are of such low probability that they are no longer worthy of consideration," Jaczko said in an October 19 speech at the quadripartite meeting of nuclear reactor safety advisory committees of France, Germany, Japan, and the US in Washington, DC. Held every four years, the meeting is not open to the public, but the NRC released Jaczko's speech October 31.

Jaczko said that the NRC staff's proposal for the Soarca project, "which the commission endorsed, involves only analyzing the consequences of events whose large early release frequency is 1x10-6 or greater." He said that he had "argued unsuccessfully that this was not the proper approach to updating the consequences analysis study," because "all this proposal does is to define a certain narrower range of events and analyze the consequences of that predefined and somewhat arbitrary frequency of occurrence."

"The only way to comprehensively address the consequences of accidents is to focus on those consequences regardless of the probability that they will occur. If we only focus on what is most likely to occur, we will always have doubts and gaps in our knowledge of events which *could* occur," Jaczko said.

As the NRC "has learned to work with risk tools and become comfortable with them, we have developed a tendency to overly rely on them. I am concerned that the staff and the commission have tended not to assess risk, but rather to use probability as a surrogate for risk. As we all know, risk equals probability times consequences, but we seem to want to focus on the probability and not the consequences," Jaczko said in his speech.

"Probability cannot be a surrogate for risk. We must get the consequences of low probability events out to the public in a properly conveyed context. If we do not, the public will always default to the 1982 study as the real consequences of an accident," Jaczko said.

In his February 9 vote sheet, released by NRC on November 2, Jaczko said that "based on the bounding limits proposed" in the staff's plan, "the consequences of events that are within the design basis of the plant, such as the large-break loss-of-coolant accident, would not be analyzed." Jaczko said he believes that "a driving force behind the effort" to update the 1982 Sandia report "is to preclude the misuse of information that exists in that report. The staff states in its plan that members of the public usually cite the extremely unlikely consequences for early fatalities and latent cancers in the 1982 study and that such an interpretation or application of this data is misleading."

But "the NRC cannot preclude individuals or groups from drawing inappropriate conclusions from technical information" and "should not tailor its analyses such that physically possible consequences are excluded," Jaczko said. "Should the staff proceed on this plan, the resultant study may lead to criticism that the commission revised the 1982 study to obtain more desirable results," he said.

Such criticism has already been leveled by Edwin Lyman, senior staff scientist at the Union of Concerned Scientists, who said at an October 25 public meeting on the project that NRC staff and industry might "try to shape results and outcome from the beginning," and that they might "screen out the highest consequence events at the beginning" of the project.

At the meeting, Christopher Hunter of NRC's Office of Nuclear Regulatory Research, or RES, said that Soarca's scenario selection will "focus on dominant accident scenarios," i.e., those which are determined to be at or above the threshold of an annual 1x10-6 release frequency. "Where possible," the analyses will "also consider high-consequence scenarios with somewhat lower frequencies," and the analyses will "lower the release frequency screening threshold by an order of magnitude for [containment] bypass scenarios." During this discussion, Schaperow said the staff "haven't decided yet" how they will proceed with the analysis if all accident scenarios assessed fall below the screening threshold.

Soarca secrecy challenged

The staff's plan for the Soarca project, Secy 05-233, was approved by the commission earlier this year. The Secy paper was not made public, nor were the commissioners' staff requirements memorandum and vote sheets (except for Jaczko's). Jason Schaperow of RES said at the meeting that the Secy and SRM had been withheld because they contained "sensitive information."

In a November 3 letter to NRC Chairman Dale Klein, Lyman requested that Secy 05-233 and its staff requirements memorandum be released to the public. "It is difficult to comprehend how the details of the project itself, ultimately including the results, can be made public, yet the commission's instructions to the staff on the project's guidelines are being kept secret. It appears that the commission has imposed significant constraints on the project, but does not wish to make those limits known to the public," Lyman said in his letter.

"Given that NRC staff and at least one commissioner have repeatedly asserted that the consequence results" of the 1982 Sandia study "are unrealistically severe, one can justifiably be concerned that the NRC has a predetermined outcome for the Soarca project," Lyman said. If the commission were to "give the appearance that it has skewed the study so that it will support this predetermined outcome," it "will badly damage the credibility of the Soarca project and render it useless as a means of increasing public confidence in nuclear power and the NRC's scientific integrity," Lyman said.

Jaczko said in his vote sheet that he favored making the Secy paper publicly available, though he supported "a

separate, secure document that describes the consequences of terrorism initiated scenarios." Lyman said in his letter that Jaczko's position "indicates that the claim that this document is too sensitive for public release is subject to dispute."

Lyman said in a November 9 e-mail that he had not yet received a reply to his letter to Klein.
—Steven Dolley, Washington

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538th Meeting of the
Advisory Committee on Reactor Safeguards
Draft Regulatory Guide DG-1145
December 7, 2006
T-2 B3, Rockville, MD

-PROPOSED AGENDA-

Cognizant Staff Engineer: David C. Fischer DCF@NRC.GOV (301) 415-6889

Topics		Presenters	Presentation Time
I	Opening Remarks	T. Kress, ACRS	8:35 am - 8:40 am
II	Staff Introductory Remarks	D. Matthews, NRO	8:40 am - 8:45 am
III	DG-1145 Overview - Purpose - Format and Structure - Developmental Basis - Status	E. Oesterle, NRO	8:45 am - 9:15 am
IV	PRA Requirements and Guidance	M. Rubin, NRR D. Harrison, NRR	9:15 am - 9:45 am
V	Industry Concerns/Public Comments	E. Oesterle, NRO Leslie Kass, NEI	9:45 am - 10:15 am
VI	Conformance, Completeness, and Consistency	E. Oesterle, NRO	10:15 am -10:20 pm
VII	Summary/Discussion	T. Kress/Members	10:20 am - 10:30 am
	BREAK		10:30 am - 10:45 am

NOTE:

- Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.
- 35 copies of the presentation materials to be provided.

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ACRS Full Committee Meeting

Presentation on DG-1145

December 7, 2006

Draft Regulatory DG-1145, "Combined License (COL) Applications for Nuclear Power Plants (LWR Edition)"

Eric R. Oesterle, Lead PM
NRO/DNRL/NGIF

DG-1145 Overview

Purpose

- Provide guidance to potential applicants on format and content for a combined license (COL) application pursuant to 10 CFR 52
- COL referencing neither a certified design (CD) nor an early site permit (ESP)
- COL referencing a CD but not an ESP
- COL referencing a CD and an ESP

DG-1145 Overview (cont'd)

Background and Developmental Basis

- Industry guidance for COL applications (NEI 04-01)
- NEI 04-01 provided guidance for "base case" COL application
- NRC interactions with external stakeholders identified several COL application scenarios
- Staff recognized the need for more comprehensive guidance for COL applicants

DG-1145 Overview (cont'd)

Development Basis

- RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)
- Updated SRP revisions (including draft 1996 updates)
- Draft NEI 04-01 guidance for COL applications
- NRC design certification and ESP experience
- SECY papers and associated SRMs

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DG-1145 Overview (cont'd)

Development Basis (cont'd)

- Proposed Part 52 rule issued on March 13, 2006 (71 FR 12782)
- Monthly public workshops (March 2006 – September 2006) ~ 500 comments
- All draft work-in-progress sections publicly available via NRC's website by June 30, 2006
- DG-1145 issued for 45-day public comment period on September 7, 2006 (71 FR 52826)

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DG-1145 Overview (cont'd)

Format and Structure

- Part C.I – guidance for a COL applicant that references neither a CD nor an ESP (consistent with proposed 10 CFR Part 52.79)
- Part C.II – additional technical information (consistent with proposed 10 CFR Part 52.80)
- Part C.III – COL applicants referencing CDs and ESPs
- Part C.IV – Miscellaneous Topics

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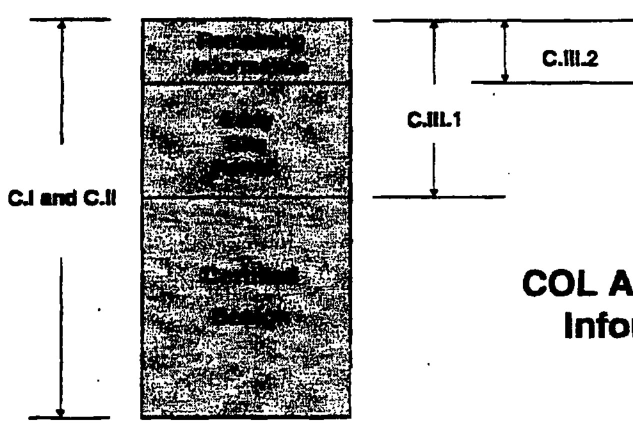
DG-1145 Overview (cont'd)



COL Application Information

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DG-1145 Overview (cont'd) Format and Structure – Part C.I

C.I.1 Introduction and General Plant Description*
C.I.2 Site Characteristics
C.I.3 Design of Structures, Systems, Components and Equipment
C.I.4 Reactor
C.I.5 RCS and Connected Systems
C.I.6 Engineered Safety Features
C.I.7 Instrumentation and Control
C.I.8 Electrical Power
C.I.9 Auxiliary Systems
C.I.10 Steam and Power Conversion System
C.I.11 Radioactive Waste Management
C.I.12 Radiation Protection
C.I.13 Conduct of Operations
C.I.14 Verification Programs
C.I.15 Transient and Accident Analyses
C.I.16 Technical Specifications
C.I.17 Quality Assurance and Reliability Assurance
C.I.18 Human Factors Engineering
C.I.19 Probabilistic Risk Assessment Information and Severe Accidents*

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DG-1145 Overview (cont'd)

Format and Structure – Part C.II

C.II.1 - Probabilistic Risk Assessment (PRA)
C.II.2 - Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)
C.II.3 - Environmental Report

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DG-1145 Overview (cont'd)

Format and Structure – Part C.III

C.III.1 - Information Needed for a COL Application Referencing a CD (consistent format with C.I)

C.III.2 - Information Needed for a COL Application Referencing a CD and an ESP (consistent format with C.I)

C.III.3 - Finality of an EIS Associated with an ESP

C.III.4 - COL Action Items

C.III.5 - Design Acceptance Criteria

C.III.6 - COL Application Timing

C.III.7 - ITAAC for COL Applications Referencing a CD and/or an ESP

DG-1145 Overview (cont'd)

Format and Structure – Part C.IV

C.IV.1 - COL Application Acceptance Review Checklist
C.IV.2 - Submittal Guidance for COLs
C.IV.3 - General Description of Change Process
C.IV.4 - Operational Programs
C.IV.5 - General and Financial Information
C.IV.6 - Limited Work Authorizations and Site Redress Plan*
C.IV.7 - Pre-Application Activities
C.IV.8 - Generic Issues
C.IV.9 - deleted
C.IV.10 - Regulatory Treatment of Non-Safety Systems (RTNSS)
C.IV.11 - relocated to App. I (responses to public workshop questions)
C.IV.12 - Applicability of Industry Guidance*

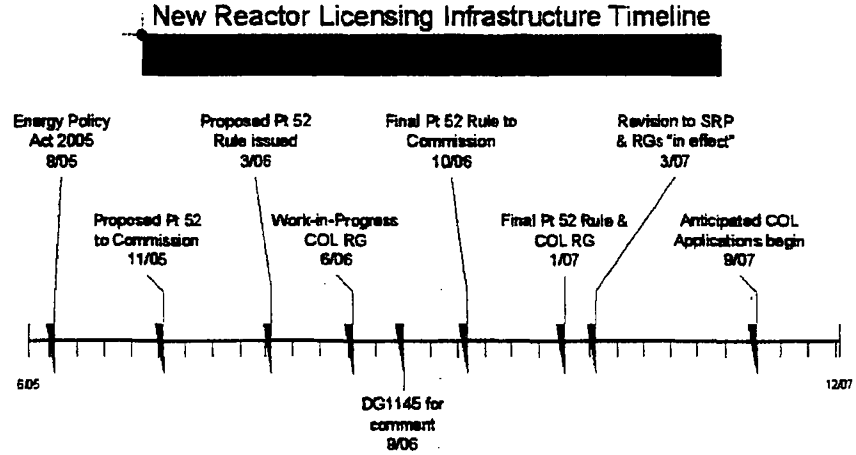
DG-1145 Overview (cont'd)

Status

- Comment period on DG-1145 closed on October 23, 2006
- Approximately 700 total comments received
- Staff is currently working to resolve public comments and revise DG-1145, as appropriate, and conform to proposed final Part 52 rule
- Process in place to ensure consistency between DG-1145 and the SRP and Reg. Guide updates
- Plan to publish final (RG 1.206) after incorporation of public comments and final issuance of the Part 52 rule
- Staff considering additional public forums to update external stakeholders on RG 1.206 prior to publication

DG-1145 Overview (cont'd)

New Reactor Licensing Infrastructure Timeline



DG-1145

PRA & Severe Accident Evaluations

ACRS Presentation

Donnie Harrison
Senior Reliability & Risk Analyst
NRR Division of Risk Assessment (DRA)

December 7, 2006

Discussion Topics

- Recent Change to Proposed 10 CFR Part 52
- Bases for Regulatory Guidance
- Objectives of PRA & Severe Accident Evaluations
- Chapter 19 Regulatory Guidance

2

Recent Change to Proposed 10 CFR Part 52

- Proposed 10 CFR Part 52 rulemaking included new 52.80(a) requirement for COL applicants to submit plant-specific PRA
- After completion of DG-1145, the NRC position changed to accept the industry comment to delete this requirement - PRA maintained available for staff inspection at the applicant's office
- Requirement deleted throughout Part 52, including the existing requirement for design certification applications

3

Impact of Change to Proposed 10 CFR Part 52

- DG-1145 will need to be revised to reflect the change in NRC position
  - Majority of guidance presented in C.II.1 (PRA) will need to be incorporated into C.I.19 (FSAR Chapter 19)
- Since FSAR Chapter 19 is a qualitative, summary description of the PRA, results, insights, uses, etc., staff audits will be necessary to fully understand, review, and confirm the bases for the PRA results and insights and adequacy for the PRA uses/applications

4

Bases for Regulatory Guidance

- NRC Policy Statements and SECYs/SRMs
- Experience with Design Certification reviews for CE System 80+, ABWR, AP-600, and AP-1000
- 10 CFR 52.79 PRA/Severe Accident Requirements

5

Objectives of PRA & Severe Accident Evaluations

- Derived from NRC Policy Statements and SECYs/SRMs
- Two Groups of Objectives
  - Identify and assess the balance of preventive and mitigative features (including operator actions) such that the plant design reflects a reduction in risk compared to existing plants(contemporary with Severe Accident Policy Statement of 1985)
  - Specific uses and applications of the PRA results and insights in support of other programs (e.g., RAP, RTNSS, ITAACs, COL and interface requirements)

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Chapter 19 Regulatory Guidance

19.1 Introduction

19.2 PRA Results and Insights

19.3 Severe Accident Evaluations

19.4 PRA Maintenance

19.5 PRA-Related ITAACs, COL Action Items, & Other Commitments

19.6 Conclusions

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DG-1145: Workshop Issues and Public Comments

Eric R. Oesterle, Lead PM
NRO/DNRL/NGIF

DG-1145: Workshop Issues and Public Comments

- Development began in January 2006
- Draft work-in-progress sections posted on the NRC's website following completion to facilitate public workshop discussions
- Monthly public workshops on DG-1145 held from March 2006 to September 2006
- Resolved and incorporated 500 public workshop comments to issue draft for public comment (Appendix I)

DG-1145: Workshop Issues and Public Comments

- Issued DG-1145 for public comment on September 7, 2006
- Comment period closed October 23, 2006
- Staff received approximately 700 public comments on DG-1145
- Comments also reflect external stakeholder concerns raised during the monthly public workshops on development of DG-1145

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DG-1145: Workshop Issues and Public Comments

- COL information availability
- Verification activities (inspections vs. ITAAC)
- First-of-a-kind-Engineering (FOAKE) inspections/audits
- Engineering design verification (EDV)

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DG-1145: Workshop Issues and Public Comments

- Plants incorporating passive safety design features
- Plant-specific PRA (LRF, CCFP, COL PRA Information)
- Maintenance Rule (breakout session)
- Digital I&C (breakout sessions)
- ITAAC

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DG-1145: Workshop Issues and Public Comments

- Environmental Reports and Finality of an EIS associated with an ESP
- Human factors engineering
- Radwaste treatment facilities
- Including guidance contained in responses to public workshop questions

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DG-1145
Conformance, Completeness, and Consistency

Eric R. Oesterle, Lead PM
NRO/DNRL/NGIF

DG-1145: Conformance, Completeness, and Consistency

- Staff is currently working to resolve public comments and revise DG-1145
- Conform DG-1145 to proposed final Part 52 rule and updated SRPs
- Internal staff review for consistency, completeness and usability began in early October 2006 (DG-1145 Reading Team)
- Team has confirmed approx. 1/3 of public comments and has made additional comments to those provided by public

DG-1145: Conformance, Completeness, and Consistency

- DG-1145 to be published final as Regulatory Guide 1.206
- RG 1.206 to be published following Commission approval of Part 52 rule and resolution of public comments
- Staff considering additional public forums to update external stakeholders on RG 1.206 prior to publication

(4)

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
REVIEW OF REGULATORY GUIDE 1.207 (DG-1144)
GUIDELINES FOR EVALUATING FATIGUE ANALYSES INCORPORATING THE LIFE REDUCTION OF METAL COMPONENTS DUE TO THE EFFECTS OF THE LIGHT-WATER REACTOR ENVIRONMENT FOR NEW REACTORS
December 7, 2006
ROCKVILLE, MARYLAND

-PROPOSED SCHEDULE-

Cognizant Staff Engineer: Charles G. Hammer, cgh@nrc.gov (301) 415-7363

	Topics	Presenters	Time
I.	**Opening Remarks**	S. Armijo, ACRS	10:45 - 10:50 am
II.	**Overview of Regulatory Guide 1.207 (DG-1144)**	H. Gonzalez, RES	10:50 - 11:20 am
III.	**Discussion of technical basis for RG 1.207 and NUREG/CR-6909**	H. Gonzalez, RES O. Chopra, Argonne National Laboratory	11:20 - 12:00 pm
IV.	**Committee Discussion**	S. Armijo, ACRS	12:00 - 12:15 pm

Note

- Presentation time should not exceed 50 percent of the total time allocated for specific items. The remaining 50 percent of the time is reserved for discussion.
- 35 copies of the presentation materials to be provided to the Committee.

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RG 1.207 - *GUIDELINES FOR EVALUATING FATIGUE ANALYSES INCORPORATING THE LIFE REDUCTION OF METAL COMPONENTS DUE TO THE EFFECTS OF THE LIGHT-WATER REACTOR ENVIRONMENT FOR NEW REACTORS*

Hipólito J. González
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Office of Nuclear Regulatory Research
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Omesh K. Chopra
Nuclear Engineering Division
Argonne National Laboratory
(630) 252-5117
okc@anl.gov

Presented to
Advisory Committee on Reactor Safeguards
Rockville, Maryland
December 7, 2006

Agenda

- Motivation
- Discuss RG 1.207
  - Objective and Implementation
  - Technical Basis
  - Regulatory Positions
- Resolution of public comments on DG-1144 and draft NUREG/CR-6909
- Conclusion

RG 1.207 User Need

- NRR User Need Request 2005-004 (January 7, 2005):
  - Develop guidance for determining the acceptable fatigue life of ASME pressure boundary components, with consideration of the LWR environment
  - For use in supporting reviews of applications that the agency expects to receive for new reactors.
  - Industry immediately notified
- High priority RG to be completed by March 2007

How RG 1.207 relates to the Regulatory Requirements

- General Design Criterion 1
  - Safety related SSC must be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function performed
- General Design Criterion 30
  - Components included in the reactor pressure boundary must be designed, fabricated, erected, and tested to the highest practical quality standards
- 10 CFR 50.55a (c), endorses ASME BPV Code for design of safety-related systems and components (Class 1)
  - ASME BPV Code Section III, includes fatigue design curves
- Fatigue design curves do not address the impact of the reactor coolant system environment

Objective and Implementation of RG 1.207

Objective

- To provide guidance for determining the acceptable fatigue life of ASME pressure boundary components, considering the LWR environment
  - Major structural materials: carbon steels, low-alloy steels, austenitic stainless steels, and Ni-Cr-Fe alloys (e.g., Alloy 600 and 690)
- Describes an approach that the NRC staff considers acceptable to support reviews of applications for new reactors

Implementation

- Applies to **New Plants**
- **No Backfitting is intended** (conservatism on current reactors)
- Regulatory guides are not substitutes for regulations, and compliance with regulatory guides is not required.

How the Technical Basis was Developed?

Technical Basis Report: NUREG/CR-6909 Rev. 1 – Effect of LWR Coolant Environment on Fatigue Life of Reactor Materials

Omesh K. Chopra
Nuclear Engineering Division
Argonne National Laboratory

Argonne

Issue - Environmental Effects on Fatigue Life

- Fatigue data indicate significant effects of LWR environment
- Data are consistent with each other & with much larger database for fatigue crack growth (da/dN)
  - in LWR environments, effects of material, loading, and environmental parameters are similar for fatigue ε-N & CGR data
- ε-N data have been evaluated to
  - identify key parameters that influence fatigue life, &
  - define range for these parameters where environmental effects are significant, i.e., establish threshold & saturation values
- If these conditions exist during reactor operation, environmental effects will be significant & must be addressed
  - subsection NB-3121 recognizes that the data used to develop the fatigue design curves did not include tests in environments that might accelerate fatigue failure

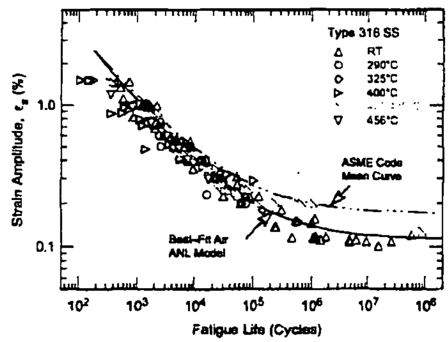
Fatigue Life

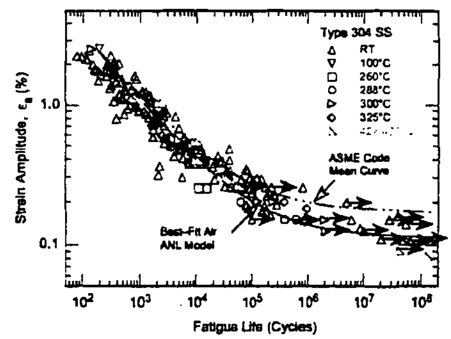
- Existing fatigue data define fatigue life of specimens as cycles to 25% load drop; typically this corresponds to a ≈3 mm crack
- Surface cracks ≈10 μm deep form early during fatigue loading
- Fatigue life associated with growth of cracks; 10 to 3000 μm
- Represented by two stages:
  Initiation: growth of cracks, < 300 μm
  Propagation: growth of cracks 300-3000 μm (EPFM)
- LWR coolant environment affects both stages:
  initiation & propagation

ASME Code Fatigue Design Curves

- Code design curves based on data obtained on small, smooth specimens in RT air under constant loading conditions
- To use small–specimen data for reactor components, best–fit curves must be adjusted to cover effects of variables that influence fatigue life but were not investigated in the data
  - such variables include mean stress, surface finish, size, & loading history. Data scatter & material variability must also be addressed
- To obtain Code design curves the best-fit curves were
  - first adjusted for effects of mean stress on fatigue life
  - then reduced by factor of 2 on stress & 20 on life to account for these variables, but not an aggressive environment

Current Code Mean Curve for Austenitic SSs





- Current Code mean curve is not consistent with existing fatigue data in air, at ε_a <0.3% it predicts significantly longer lives
- New design curves have been proposed that are consistent with the existing fatigue data

Environmental Effects on Carbon & Low-Alloy Steels

- The effects of critical parameters on fatigue life:
  - Steel type: effects identical for carbon & low-alloy steels
  - **Strain amp**: strain threshold near fatigue limit; no effect below threshold
  - Strain rate: logarithmic decrease in life below 1%/s, saturation at 0.001%/s; moderate effects above 1%/s
  - Temperature: linear decrease in life above 150°C; moderate effects below 150°C
  - Dissolved Oxygen: logarithmic decrease in life above 0.04 ppm, saturation at 0.5 ppm; moderate effects below 0.04 ppm
  - Sulfur: effects increase with increasing S level, saturation at 0.015 wt.%
  - Surface roughness: life of rough specimens is decreased in air; in high–DO water, surface roughness has little or no effect on fatigue life
  - Flow rate: in high–DO water, effects decrease with increasing flow rate

Environmental Effects on Austenitic Stainless Steels

- The effects of critical parameters on fatigue life:
  - Steel type: effects identical for wrought & cast austenitic stainless steels
  - **Strain amp**: threshold near fatigue limit; no effect below threshold
  - Strain rate: logarithmic decrease in life below 0.4%/s, saturation at 0.0004%/s; moderate effects above 0.4%/s
  - Temperature: linear decrease in life above 150°C; moderate effects below 150°C
  - Dissolved Oxygen: in high–DO, effect may be lower for some steels; in low–DO, effect significant for all steels & heat treat conditions;
  - Surface roughness: life of rough specimens decreased in air & low–DO water
  - Flow rate: no effect of flow rate on fatigue life in high-purity water

Carbon and Low-Alloy Steels

Air	$\ln[N] = 6.583 - 1.975 \ln(\varepsilon_a - 0.113)$	(CSs)	
	$\ln[N] = 6.449 - 1.808 \ln(\varepsilon_a - 0.151)$	(LASs)	
Env	$\ln[N] = 5.951 - 1.975 \ln(\varepsilon_a - 0.113) + 0.101\ S^*T^*O^*R^*$		(CSs)
	$\ln[N] = 5.747 - 1.808 \ln(\varepsilon_a - 0.151) + 0.101\ S^*T^*O^*R^*$		(LASs)

where	$S^* = S$	($S \leq 0.015$ wt.%)
	$S^* = 0.015$	($S > 0.015$ wt.%)
	$T^* = 0$	($T < 150°C$)
	$T^* = T - 150$	($T = 150$ to $320°C$)
	$O^* = 0$	($DO < 0.04$ ppm)
	$O^* = \ln(DO/0.04)$	(0.04 ppm $< DO \leq 0.5$ ppm)
	$O^* = \ln(12.5)$	($DO > 0.5$ ppm
	$R^* = 0$	($R > 1\%/s$)
	$R^* = \ln(R)$	($0.001 \leq R \leq 1\%/s$)
	$R^* = \ln(0.001)$	($R < 0.001\%/s$)

- These expressions represent average fatigue life of the median material

Wrought & Cast Austenitic SSs

Air $\ln[N] = 6.891 - 1.920 \ln(\varepsilon_a - 0.112)$

Env $\ln[N] = 6.157 - 1.920 \ln(\varepsilon_a - 0.112) + T^*O^*R^*$

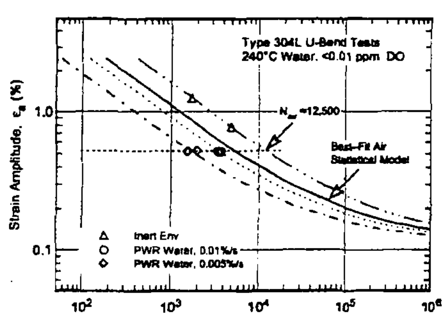
where

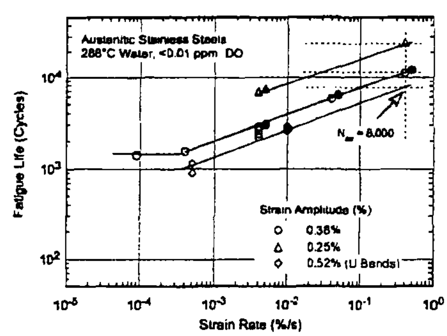
$T^* = 0$	$(T < 150°C)$
$T^* = (T - 150)/175$	$(150 \leq T < 325°C)$
$T^* = 1$	$(T \geq 325°C)$
$O^* = 0.281$	(all DO levels)
$R^* = 0$	$(R > 0.4\%/s)$
$R^* = \ln(R/0.4)$	$(0.0004 \leq R \leq 0.4\%/s)$
$R^* = \ln(0.0004/0.4)$	$(R < 0.0004\%/s)$

- These expressions represent average fatigue life of the median material

Stainless Steel Tube U-Bend Tests in PWR Water at 240°C

- Applicability of laboratory data to component behavior has been demonstrated by several component tests





- Measured environmental reduction factor
  F_{en} =10,000/1,728 =5.8 at 0.0005%/s & =10,000/3,624 = 2.8 at 0.01%/s.
  Predicted values are 5.5 and 3.6, respectively

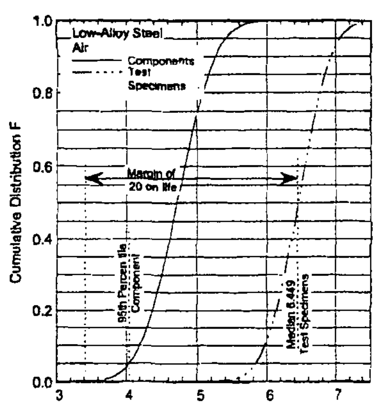
Fatigue Life of Components

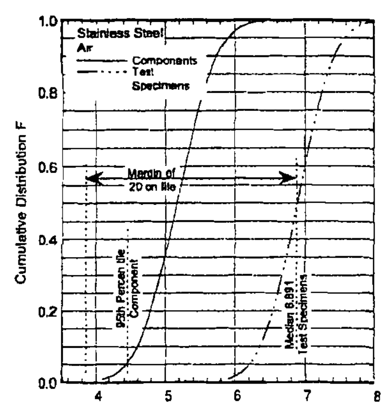
- Available information reviewed to better define adjustment factor on life that must be applied to mean-data curve to account for effects of variables that influence life but were not explicitly addressed in the data

Parameter	ASME Code	Presented Study
Material Variability & Data Scatter	2.0	2.1 - 2.8
Size	2.5	1.2 - 1.4
Surface Finish	4.0	2.0 - 3.5
Loading History	–	1.2 - 2.0
Total Adjustment Factor	**20**	**6 - 27**

- Monte Carlo simulations performed to determine distribution of A for adjusted fatigue curve that represents behavior of actual component.
- Use material variability & data scatter results from present analysis
- Assume a lognormal distribution for effects of size, surface finish, & loading history, & min and max values of adjustment factor assumed to represent 5th and 95th percentile, respectively
- Assume effects can be considered as independent based on engineering judgment

Fatigue Design Adjustment Factors





- Monte Carlo analysis suggests adjustment applied to mean values of specimen fatigue life to bound component fatigue life of 95% of population is ≈12. Thus, current Code requirements of factor of 20 on life is conservative by about a factor of ≈1.7 for components

Methods for Incorporating Environmental Effects

- Two approaches proposed for incorporating effects of LWR coolant environments into Code fatigue evaluations:
  - develop new fatigue design curves for LWR environments
  - use an environmental fatigue correction factor F_{en}
- Because fatigue life in LWR environments depends on several loading & environmental parameters, design curve approach would require developing multiple design curves to cover range of conditions or a conservative bounding curve
- The F_{en} approach is relatively simple and flexible enough to address effects without unnecessary conservatism

F_{en} Method for Incorporating Environmental Effects

- F_{en} is defined as ratio of fatigue life in air at RT to that in water under service conditions

$$\ln[F_{en}] = \ln(N_{RTair}) - \ln(N_{water})$$

$$F_{en} = \exp(0.632 - 0.101\ S^*T^*O^*R^*) \quad \text{(Carbon Steels)}$$

$$F_{en} = \exp(0.702 - 0.101\ S^*T^*O^*R^*) \quad \text{(Low-Alloy Steels)}$$

$$F_{en} = \exp(0.734 - T^*O^*R^*) \quad \text{(Stainless Steels)}$$

$$F_{en} = 1 \quad (\varepsilon_a \leq 0.07\%\ \text{CLAS}\ \&\ \leq 0.10\%\ \text{SSs})$$

- To incorporate environmental effects, fatigue usage based on air curve is multiplied by F_{en}

$$U_{en} = U_1 F_{en,1} + U_2 F_{en,2} \ldots\ U_n F_{en,n}$$

Fen Method (Contd.)

- For CSs & LASs, current Code design curves are either consistent or conservative with respect to existing data
  - usage factors can be based on current Code design curves, or
  - to reduce conservatism, use design curves based on ANL models and adjustment factors of 2 & 12
- For austenitic SSs & Ni-Cr-Fe alloys, current Code design curve for SSs is nonconservative with respect to existing data
  - usage factors should be determined from the new design curves based on ANL model and adjustment factors of 2 & 12
  - current Code design curve should not be used because it will yield nonconservative estimates of CUF

Regulatory Positions

Regulatory Position 1: Carbon and Low-Alloy Steels

✓ Calculate fatigue usage in air with ASME Code Analysis procedures +
  - ✓ ASME Code air curves, **or**
  - ✓ New ANL model air curves

✓ Calculate the F_{en} using
  - ✓ Equation A.2 (CS),

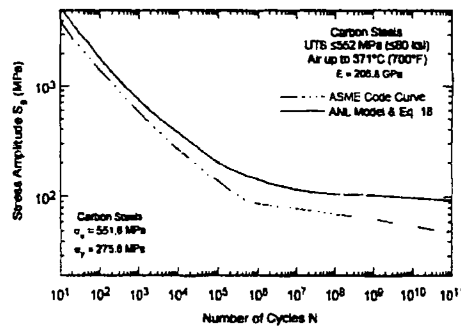
    $F_{en} = \exp(0.632 - 0.101\ S^*T^*O^*\dot{\varepsilon}^*)$
  - ✓ Equation A.3 (LAS)

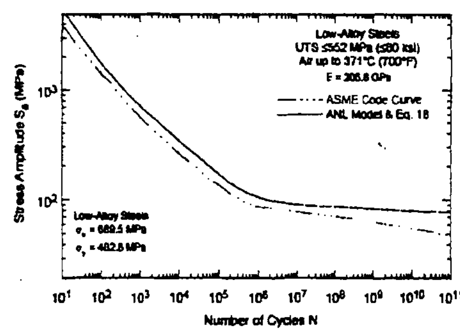
    $F_{en} = \exp(0.702 - 0.101\ S^*T^*O^*\dot{\varepsilon}^*)$

    (Appendix A of NUREG/CR-6909)

✓ Calculate the environmental fatigue usage (U_{en})

$U_{en} = U_1\ F_{en,1} + U_2\ F_{en,2} \ldots\ U_n\ F_{en,n}$





Regulatory Position 2: Austenitic Stainless Steels

✓ Calculate fatigue usage in air with ASME Code Analysis procedures +
  - ✓ **New ANL model air SS curve**

✓ Calculate the F_{en} using
  - ✓ Equation A.9

    $F_{en} = \exp(0.702 - 0.101\ S^*T^*O^*\dot{\varepsilon}^*)$

    (Appendix A of NUREG/CR-6909)

✓ Calculate the environmental fatigue usage (U_{en})

$U_{en} = U_1\ F_{en,1} + U_2\ F_{en,2} \ldots\ U_n\ F_{en,n}$