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

June 15, 2006

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Diaz:

SUBJECT: SUMMARY REPORT - 532nd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, MAY 4-5, 2006, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 532nd meeting, May 4-5, 2006, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports, letter, and memoranda:

REPORTS:

Reports to Nils J. Diaz, Chairman, NRC, from Graham B. Wallis, Chairman, ACRS:

- Report on the Safety Aspects of the License Renewal Application for the Brunswick Steam Electric Plant, Units 1 and 2, dated May 17, 2006
- Beaver Valley Extended Power Uprate Application, dated May 22, 2006
- Proposed Revisions to 10 CFR Part 52: Licenses, Certifications, and Approvals for Nuclear Power Plants, and Conforming Amendments to Applicable NRC Regulations, dated May 22, 2006
- R. E. Ginna Extended Power Uprate Application, dated May 22, 2006

LETTER:

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

- Modified Draft Final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," dated May 17, 2006

MEMORANDA:

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

- 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," dated May 5, 2006
- Clinton Early Site Permit Application - Final Safety Evaluation Report Changed Pages Prior to Publishing as a NUREG, dated May 8, 2006

HIGHLIGHTS OF KEY ISSUES

1. Final Review of the License Renewal Application for the Brunswick Steam Electric Plant

The Committee met with representatives of the NRC staff and the Carolina Power and Light (CP&L) Company to discuss the license renewal application for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2 and the associated final Safety Evaluation Report (SER). CP&L requested approval for continued operation of each unit for 20 years beyond the current license expiration dates. The operating licenses for Units 1 and 2 expire on September 8, 2016, and December 27, 2014, respectively. Each unit is a General Electric BWR 4 with a unique Mark I containment. The containment is constructed of reinforced concrete with a steel liner. CP&L described operating experience with the drywell liners; operating experience with vibration from extended power uprates; major equipment replacements and repairs; major exceptions to the Generic Aging Lessons Learned Report; and the commitment tracking system. The draft SER was issued on December 20, 2005, with no open or confirmatory items. As a result of the staff's review, several components were brought into scope of license renewal. The staff described a new two-tiered process for reviewing the scoping of balance of plant systems. This application was the first to be reviewed using this new process. The final SER issued on March 31, 2006, concluded that the requirements of 10 CFR 54.29(a) have been met.

Committee Action

The Committee issued a report to the NRC Chairman, dated May 17, 2006, concluding that the programs committed to and established by the applicant to manage age-related degradation provide reasonable assurance that BSEP Units 1 and 2 can be operated in accordance with their current licensing basis for the period of extended operation with no undue risk to the health and safety of the public. The Committee recommended that the application for renewal of the operating licenses for BSEP, Units 1 and 2 be approved. The Committee also concluded that the staff's new two-tiered process for reviewing the scoping of balance of plant systems was effective and recommended that this process be used in the review of future license renewal applications.

2. Final Review of the Extended Power Uprate Application for R. E. Ginna Nuclear Plant

The Committee reviewed the application by Constellation Energy for an increase of approximately 17 percent power level for the R.E. Ginna Nuclear Power Plant (Ginna). The committee considered the revised safety evaluation results, system impacts, component vibration, flow-accelerated corrosion, power ascension and testing, and the risk aspects of this application. The Committee noted that the licensee had undertaken an evaluation of plant changes that could be made at the time of the power uprate that would result in an overall decrease in core damage frequency (CDF). The licensee has committed to undertaking a set of modifications that will have a net impact on CDF and large early release frequency (LERF) such that after the EPU, the CDF and LERF will be slightly less than the pre-EPU values.

Committee Action

The Committee issued a report to the NRC Chairman, dated May 22, 2006, recommending that the application for a power uprate at Ginna be approved.

3. Final Review of the Extended Power Uprate Application for the Beaver Valley Nuclear Plant

The Committee reviewed the application by FirstEnergy Nuclear Operating Company for an increase of approximately 8 percent power level for the Beaver Valley Power Station, Units 1 and 2. The committee considered the revised safety evaluation results, the containment analyses, reactor vessel integrity, component vibration, flow-accelerated corrosion, power ascension and testing, and the risk aspects of this application. It heard presentations by the staff concerning boron concentration following a loss-of-coolant accident, and noted that the staff performed a number of independent calculations to verify the analytical results reported by the licensee for this event, as well as several other operational transients and accidents.

Committee Action

The Committee issued a report to the NRC Chairman, dated May 22, 2006, recommending that the application for a power uprate at Beaver Valley be approved.

4. Proposed Revisions to 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants"

The Committee met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss proposed revisions to 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." The NRC staff characterized the proposed changes to Part 52 and highlighted several proposals that could affect safety requirements [e.g., emergency preparedness requirements at the early site permit (ESP) and combined license (COL) stages, quality assurance requirements for ESP applicants, reporting requirements for ESPs and design certifications, and probabilistic risk assessment (PRA) requirements for COLs]. NEI highlighted several industry concerns with the proposed rule (e.g., extensive rule changes being made on the verge of COL applications, the potential for level 3 PRA requirements/guidance for COL applicants, reporting requirements for ESPs) and identified several areas where industry thought the rule could be improved (e.g., to include a change process for severe accident mitigation features of certified reactor designs, to include provisions for limited work authorizations).

Committee Action

The Committee issued a report to the NRC Chairman, dated May 22, 2006, recommending that a level 3 PRA consequence analysis not be required at the ESP stage, that COL holders be required to keep their PRAs up to date but not require that they be submitted to the NRC, that it should be sufficient for the ESP applicant to identify only the "major features" of the site emergency plan, that the definition of major features be specified in regulatory guidance documents, and that operation up to 5% power be permitted with FEMA-identified deficiencies in a COL holder's emergency plan (as is currently allowed for power plants licensed under Part 50).

5. NRC Staff's Response to ACRS Comments on the Draft Final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants"

The Committee met with representatives of the NRC staff to discuss their proposed resolution of ACRS comments on Revision 4 to Regulatory Guide 1.97, which the Committee had reviewed during the 530th meeting. The staff discussed the specific regulatory position of concern, the Committee's comments on the regulatory position, and the proposed modifications to the regulatory position. The proposal removes the previous guidance regarding partial conversions of accident monitoring instrumentation and modifies Regulatory Position 1 to provide additional guidance to current operating reactor licensees with regard to performing modifications to accident monitoring instrumentation. The Committee also heard statements from two members of the public supporting the modifications to the Regulatory Guide.

Committee Action:

The Committee issued a letter to the EDO, dated May 17, 2006, recommending that the staff issue the Regulatory Guide 1.97, Revision 4, as final.

6. Subcommittee Report on Reliability and Probabilistic Risk Assessment

The Subcommittee discussed the probabilistic risk assessment (PRA) for the Economic Simplified Boiling Water Reactor (ESBWR), an advanced design from General Electric (GE) that is in the process of being certified by the NRC. The subcommittee identified several issues for further examination.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's response of April 20, 2006, to comments and recommendations included in the March 28, 2006 ACRS letter on the Draft Final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants."

  The Committee decided that it was satisfied with the EDO's response.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from April 6, 2006, through May 3, 2006, the following Subcommittee meetings were held:

- Reliability and Probabilistic Risk Assessment - April 20-21, 2006

The Subcommittee reviewed the PRA for General Electric's next generation simplified boiling water reactor, the ESBWR.

- Power Uprates - April 25-27, 2006

The Subcommittee reviewed the application by FirstEnergy for an 8% power uprate for Beaver Valley Power Station, Units 1 and 2. The Subcommittee also reviewed the small-break LOCA portion of the staff's evaluation related to the Ginna Extended Power Uprate.

- Reliability and Probabilistic Risk Assessment - April 28, 2006

The Subcommittee on Reliability and Probabilistic Risk Assessment was briefed by the NRC staff, Nuclear Energy Institute (NEI), Electric Power Research Institute (EPRI), and several pilot plant licensees on Risk Management Technical Specifications Initiative 4b, "Risk-Informed Completion Times."

- Planning and Procedures - May 3, 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 two-tiered process used by the staff in reviewing the scoping of balance-of-plant systems at the Brunswick Nuclear Plant should be used in reviewing future license renewal applications.

- The Committee would like an opportunity to review the draft final version of 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," after reconciliation of public comments.

- The Committee looks forward to reviewing the progress made by the staff and/or the industry with regard to a more detailed treatment of the thermal-hydraulic conditions within the core region to better define the conditions leading to recirculation and mixing within the vessel and lower plenum.

PROPOSED SCHEDULE FOR THE 533rd ACRS MEETING

The Committee agreed to consider the following topics during the 533rd ACRS meeting, to be held on May 31, 2006, through June 1, 2006:

- Draft Final Generic Letter, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations"
- Draft Final Generic Letter 2006-xx, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems"
- Interim Staff Guidance on Aging Management Program for Inaccessible Areas of Boiling Water Reactor (BWR) Mark I Containment Drywell Shell
- Overview of New Reactor Licensing Activities
- Status Report on the Quality Assessment of Selected NRC Research Projects

Sincerely,

Graham B. Wallis

Graham B. Wallis
ACRS Chairman

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

June 15, 2006

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

FROM: Graham B. Wallis
ACRS Chairman

Graham B. Wallis

SUBJECT: CERTIFIED MINUTES OF THE 532nd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), MAY 4-5, 2006

I certify that based on my review of the minutes from the 532nd 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.

CERTIFIED

Date Issued: 6/7/2006
Date Certified: 6/15/2006

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MAY 4-5, 2006

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APPENDICES

MINUTES OF THE 532nd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MAY 4-5, 2006
ROCKVILLE, MARYLAND

The 532nd meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on May 4-5, 2006. Notice of this meeting was published in the *Federal Register* on April 18, 2006 (65 FR 19910) (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. There were no written statements or requests for time to make oral statements from members of the public regarding the meeting.

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: ACRS Members: Dr. Graham B. Wallis (Chairman), Dr. William J. Shack (Vice Chairman), Mr. John D. Sieber, (Member-at-Large), Dr. George E. Apostolakis, Dr. J. Sam Armijo, Dr. Mario V. Bonaca, Dr. Richard S. Denning, 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. and reviewed the schedule for the meeting. He summarized the agenda topics for this meeting and discussed the administrative items for consideration by the full Committee.

II. Final Review of the License Renewal Application for the Brunswick Steam Electric Plant (Open)

[Note: Mr. Michael Junge was the cognizant Staff Engineer and Mr. Cayetano Santos was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff and the Carolina Power and Light (CP&L) Company to discuss the license renewal application for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2 and the associated final Safety Evaluation Report (SER). CP&L requested approval for continued operation of each unit for 20 years beyond the current license expiration dates. The operating licenses for Units 1 and 2 expire on September 8, 2016, and December 27, 2014, respectively. Each unit is a General Electric BWR 4 with a unique Mark I containment. The containment is constructed of reinforced concrete with a steel liner. CP&L described operating experience with the drywell liners; operating experience with vibration from extended power uprates; major equipment replacements and repairs; major exceptions to the Generic Aging Lessons Learned Report; and the commitment tracking system. The draft SER was issued on December 20, 2005, with no open or confirmatory items. As a result of the staff's review, several components were brought into scope of license renewal. The staff described a new two-tiered process for reviewing the scoping of balance of plant systems. This application was the first to be reviewed using this new process. The final SER issued on March 31, 2006, concluded that the requirements of 10 CFR 54.29(a) have been met.

Committee Action

The Committee issued a report to the NRC Chairman, dated May 17, 2006, concluding that the programs committed to and established by the applicant to manage age-related degradation provide reasonable assurance that BSEP Units 1 and 2 can be operated in accordance with their current licensing basis for the period of extended operation with no undue risk to the health and safety of the public. The Committee recommended that the application for renewal of the operating licenses for BSEP, Units 1 and 2 be approved. The Committee also concluded that the staff's new two-tiered process for reviewing the scoping of balance of plant systems was effective and recommended that this process be used in the review of future license renewal applications.

III. Final Review of the Extended Power Uprate Application for R. E. Ginna Nuclear Plant (Open)

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

The Committee reviewed the application by Constellation Energy for an increase of approximately 17 percent power level for the R.E. Ginna Nuclear Power Plant (Ginna). The Committee considered the revised safety evaluation results, system impacts, component vibration, flow-accelerated corrosion, power ascension and testing, and the risk aspects of this application. The Committee noted that the licensee had undertaken an evaluation of plant changes that could be made at the time of the power uprate that would result in an overall decrease in core damage frequency (CDF). The licensee has committed to undertaking a set of modifications that will have a net impact on CDF and large early release frequency (LERF) such that after the EPU, the CDF and LERF will be slightly less than the pre-EPU values.

Committee Action

The Committee issued a report to the NRC Chairman, dated May 22, 2006, recommending that the application for a power uprate at Ginna be approved.

IV. Final Review of the Extended Power Uprate Application for the Beaver Valley Nuclear Plant (Open)

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

The Committee reviewed the application by FirstEnergy Nuclear Operating Company for an increase of approximately 8 percent power level for the Beaver Valley Power Station, Units 1 and 2. The Committee considered the revised safety evaluation results, the containment analyses, reactor vessel integrity, component vibration, flow-accelerated corrosion, power ascension and testing, and the risk aspects of this application. It heard presentations by the staff concerning boron concentration following a loss-of-coolant accident, and noted that the staff performed a number of independent calculations to verify the analytical results reported by the licensee for this event, as well as several other operational transients and accidents.

Committee Action

The Committee issued a report to the NRC Chairman, dated May 22, 2006, recommending that the application for a power uprate at Beaver Valley be approved.

V. Proposed Revisions to 10 CFR Part 52, "License, Certifications, and Approvals for Nuclear Power Plants" (Open)

[Note: Mr. David Fischer was the cognizant Staff Engineer and Mr. Michael Snodderly 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 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (71 FR 12782 to 12932, dated March 13, 2006).

Dr. Kress opened the session with a very brief description of the proposed changes to Part 52 and associated regulations. Ms. Eileen McKenna, of the NRC staff, was the team lead for getting the revised Part 52 rulemaking completed. Part 52 establishes the framework under which many of the new reactor applications will be submitted and processed. The staff is not specifically seeking a letter from the ACRS on the proposed rule but the public comment period ends in May and the Commission requested the proposed final rule by October 2006. The staff does not anticipate coming back to the Committee prior to issuing the final rule.

Presentation by the NRC Staff

Ms. Nanette Gilles of NRR's Division of New Reactor Licensing introduced herself. She is one of the co-authors of the proposed revision to Part 52, along with Jerry Wilson, and Geary Mizuno, in the Office of the General Counsel. Ms. Gilles said that the purpose of the briefing was to familiarize the Committee with the key objectives of the rulemaking and to provide a general overview of the changes to Part 52, as well as other parts of 10 CFR, with a focus on the changes that are related to safety requirements. Ms. Gilles stated that the proposed rule published in the *Federal Register* on Mach 13, 2006 (71 FR 12781), supercedes a previously published proposed revision to Part 52 (68 FR 40026 dated July 3, 2003). The current proposed revision is based on public comments received on the previously published proposed revision to Part 52 as well as on lessons learned from the staff's review of the first three early site permit (ESP) applications, review of the AP1000 design certification, and meetings with industry on the combined license (COL) process. The rewritten Part 52 contains five subparts: 1) ESPs, 2) standard design certifications, 3) COLs, 4) standard design approvals, and 5) manufacturing licenses. She gave a very brief description of each process. A standard design approval is the same as a standard design certification except without the certification rulemaking, hearing, or Commission review. The Committee discussed with the staff the pros and cons of a design certification (e.g., more finality with regard to the design) versus a design approval (e.g., staff technical review is completed, the applicant could start construction sooner, but the design could be challenged). Ms. Gilles also stated that ESPs are applicable for 20 years. The only appendices that remain in the revised Part 52 are one for each certified design. (i.e., General Electric advanced boiling water reactor, Combustion Engineering System 80 Plus, Westinghouse AP600, and Westinghouse AP1000).

Ms. Gilles said that there were two actions that accounted for a vast majority of the changes in the proposed rule. First, the organization and content of each of the five subparts was standardized. Second, conforming changes throughout the rest of 10 CFR were made. The staff tried to make sure all of the other technical and procedural requirements recognized that the licensing process in Part 52 existed and tried to be explicit as to which requirements applied to each of the five processes. Generally, the proposed rule keeps technical requirements in Part 50, Part 100, etc. and keeps the procedural requirements in Part 52. There was a concerted effort not to change the technical requirements in other parts, unless the change was necessitated by the virtue of the structure of the Part 52 licensing process as compared to the old construction permit / operating license process. The revised rule will enhance the NRC's effectiveness and efficiency in implementing the Part 52 licensing process and it will provide clarity regarding the applicability of technical and procedural requirements to each of the Part 52 regulatory processes.

Ms Gilles highlighted several aspects of the proposed rule that could affect safety requirements [e.g., emergency preparedness requirements at the early site permit (ESP) and combined license (COL) stages, quality assurance requirements for ESP applicants, reporting requirements for ESPs and design certifications, probabilistic risk assessment (PRA) requirements for COLs]. Three proposed changes to the emergency preparedness requirements stemmed from the lessons learned during the early site permit reviews. There is a provision in the current early site permit subpart that requires applicants to identify physical characteristics unique to the proposed site that could pose a "significant impediment" to the development of emergency plans. The first proposed change would add a requirement that if such a physical characteristic is identified, the applicant would also be required to identify measures which would, when implemented, mitigate that impediment to the development of emergency plans. The proposed mitigation measures would most likely show up as a permit condition in the early site permit. Dr. Powers questioned the need for this requirement. He stated that having applicants identify (or not identify) mitigative measures was not a problem during the review of the first three early site permit applications because none of the proposed sites had a "significant impediment." Ms. Gilles explained that at the ESP application stage, in addition to identifying significant impediments, applicants may also choose to either provide the "major features" of the emergency plans or provide complete and integrated emergency plans. The degree of finality on the emergency preparedness issue would depend on the level of detail provided by the applicant. Under either of these latter two options, the proposed rule would now require that the applicant also submit proposed inspections, tests, and analyses acceptance criteria (ITAAC) that the holder of the combined license referencing the ESP shall perform and/or meet. At the COL stage, an applicant has only one option, and that is to provide complete emergency plans. Dr. Powers indicated that, base on the review of the first three early site permit applications, better guidance is needed on what constitutes a "major feature." In his opinion, significant impediments would constitute a major feature. Dr. Powers questioned whether there would be sufficient emergency preparedness information available at the ESP stage to reach the level of finality sought by the industry and staff. The third proposed rule change, related to emergency preparedness, is that COL applicants that reference an ESP will be required to update the emergency preparedness information with any new information.

Ms. Gilles stated that the proposed rule now contains an explicit requirement that the Appendix B quality assurance requirements apply to ESP applicants (e.g., in collecting soil boring data). The proposed rule also clarifies the applicability of 10 CFR Part 21 and 10 CFR 50.55(e) to entities that hold a permit or a license under Part 52. Finally, Ms. Gilles discussed the PRA requirements in Part 52. Currently, the rule contains a requirement that design certification and COL applicants submit a PRA with their application. At the Commission's direction, the staff asked whether the Commission should adopt in the final rule a new provision that would require COL holders to update their PRA and submit it to the NRC periodically throughout the life of the facility (e.g., on a schedule either similar to that for FSAR updates or perhaps every other refueling outage). Dr. Apostolakis asked whether the kind of PRA was at issue, as well as the just having an up-to-date PRA. Ms. Gilles explained that in the proposed rule sent to the Commission, there was an attempt to address the kind of PRA that should be required by the rule. She said that the Commission directed the staff to take that language out of the rule and to address those issues in the regulatory guidance associated with Part 52. The Committee discussed with the staff the pros and cons of having the COL holder update its PRA, and having them submit the updated PRA to the NRC. The Committee also discussed with the staff what type of PRA should be required for COL applicants by the rule. The staff suggested that because standards are not yet available for all mode (e.g., shutdown and low power operation) or for most external events perhaps the rule should not be prescriptive at this point. Rather, the staff plans to provide guidance to applicants on PRA scope, as directed by the

Commission.

Presentation by Industry

Mr. Russell Bell from the Nuclear Energy Institute (NEI) highlighted several industry concerns with the proposed rule (e.g., extensive rule changes being made on the verge of COL applications, the potential for level 3 PRA requirement/guidance for COL applicants, reporting requirements for ESPs) and identified several areas where industry thought the rule could be improved (e.g.; to include a change process for severe accident mitigation features, or other features, of certified reactor designs; to include provisions for limited work authorizations). He said that industry's single biggest concern with the staff's proposed rule was the addition of PRA scope requirements, and industry was pleased to see those requirements deleted from the rule by the Commission. Dr. Apostolakis asked when the COL regulatory guide, containing the PRA scope guidance, would be issued. Mr. Beckner, Deputy Director of NRR's Division of New Reactor Licensing, said that a draft would be out in June of 2006. Mr. Bell indicated that industry would also object to having the PRA scope guidance in the COL regulatory guide, absent having clear agreed-upon standards for doing a full-scope PRA. When asked about periodically updating the PRA, Mr. Bell indicated that licensees are currently periodically updating their PRAs, so he said such a requirement would not be an imposition or an issue. However, he questioned the rationale for requiring that the PRA and the periodic updates to the PRA be submitted to the staff. Dr. Kress asked Mr. Bell his view on requiring a radiological consequence at the ESP stage. Mr. Bell said he did not like it because ESP applicants may not know the details of the plant design they will put on the site (e.g., source term, mitigation systems, etc.). Mr. Bell also voiced an objection to a requirement in the proposed rule that ESP holders periodically update their emergency planning information because, he said, nobody may reference the ESP. Dr. Kress asked for Mr. Bell's opinion on the provision in the proposed rule that would allow COL holders to operate their plant up to 5% power even though there might be an impediment to emergency planning brought forth by FEMA. Mr. Bell said that this is the current practice, i.e., for plants licensed under Part 50, was mutually agreed upon by FEMA and the NRC. Mr. Bell said that operation up to 5% with an emergency preparedness issue is a practical issue for the COL holder and not a safety or emergency planning concern. Mr. Maynard noted that there is a four- to six-month period during which new plants conduct low-power testing. Mr. Bell also voiced an objection to the proposed reporting requirements for ESP applicants, design certification applicants, and ESP holders. He also expressed concern over the proposed requirement that applicants address international operating experience (i.e., how it will be done and whether it is necessary). He questioned whether environmental reviews, completed at the ESP stage, necessarily needed to be re-done at the COL application stage. Dr. Kress asked Mr. Bell's opinion of the provision in the proposed rule which would restrict the use of ESPs and design certifications to COL applicant under Part 52 (i.e., licensees that applied for a construction permit under Part 50 would not be permitted to reference an ESP or design certification). Mr. Bell said that industry liked flexibility and that they did not want to rule out any particular licensing scenario.

Committee Action

The Committee issued a report to the NRC Chairman dated May 22, 2006, recommending that a level 3 PRA consequence analysis not be required at the ESP stage, that COL holders be required to keep their PRAs up to date but not require that they be submitted to the NRC, that it should be sufficient for the ESP applicant to identify only the "major features" of the site emergency plan, that the definition of major features be specified in regulatory guidance documents, and that operation up to 5% power be permitted with FEMA-identified deficiencies in a COL holder's emergency plan (as is currently allowed for power plants licensed under Part 50).

VI. NRC Staff's Response to ACRS Comments on the Draft Final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation of Nuclear Power Plants" (Open)

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

Mr. John Sieber, the cognizant Committee member for this issue, introduced the topic. Mr. Sieber provided an overview of the topic. He reminded the Members that they reviewed a previous draft of Revision 4 of Regulatory Guide 1.97 during the 530th meeting in March 2006, and that the Committee provided three recommendations regarding the regulatory guide. The Regulatory Guide endorses IEEE Std 497-2002, with exceptions. The Committee recommended the staff revise Regulatory Position 1 to allow licensees to adopt the proposed standard to modify individual accident monitoring instruments without a complete analysis of all accident monitoring instrumentation. The Committee agreed that licensees should not be allowed to partially use the new standard to eliminate or reclassify accident monitoring instrumentation required by earlier standards unless Revision 4 of the Regulatory Guide is adopted in its entirety. He noted that two members of the public requested an opportunity to address the Committee. Mr. Sieber then asked Mr. George Tartal of the Office of Nuclear Regulatory Research to begin the staff presentation to address the Committee's concerns.

NRC Staff Presentation

Mr. Tartal first outlined his presentation, then reviewed the staff's interpretation of the Committee's comments from the March meeting. He stated that the staff concluded that the ACRS agreed with Regulatory Position 4, but had concerns with Regulatory Position 1 of the previous draft of the guide. Mr. Tartal then reviewed the previous text of Regulatory Position 1 and reviewed the intent of Revision 4 of the guide to apply to new nuclear power plants. He noted that current reactors have the option to voluntarily convert from Revision 3 to Revision 4, which involves adapting the plant's entire accident monitoring program. Such a conversion could involve physical modifications and licensing basis changes, which could result in significant cost implications. The staff had recommended against partial conversions due to the potential loss of accident monitoring variables or interactions among variables if a complete analysis was not performed.

Mr. Tartal then reviewed the conclusion and recommendations from the Committee's previous letter and described the staff's resolution of those comments. The staff agrees that a more flexible regulatory position should be provided, and described the analysis a plant could perform to support partial modifications without a full conversion to Revision 4. The staff expects such an analysis to produce a comparison of required variables under the two revisions, where the differences may be addressed in the accident monitoring instrumentation commitments.

Mr. Barry Marcus of the Office of Nuclear Reactor Regulation then provided a discussion of examples of why such an analysis is important. The first example he discussed involved the downgrading of position indication for safety relief valves. A BWR Owners Group topical report provides justification for use of other variables for monitoring the status of the main steam system, allowing the downgrading of the safety relief valve position indication. Under Revision 4 of Regulatory Guide 1.97, that indication may be removed from the program completely. The second example illustrated how the condensate storage tank level, a key variable for monitoring the status of the auxiliary feedwater system, could similarly be removed from the program.

Mr. Tartal concluded the staff's formal presentation by discussing the changes to Regulatory Position 1 to address the Committee's recommendations.

Mr. Sieber then introduced Mr. Wesley Bowers, Exelon (and Chairman of the BWR Owners Group Regulatory Guide 1.97 Committee), and Mr. Bill Horin, council to the Nuclear Utility Group on Equipment Qualification, to make comments as members of the public. Mr. Bowers also works on the IEEE standards committee that created IEEE Std 497-2002. He briefly described the interest in Revision 4 by the BWR Owners

Group because of the closer linkage between the accident monitoring instrumentation requirements and the emergency operating procedures. He stated that the staff's latest language does provide the flexibility for existing plants to adopt Revision 4. Mr. Horin stated that his organization also now fully supports the revised language in the guide.

During the above discussions, the ACRS Members and NRC staff made the following points:

- Dr. Wallis noted that the examples of downgrading and removing accident monitoring variables can become complicated, and therefore it needs to be taken seriously.

- Mr. Sieber stated that the revisions to Revision 4 of Regulatory Guide 1.97 address the Committee's concerns in that it makes sense to examine the full set of functions before any changes are allowed.

- Mr. Maynard stated that the process of discussion among the staff, the Committee, and the utilities arrived at the right answer for the right reasons.

Committee Action:

The Committee issued a letter to the EDO, dated May 17, 2006, recommending that the staff issue the revised Draft Regulatory Guide 1.97, Revision 4, as final.

VII. Subcommittee Report on Reliability and Probabilistic Risk Assessment

The Subcommittee discussed the probabilistic risk assessment (PRA) for the Economic Simplified Boiling Water Reactor (ESBWR), an advanced design from General Electric (GE) that is in the process of being certified by the NRC. The subcommittee identified several issues for further examination.

X. 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

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

- The Committee considered the EDO's response of April 20, 2006, to comments and recommendations included in the March 28, 2006 ACRS letter on the Draft Final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants."

  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 May 3, 2006. The following items were discussed:

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

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

Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through July 2006 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

The Subcommittee also discussed and developed recommendations on items requiring Committee action.

Candidates to Fill the Vacancy on the Committee

The ACRS Member Candidate Screening Panel and the members interviewed several candidates for membership on the ACRS. The ACRS Chairman provided the members' views to the Panel and the Panel will send a slate of candidates to the Commission in the near future, recommending that the Commission appoint three new members to the ACRS.

Staff Requirements Memorandum (SRM Related to ACRS Request for Additional Resources to Handle Anticipated Increased Workload

In a December 20, 2005 SRM, resulting from the ACRS meeting with the NRC Commissioners on December 8, 2005, the Commission stated that:

> "Following its retreat in January 2006, the ACRS should inform the Commission how the Committee plans to manage the increased workload resulting from anticipated receipt of new reactor designs and combined license (COL) applications"

The Committee responded to the Commission in a report dated March 15, 2006, recommending that the Commission:

- Authorize an increase in the number of Committee members to the maximum of 15 by 2008
- Approve a gradual increase in the ACRS staff, beginning in FY 2006, (2 senior staff engineers, 2 senior technical advisors, and 1 administrative assistant)
- Approve the necessary travel resources for holding additional Subcommittee meetings beginning in FY 2007.

In an SRM dated April 13, 2006, the Commission responded to the Committee's request stating the following:

- The Commission has approved an increase in the number of ACRS members to the maximum of 15 by FY 2008
- The overall budget for the ACRS, including FTE for ACRS members and staff, travel funds, and other expenses should continue to be addressed through the budget process
- In determining if additional resources are needed, the ACRS should continue to look at its current budgeted and baseline activities to determine if the level of ACRS support for some of these activities can be reduced or eliminated.

- Some statements in the March 15, 2006 Committee's report could lead to misinterpretation of the breadth of required ACRS activity under Section 29 of the Atomic Energy Act of 1954, as amended. The Committee should carefully consider what is statutorily required of the Committee, including the activities requested by the Commission, as the Committee identifies, prioritizes, and describes its proposed activities.

- The ACRS and the staff should continue to work together to ensure that staff and ACRS reviews of important technical issues are coordinated in a manner to ensure timely resolution of these issues.

The ACRS/ACNW FY 2007-2008 budget request submitted to the Office of the Chief Financial Officer on March 23, 2006 included a request to increase the number of ACRS members to the maximum of 15; to increase the staff support (5 FTE); and provide the necessary travel resources. This accommodates the Commission direction that the overall budget for the ACRS, including FTE for ACRS members and staff, travel funds, and other expenses should continue to be addressed through the budget process.

With regard to the Commission statement that the ACRS should continue to look at its current budgeted and baseline activities to determine if the level of ACRS support for some of the activities can be reduced or eliminated, it should be noted that the ACRS has a process in place to prioritize items proposed for review during each ACRS meeting. The Planning and Procedures Subcommittee plays a key role in implementing this process. This process was presented to, and discussed with the Commissioners, during the ACRS meeting with the Commission on April 11, 2003. During CY 2005, the Committee decided either not to review, or defer its review, after reconciliation of public comments, about 30 regulatory matters. This is twice as much as that for CY 2004.

With regard to Commission comment on the baseline activities listed in the enclosure to the March 15, 2006 ACRS report, it should be noted that even if some of these items are not explicitly called out in the Atomic Energy Act and may not fall within the statutory purview of the Committee, in accordance with 10 CFR 1.13 the Committee on its own initiative may conduct reviews of specific generic matters or nuclear facility safety-related items. Even with this flexibility, the Committee decided not to address proactive initiatives unless resources permit, and give high priority to the items of significant importance to the Agency.

Quadripartite Meeting Status

The members should provide final papers and power point presentation slides by

Friday, July 28, 2006. We are anticipating receiving the abstract from the Japanese (NSC) prior to the end of May 2006. The Germans (RSK) and French (GPR) have provided most of their abstracts. Draft letters have been prepared for the ACRS Chairman's review and comment, inviting Commissioners, EDO, and NRC Program Office Directors to participate/attend the Quadripartite Meeting.

Streamlining the NRR Rulemaking Process

In a memorandum (COMEXM-06-0006) dated April 7, 2006, Chairman Diaz and Commissioner McGaffigan sent a proposal to Commissioners Merrifield, Jaczko, and Lyons for streamlining the NRR Rulemaking Process. In that memo, it is stated that "... not withstanding 10 CFR 2.809 and the Memorandum of Understanding between the ACRS and the EDO, the staff may waive review by the ACRS at the proposed rule stage." Also, it is stated "comments from the ACRS may be submitted to the Commission either during the comment period for the proposed rule, or following the close of the public comment period, but prior to issuance of the final rule."

If implemented, this proposal will limit the number of opportunities that the ACRS has to review a proposed rule. Also, this will contradict Commission direction in previous SRMs. For example, in the April 5, 2000 SRM, the Commission stated that the ACRS should work with the NRC staff to enhance efforts to risk-inform 10 CFR Part 50, including Appendices A and B. In an April 13, 2006 SRM, the Commission stated that the ACRS and the staff should continue to work together to ensure that staff and ACRS reviews of important technical issues are coordinated in a manner to ensure timely resolution of these issues.

Without involvement by the ACRS in the early stages of the development of a proposed rule, the Committee may not be able to contribute effectively to the development of a rule. During a survey of the NRC staff related to 2005 self-assessment of ACRS, some NRC staff members stated that "Early interaction by the ACRS with the EDO and the NRC staff on the regulatory significance of complex technical issues was very useful."

A draft SRM is being circulated for comment. The ACRS staff, in consultation with the ACRS Chairman, provided comments on the draft SRM for consideration by the Commission.

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

Each year, the members visit a nuclear plant and meet with the Regional Administrator to discuss items of mutual interest. During its April 2006 meeting, the Committee decided to visit the Limerick Nuclear Plant and meet with the Region I Administrator.

The proposed dates for the plant visit and meeting with the Regional Administrator are Tuesday, July 25 thru Thursday, July 27, 2006. Mr. Sieber, Plant Operations Subcommittee Chairman, has agreed to develop a list of proposed topics.

Re-design of ACRS Conference Room

The ACRS Conference room is in the process of being upgraded to improve the audio/visual capabilities, including improving the projector and the teleconferencing/video-teleconferencing capabilities. Additionally, as a result of the Commission's approval to allow the ACRS to expand to its statutory limit of 15 members, the conference room table will be re-designed to accommodate the increase in membership and expanded use of laptop computers. The ACRS/ACNW Office staff is in the process of contracting this job, in an attempt to have this work done prior to the end of FY 2006 (September 30, 2006).

Ethics Training

NRC employees are required to complete an ethics training based on the government-wide standards of conduct regulations. The annual ethics training for ACRS members will be held on June 2, 2006. The topics include Office of Government Ethics regulations, security issues, and official government travel guidelines.

Member Issue — Issues Related to Regulatory Guide 1.174

Dr. Kress stated that there are some "incoherences" with the current Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." He has documented his concerns. If and when the Committee reviews proposed Revision 2 to Regulatory Guide 1.174, he would like to raise these issues.

Based on recent conversation with the NRC staff, we understand that the staff is in the process of revising Revision 1 to Regulatory Guide 1.174 to address PRA quality. The staff has not yet decided whether to address the late containment failure issue in this revision. The staff plans to issue proposed Revision 2 to Regulatory Guide 1.174 in Fall 2006 for public comment. The staff may seek ACRS review after reconciliation of public comments. Other comments on Regulatory Guide 1.174 provided by Mr. Thadani and Dr. Wallis were also discussed.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 533rd ACRS Meeting, May 31 – June 2, 2006.

The 532nd ACRS meeting was adjourned at 6:00 p.m. on May 5, 2006.

APPENDIX I

Interim staff guidance	ADAMS accession No.
Comments on Draft FCSS ISG-10, Rev.1 and Resolution.	ML060470150

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

Dated at Rockville, Maryland this 6th day of April 2006.

For the Nuclear Regulatory Commission.

Melanie A. Galloway,

Chief, Technical Support Group, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards.

[FR Doc. E6-5700 Filed 4-17-06; 8:45 am]

BILLING CODE 7590-01-P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Subcommittee Meeting on Planning and Procedures; Notice of Meeting

The ACRS Subcommittee on Planning and Procedures will hold a meeting on May 3, 2006, Room T-2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed pursuant to 5 U.S.C. 552b(c)(2) and (6) to discuss organizational and personnel matters that relate solely to the internal personnel rules and practices of the ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows: *Wednesday, May 3, 2006, 10:30 a.m.–12 Noon.*

The Subcommittee will discuss proposed ACRS activities and related matters. 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. Sam Duraiswamy (telephone: 301-415-7364) between 7:30 a.m. and 4:15 p.m. (ET) 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 in the agenda.

Dated: April 11, 2006.

Michael R. Snodderly,

Acting Branch Chief, ACRS/ACNW.

[FR Doc. E6-5704 Filed 4-17-06; 8:45 am]

BILLING CODE 7590-01-P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting Notice

In accordance with the purposes of sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on May 4-5, 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, May 4, 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 a.m.: Final Review of the License Renewal Application for the Brunswick Steam Electric Plant (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Carolina Power and Light Company regarding the license renewal application for the Brunswick Steam Electric Plant and the associated NRC staff's final Safety Evaluation Report.

10:15 a.m.–12:15 p.m.: Final Review of the Extended Power Uprate Application for R.E. Ginna Nuclear Plant (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Rochester Gas and Electric Company regarding the extended power uprate application for R.E. Ginna Nuclear Plant and the associated NRC staff's Safety Evaluation.

1:15 p.m.–3:15 p.m.: Final Review of the Extended Power Uprate Application for the Beaver Valley Nuclear Plant (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and FirstEnergy regarding the extended power uprate application for the Beaver Valley Nuclear Plant and the associated NRC staff's Safety Evaluation.

3:30 p.m.–5 p.m.: Proposed Revisions to 10 CFR Part 52, "License, Certifications, and Approvals for Nuclear Power Plants" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding proposed revisions to 10 CFR part 52, "License, Certifications, and Approvals for Nuclear Power Plants."

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

Friday, May 5, 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.: NRC Staff's Response to ACRS Comments on the Draft Final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding their response to ACRS comments included in its March 28, 2006 letter on the Draft Final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants."

9:30 a.m.–9:45 a.m.: Subcommittee Report (Open)—The Committee will hear a report by and hold discussions with the cognizant Chairman of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment (PRA) regarding review of the PRA for the Economic Simplified Boiling Water Reactor (ESBWR) design.

10 a.m.–10:45 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.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

7 p.m.–7:30 p.m.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 29, 2005 (70 FR 56936). 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 4:15 p.m., e.t.

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., e.t., 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.

Dated: April 11, 2006.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E6–5707 Filed 4–17–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Federal Register Notice

DATE: Weeks of April 17, 24, May 1, 8, 15, 22, 2006.

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

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of April 17, 2006—Tentative

There are no meetings scheduled for the Week of April 17, 2006.

Week of April 24, 2006—Tentative

Monday, April 24, 2006

2 p.m. Meeting with Federal Energy Regulatory Commission (FERC), FERC Headquarters, 888 First St., NE., Washington, DC 20426, Room 2C (Public Meeting). Contact: Mike Mayfield, 301–415–3298).

This meeting will be webcast live at the Web address *http://www.ferc.gov.*

Wednesday, April 26, 2006

1 p.m. Discussion of Management Issues (closed—ex. 2).

Thursday, April 27, 2006

1:30 p.m. Meeting with Department of Energy (DOE) on New Reactor Issues (Public Meeting).

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

Week of May 1, 2006—Tentative

Tuesday, May 2, 2006

9:30 a.m. Briefing on status of Emergency Planning Activities—Morning Session (Public Meeting) (Contact: Eric Leeds, 301–415–2334).

1 p.m. Briefing on Status of Emergency Planning Activities—Afternoon Session (Public Meeting).

These meetings will be webcast live at the Web address *http://www.nrc.gov.*

Wednesday, May 3, 2006

9 a.m. Briefing on status of Risk-Informed, Performance-Based Regulation (Public Meeting) (Contact: Eileen McKenna, 301–415–2189).

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

Week of May 8, 2006—Tentative

There are no meetings scheduled for the Week of May 8, 2006.

Week of May 15, 2006—Tentative

Monday, May 15, 2006

1 p.m. Briefing on Status of Implementation of Energy Policy Act of 2005 (Public Meeting) (Contact: Scott Moore, 301–415–7278).

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

Tuesday, May 16, 2006

9:30 a.m. Briefing on Results of the Agency Action Review Meeting—Reactors/Materials (Public Meeting) (Contact: Mark Tonacci, 301–415–4045).

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

Week of May 22, 2006—Tentative

Monday, May 22, 2006

9:30 a.m. Briefing on Equal Employment Opportunity (EEO) Program (Public Meeting) Contact: Corenthis Kelly, 301–415–7380).

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

Week of May 22, 2006—Tentative

Monday, May 22, 2006

9:30 a.m. Briefing on Equal Employment Opportunity (EEO) Program (Public Meeting) (Contact: Corenthis Kelly, 301–415–7380.

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

Wednesday, May 24, 2006

9:30 a.m. Discussion of Security Issues (closed—ex. 1).

1:30 p.m. All Employees Meeting (Public Meeting). Marriott Bethesda North Hotel, Salons, D–H, 5701 Marinelli Road, Rockville, MD 20852.

* * * * *

*The schedule for Commission meetings is subject to change on short

APPENDIX II

April 11, 2006

SCHEDULE AND OUTLINE FOR DISCUSSION
532nd ACRS MEETING
MAY 4-5, 2006

<u>THURSDAY, MAY 4, 2006, 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) (GBW/JTL/SD)
   1.1) Opening statement
   1.2) Items of current interest

2) 8:35 - ~~10:00~~ **9:42** A.M. <u>Final Review of the License Renewal Application for the Brunswick Steam Electric Plant</u> (Open) (JDS/CS/MJ)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and Carolina Power and Light Company regarding the license renewal application for the Brunswick Steam Electric Plant and the associated NRC staff's final Safety Evaluation Report.

~~10:00~~ 9:42 - 10:15 A.M. *****BREAK*****

3) 10:15 - ~~12:15~~ **12:30** P.M. <u>Final Review of the Extended Power Uprate Application for R. E. Ginna Nuclear Plant</u> (Open) (RSD/RC)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff and Rochester Gas and Electric Company regarding the extended power uprate application for R. E. Ginna Nuclear Plant and the associated NRC staff's Safety Evaluation.

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

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

4) 1:15 - ~~3:15~~ P.M. 3:00 — Final Review of the Extended Power Uprate Application for the Beaver Valley Nuclear Plant (Open) (RSD/RC)

4.1) Remarks by the Subcommittee Chairman
4.2) Briefing by and discussions with representatives of the NRC staff and FirstEnergy regarding the extended power uprate application for the Beaver Valley Nuclear Plant and the associated NRC staff's Safety Evaluation.

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

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

5) 3:30 - 5:00 P.M. — Proposed Revisions to 10 CFR Part 52, "License, Certifications, and Approvals for Nuclear Power Plants" (Open) (TSK/DCF/MRS)

5.1) Remarks by the Subcommittee Chairman
5.2) Briefing by and discussions with representatives of the NRC staff regarding proposed revisions to 10 CFR Part 52, "License, Certifications, and Approvals for Nuclear Power Plants," and related matters.

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

5:00 - 5:15 P.M. — ***BREAK***

6) 5:15 - 7:00 P.M. — Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

6.1) Final Review of the License Renewal Application for Brunswick Steam Electric Plant (JDS/CS/MJ)
6.2) Final Review of the Extended Power Uprate Application for R. E. Ginna Nuclear Plant (RSD/RC)
6.3) Final Review of the Extended Power Uprate Application for the Beaver Valley Nuclear Plant (RSD/RC)
6.4) Proposed Revisions to 10 CFR Part 52 (TSK/DCF/MRS)

FRIDAY, MAY 5, 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. 9:02 — NRC Staff's Response to ACRS Comments on the Draft Final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants" (Open) (JDS/EAT)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff regarding their response to ACRS comments included in its March 28, 2006 letter on the Draft Final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants."

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

9)	~~9:30~~ - 9:45 A.M. **9:03**	Subcommittee Report (Open) Report by and discussions with the Chairman of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment (PRA) regarding review of the PRA for the Economic Simplified Boiling Water Reactor (ESBWR) design.
	9:45 - 10:00 A.M.	*****BREAK*****
10)	10:00 - 10:45 A.M.	Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GBW/JTL/SD) 10.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings. 10.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.
11)	10:45 - ~~11:00~~ A.M. **11:18**	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.
12)	11:00 - 7:00 P.M. **(12:15-1:15 P.M. LUNCH)**	Preparation of ACRS Reports (Open) Discussion of proposed ACRS reports on: 12.1) Final Review of the License Renewal Application for Brunswick Steam Electric Plant (JDS/CS/MJ) 12.2) Final Review of the Extended Power Uprate Application for R. E. Ginna Nuclear Plant (RSD/RC) 12.3) Final Review of the Extended Power Uprate Application for the Beaver Valley Nuclear Plant (RSD/RC) 12.4) Proposed Revisions to 10 CFR Part 52 (TSK/DCF/MRS) 12.5) NRC Staff's Response to ACRS Comments on the Draft Final Revision 4 to Regulatory Guide 1.97 (JDS/EAT)

13) 7:00 - 7:30 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.**

APPENDIX III

MEETING ATTENDEES
532nd ACRS MEETING
MAY 4-5, 2006

NRC STAFF (5/4/2006)

J. Davis, NRR
A. Rivera, NRR
J. Hamman, NRR
D. Harrison, NRR
J. Tatum, NRR
G. Armstrong, Jr.
B. Lee, NRR
G. Makar, NRR
J. Nakoski, NRR
S. Miranda, NRR
N. Ray, NRR
R. Laufer, NRR
P. Prescott, NRR
T. Scarbrough, NRR
F. Orr, NRR
S. Laur, NRR
C. Wu, NRR
D. Duvigneaud, NRR
T. Colburn, NRR
K. Wood, NRR
S. Mitra, NRR
M. Heath, NRR
D. Merzke, NRR
J. Medoff, NRR
D. Shum, NRR
J. Nickolaus, NRR
K. Tanabe, NRR
Y. C. Li, NRR
T. Cheng, NRR
H. Ashan, NRR
C. Li, NRR
A. Stubbs, NRR
L. Lund, NRR
L. Regner, NRR
T. Ford, NRR
P. T. Kuo, NRR
J. Zimmerman, NRR
K. Chang, NRR
P. Chen, NRR
B. Rogers, NRR
M. Stutzke, NRR
S. Laur, NRR
J. Tatum, NRR
A. Stubbs, NRR
G. Makar, NRR
R. Laufer, NRR
P. Clifford, NRR
S. Miranda, NRR
C. Wu, NRR
M. Gutierrez, NRR
K. Wood, NRR
N. Gilles, NRR
J. Wilson, NRR
A. El-Bassioni, NRR
W. McKenna, NRR
S. Alexander, NRR
N. Kadambi, RES
M. Tshultz, NRR
J. Calvo, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

R. Surman, Westinghouse
G. Wrobel, Constellation
R. Gillon, Constellation
J. Hartz, Westinghouse
M. Testa, First Energy
C. Keller, First Energy
J. Dunne, Constellation
R. Caceo, Constellation
D. Holm, Constellation
J. Pacher, Constellation
N. Hanley, Stone & Webster
M. Finley, Constellation
G. Verdin, constellation
J. Maracek, FENOC
C. McHugh, Westinghouse
D. Dominics, Westinghouse
C. Savage, Westinghouse
A. Burger, FENOC
D. Durkosh, First Energy
F. W. Etzel, First Energy
A. Levin, Areva NP
D. Fink, Westinghouse
Y. Sing, Westinghouse
G. Kammerdeiner, First Energy
M. Grantham, Progress Energy
J. Lane, Progress Energy
T. Overton, Progress Energy
C. Mallner, Progress Energy
L. Beller, Progress Energy
J. Donahue, Progress Energy
M. Heath, Progress Energy
G. Miller, Progress Energy
R. Stewart, Progress Energy
B. Kitchen, Progress Energy
D. Kunsemiller, FENOC
P. Burke, NMC-Monticello
J. Pairitz, NMC-Monticello
J. Poehler, Constellation Energy
T. Cleary, Dominion
M. Marolin, First Energy
J. Hall, Westinghouse

MEETING ATTENDEES (continued)

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC (5/4/2006)

K. Frederick, FENOC
H. Hearat, FENOC
R. Bain, Stone & Webster
D. Durkosh, FENOC
D. Grabski, FENOC
J. DeBlasio, Westinghouse
P. Sena, FENOC
S. Traiforos, LINK
T. Yamada, JNES

NRC STAFF (5/5/2006)
M. Waterman, RES
S. Arndt, RES
B. Kemner, RES
B. Marcus, NRR
J. Lamb, OEDO
G. Tartal, RES
A. Howe, NRR
H. Gonzales, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

J. A. Beard, GE Nuclear
J. M. Ronney, GE BWROG
W. Bowers, Exelon
B. Horin, Winston & Strawn

APPENDIX IV: FUTURE AGENDA

May 9, 2006

SCHEDULE AND OUTLINE FOR DISCUSSION
533rd ACRS MEETING
MAY 31 - JUNE 1, 2006

WEDNESDAY, MAY 31, 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 - 11:30 A.M. **(10:00-10:15 BREAK)** Draft Final Generic Letter, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations" (Open) (RSD/MAJ/HPN)
- 2.1) Remarks by the Subcommittee Chairman
- 2.2) Briefing by and discussions with representatives of the NRC staff and Nuclear Energy Institute regarding the draft final Generic Letter, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations."

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

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

3) 1:30 - 3:00 P.M. Draft Final Generic Letter 2006-xx, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems" (Open) (MVB/CS)
- 3.1) Remarks by the Subcommittee Chairman
- 3.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final Generic Letter 2006-xx, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems."

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

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

4) 3:15 - 4:15 P.M. Interim Staff Guidance on Aging Management Program for Inaccessible Areas of Boiling Water Reactor (BWR) Mark I Containment Drywell Shell (Open) (MVB/CS)
- 4.1) Remarks by the Subcommittee Chairman
- 4.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed Interim Staff Guidance on Aging Management Program for Inaccessible Areas of BWR Mark I Containment Drywell Shell.

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

4:15 - 4:30 P.M. ***BREAK***

5) 4:30 - 6:30 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
5.1) Draft Final Generic Letter, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations" (RSD/MAJ/HPN)
5.2) Draft Final Generic Letter 2006-xx, "Inaccessible or Underground Cable Failure that Disable Accident Mitigation Systems" (MVB/CS)

THURSDAY, JUNE 1, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

6) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)
6.1) Opening statement
6.2) Items of current interest

7) 8:35 - 11:00 A.M. **(10:00-10:15 BREAK)** Overview of New Reactor Licensing Activities (Open) (TSK/DCF)
7.1) Remarks by the Subcommittee Chairman
7.2) Briefing by and discussions with representatives of the NRC staff regarding staff's activities associated with the licensing of new reactors; early site permits; and combined license applications, as well as the related schedule and milestones.

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

8) 11:15 - 11:45 A.M. Subcommittee Report (Open) (MVB/CS)
Report by and discussions with the Chairman of the Plant License Renewal Subcommittee regarding interim review of the license renewal application for the Monticello Nuclear Power Plant.

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

9) 12:45 - 1:15 P.M. Status Report on the Quality Assessment of Selected NRC Research Projects (Open) (GBW/HPN)
Report by and discussions with the cognizant Panel Chairman regarding the status of the quality assessment of selected NRC research projects.

10) 1:15 -2:00 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GBW/JTL/SD)
10.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.

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

11) 2:00 - 2:15 P.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.

2:15 - 2:30 P.M. ***BREAK***

12) 2:30 - 6:30 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
12.1) Draft Final Generic Letter, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations" (RSD/MAJ/HPN)
12.2) Draft Final Generic Letter 2006-xx, "Inaccessible or Underground Cable Failure that Disable Accident Mitigation Systems" (MVB/CS)

13) 6:30 - 7:00 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:

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- 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
532nd ACRS MEETING
MAY 4-5, 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 NO. — DOCUMENTS

1 Opening Remarks by the ACRS Chairman

A. Items of Interest dated May 4-5, 2006

2 Final Review of the License Renewal Application for the Brunswick Steam Electric Plant

1. Brunswick Steam Electric Plant Units 1 and 2 presentation by Progress Energy [Viewgraphs]
2. Brunswick Steam Electric Plant (BSEP) Units 1 and 2 License Renewal Final Safety Evaluation Report presentation by NRR [Viewgraphs]

3 Final Review of the Extended Power Uprate Application for R. E. Ginna Nuclear Plant

3. Ginna Extended Power Uprate presentation by Constellation Energy [Viewgraphs]
4. NRC Staff Review of Extended Power Uprate Application for R. E. Ginna Nuclear Power Plant presentation by NRR [Viewgraphs]

4 Final Review of the Extended Power Uprate Application for the Beaver Valley Nuclear Plant

5. Beaver Valley Power Station Extended Power Uprate presentation by FENOC [Viewgraphs]
6. NRC Staff Review of Extended Power Uprate Application for Beaver Valley Power Station, Unit Nos. 1 and 2 presentation by NRR [Viewgraphs]

5 Proposed Revisions to 10 CFR Part 52, "License, Certifications and Approvals for Nuclear Power Plants"

7. Part 52 Rulemaking presentation by NRR [Viewgraphs]

8 NRC Staff's Response to ACRS Comments on the Draft Final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants"

8. NRC Staff's Response to ACRS Comments on the Draft Final Revision 4 to Regulatory guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants" presentation by G. Tartal, RES [Viewgraphs]
9. Design and qualification Requirements in Regulatory Guide 1.97, Draft Rev. 4 presentation by W. Bowers, Exelon Corp. [Viewgraphs]

Appendix V
532nd ACRS Meeting

10 <u>Future ACRS Activities/Report of the Planning and Procedures Subcommittee</u>

10. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - May 3, 2006 [Handout #10.1]

11 <u>Reconciliation of ACRS Comments and Recommendations</u>

11. Reconciliation of ACRS Comments and Recommendations [Handout #11.1]

Appendix V
532nd ACRS Meeting

MEETING NOTEBOOK CONTENTS

TAB — DOCUMENTS

2 Review of the License Renewal of the Brunswick Steam Electric Plant, Units 1 and 2

1. Proposed Agenda
2. Status Report
3. Brunswick Steam Electric Plant, Units 1 and 2, Application for Renewed Operating Licenses, dated October 18, 2004
4. Brunswick Steam Electric Plant Inspection Report 05000325/2005008; 05000324/2005008 dated July 22, 2005
5. Audit and Review Report for Plant Aging Management Programs (AMPs) and Aging Management Reviews (AMRs), Brunswick Steam Electric Plant, Units 1 and 2, dated June 21, 2005
6. The Final Safety Evaluation Report Related to the License Renewal of the Brunswick Steam Electric Plant (BSEP), Units 1 and 2, dated March 31, 2006

3 R. E. Ginna Nuclear Power Plant Extended Power Uprate

7. Table of Contents
8. Proposed Schedule
9. Status Report
10. Draft Meeting Summary from Power Uprate Subcommittee meeting on March 15-16, 2006
11. Memorandum from Catherine Haney to John Larkins, "R. E. Ginna Nuclear Power Plant, Draft Safety Evaluation for Proposed Extended Power Uprate (TAC No. MC7382)," dated April 6, 2006

4 Beaver Valley Power Station Extended Power Uprate

12. Table of Contents
13. Proposed Schedule
14. Status Report
15. Memorandum from Catherine Haney to John Larkins, "Beaver Valley Power Station, Units Nos. 1 and 2 (BVPS-1 and 2), Revised Draft Safety Evaluation for Proposed Extended Power Uprate (Tac Nos. MC4645 and MC4646)," April 13, 2006

5 Proposed Revisions to Part 52

16. Table of Contents
17. Proposed Agenda
18. Status Report for Part 52
19. SECY-05-0203, Revision Proposed Rule to Update 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" dated November 3, 2005
20. Staff Requirements, SECY-05-0203, Revision Proposed Rule to Update 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" dated January 30, 2006

21. *Federal Register Notice*: Proposed Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants"
22. Memorandum dated December 14, 2005, from Marvin S. Fertel, Nuclear Energy Institute, to Nils J. Diaz, Chairman, NRC, Subject: Industry Comments on the Part 52 rulemaking Proposal

8 Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants"

23. Table of Contents
24. Proposed Schedule
25. Status Report
26. Letter from G. Wallis, Chairman, ACRS, to L. Reyes, EDO, NRC, "Draft Final Revision 4 to Regulatory Guide 1.97, 'Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants'," dated March 28, 2006
27. Letter from L. Reyes, EDO, NRC, to G. Wallis, Chairman, ACRS, "Draft Final Revision 4 to Regulatory Guide 1.97, 'Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants'," dated April 20, 2006
28. Regulatory Guide 1.97 (draft was issued as DG-1128, dated June 2005), "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4, dated April 2006
29. IEEE Std 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," dated September 30, 2002

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

532nd FULL COMMITTEE MEETINGS

May 4, 2005
Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
Jim Davis	NRR/DLR/RLRC
Aida Rivera	NRR/DE/EQVA
Joyce Hamman	NRR/DE/EQVA
Donnie Harrison	NRR/DRA/APLA
James Tatum	NRR/DSS/SBPB
Gary Armstrong, Jr.	NRR/DIRS/IOLB
Brian Lee	NRR/~~DSS~~/SCVB
Greg Makar	NRR/DCI/CSGB
John A. Nakoski	NRR/~~DSS~~/SPWB
Sam Miranda	NRR/DSS/SPWB
Neil Ray	NRR/DCI/CVIB
Rich Laufer	NRR/DORL/LPLI-1
Paul Prescott	NRR/DE/EQVA
Thomas Scarbrough	NRR/DCI/CPTB
FRANK ORR	NRR/DSS/SPWB
Steve Laur	NRR/DRA
Cheng-Ih (John) Wu	NRR/DE/EEMB
DyLanne Duvigneaud	NRR/DORL/LPL II-2
TIMOTHY G. COLBURN	NRR/DORL/LPL I-1
KENT WOOD	NRR/DSS/SPWB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

532nd FULL COMMITTEE MEETINGS

May 4, 2005
Date

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NAME	NRC ORGANIZATION
S. K. MITRA	NRR / NLR.
MAURICE HEATH	NRR/DLR
DANIEL MERZKE	NRR /DLR
JAMES MEDOFF	NRR/DCI/CFEB
DAVID SHUM	NRR/DSS/SBPB
JIM NICKOLAUS	" " "
Kiyoto Tanabe	NRR/DLR/RLRC/F.A.
Y.C. (Renee) Li	NRR/DE/EEMB
Thomas Cheng	NRR/DE/EGCB
Hans Ashar	NRR/DE/EGCB
Chang Li	NRR/DSS/~~SPLB~~ SBPB
ANGELO STUBBS	NRR/DSS/SBPB
Louise Lund	NRR/DLR/RLRA
LISA REGNER	NRR/DORL
Tanya Ford	NRR/DSS/SBWB
P T Kuo	NRR/DLR
Jake Zimmerman	NRR/DLR/RLRB
Ken Chang	NRR/DLR/RLRC
Pei-Ying Chen	NRR/DE/EEMB
Bill Rogers	NRR/DE/EQVB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

532nd FULL COMMITTEE MEETINGS

May 4, 2005
Date

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NAME	NRC ORGANIZATION
MARTY STUTZKE	NRR/DRA/APLA
Steven Laur	NRR/DRA/APLA
James Tatum	NRR/DSS/SBPB
ANGELO STUBBS	NRR/DSS/SBPB
GREG MAKAR	NRR/DCI/CSGB
Rich Laufer	NRR/DORL/LPLI-1
PAUL CLIFFORD	NRR/DSS/SNPB
Sam Miranda	NRR/DSS/SPWP
CHENG-IH (John) Wu	NRR/DE/EEMB
Mauricio Gutierrez	NRR/DE/EEMB
KENT WOOD	NRR/DSS/SWPB
Nanette Gilles	NRR/DNRL/NGDB
JERRY WILSON	NRR/DNRL/NGDB
Adel El-Bassioni	NRR/APLA
Eileen McKenna	NRR/DPR
STEPHEN ALEXANDER	NRC/NRR/DRA/APOB
NP KADAMBI	RES
Michael Tschiltz	NRR/DRA

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

532nd FULL COMMITTEE MEETINGS

May 4, 2006
Date

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	NAME	AFFILIATION
1	Ralph Surman	Westinghouse Electric Co.
2	George Wrobel	Constellation
3	Roy J. Dillow	Constellation
4	J. HARTZ	WESTINGHOUSE ELECTRIC
5	MIKE TESTA	FIRST ENERGY
6	Colin Keller	First Energy
7	JIM DUNNE	CONSTELLATION
	ROBERT CAVEDO	CONSTELLATION
9	DAVID HOLM	CONSTELLATION
10	Joseph Pacher	Constellation
11	Norman Hanley	Stone & Webster
12	MARK FINLEY	Constellation
13	Gord Verdin	Constellation
14	John Maracek	FENOC
15	CHRIS McHUGH	WESTINGHOUSE
16	DAVE DOMINICIS	WESTINGHOUSE
17	CHRIS SAVAGE	Westinghouse
18	DAVID FINK	Westinghouse
19	Yixing Sung	Westinghouse
	GREG KAMMERDEINER	FIRST ENERGY

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

532nd FULL COMMITTEE MEETINGS

May 4, 2006
Date

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	NAME	AFFILIATION
1	MARK GRANTHAM	PROGRESS ENERGY
2	Jeff Lane	Progress Energy
3	THOMAS OVERTON	PROGRESS ENERGY
4	Christopher Mallner	Progress Energy
5	Lenny Beller	Progress Energy
6	JOE Donahue	Progress Energy
7	Mike Heath	Progress Energy
8	GARRY MILLER	PROGRESS ENERGY
9	ROGER STEWART	PROGRESS ENERGY
10	BOB KITCHEN	PROGRESS ENERGY
11	David Kunsemiller	FENOC
12	PATRICK BURKE	NMC- Monticello
13	Joe PAIRITZ	NMC Monticello
14	JEFF Poehler	Constellation Energy
15	ANTHONY GURGER	FENOC
16	TOM CLEARY	DOMINION
17	DON DURKOSH	FIRSTENERGY
18	Mark Marolan	First Energy
19	F. WILLIAM ETZEL	FIRST ENERGY
	Jeffry HALL	Westinghouse
	Alan Levin	AREVA NP

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

532nd FULL COMMITTEE MEETINGS

May 4, 2006
Date

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	NAME	AFFILIATION
1	Ken Frederick	FENOC
2	Henry Hegrat	FENOC
3	Robert Bain	Stone & Webster
4	DON DURKOSH	FENOC
5	Dave Grabski	FENOC
6	JOHN DeBLASIO	WESTINGHOUSE
7	PETE SENA	FENOC
	SPYROS TRAIFOROS	LINK
9	Jose Calvo	DNRL
10	~~[illegible]~~	
11	Tomoko Yamada	JNES
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

532nd FULL COMMITTEE MEETINGS

May 5, 2005
Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
Mike Waterman	RES/DFERR/ERA/IEEB
STEVEN ARNDT	RES/DFERR/IEEB
BILL KEMPER	RES/DFERR/IEEB
Barry Marcus	NRR/DE/EICB
JOHN G. LAMB	EDO OFFICE
George Tartal	RES/DFERR/IEEB
Allen Howe	NRR/EICB
Hipolito González	RES/DFERR/ME

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

532nd FULL COMMITTEE MEETINGS

May 5, 2006
Date

PLEASE PRINT - PUBLIC SIGN IN

	NAME	AFFILIATION
1	J. Alan Beard	GE Nuclear
2	J M. Kenny	GE BWROG
3	Wes Bowers	Exelon
4	BILL HORIN	WINSTON + STRAWN
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A

ITEMS OF INTEREST

532ND ACRS MEETING

MAY 4-5, 2006

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
532nd MEETING
May 4-5, 2006

YELLOW ANNOUNCEMENT

OTHER NEWS ARTICLE

INSIDE NRC

April 21, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - SECY-06-0041 - PROPOSED STRATEGY TO SUPPORT IMPLEMENTATION OF THE NEW-REACTOR CONSTRUCTION INSPECTION PROGRAM

The Commission has approved an initial approach for implementing the Construction Inspection Program (CIP) for new reactors. This approach will create a dedicated organization in the Region II Office in Atlanta, Georgia, that will have total responsibility for all construction inspection activities across the country, including both the day-to-day onsite inspections and the specialized inspection resources needed to support NRC oversight of the construction of any new nuclear power plants. The Regional Administrator will ensure appropriate management oversight of the initial CIP efforts while maintaining focus on the NRC mission in the safety oversight of Region II operating facilities. This approach is intended to ensure consistency in implementing the new inspection program and quickly incorporate ongoing lessons learned into this entire program.

This initial organization may well need to change in the next few years based on how the new construction environment actually evolves.

This initial approach should be reviewed at least annually to ensure that the safety oversight of operating facilities is not adversely affected and to consider alternatives, as appropriate, to address developments in the actual construction of new facilities.

As construction nears completion, operations resident staff should be assigned to the site from the Region in which the plant would be operating.

The EDO should take action as necessary, including that which may require Commission approval, if organizational needs change in response to expanding construction activities, to ensure the Regional Administrator in Region II is able to maintain the appropriate focus on operating reactors. One example might be creating a second Deputy Regional Administrator devoted to the construction inspection program.

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

April 14, 2006

MEMORANDUM TO: Commissioner Jaczko

FROM: Kenneth R. Hart, Acting Secretary **/RA/**

SUBJECT: COMGBJ-06-0001- ESTABLISHING A POLICY FOR THE REVIEW OF NEW POWER REACTOR COMBINED OPERATING LICENSES

This memorandum is to inform you that the Commission has agreed to task the staff to prepare a paper addressing how it intends to manage the large number of combined operating license applications in the next few years, referred to as the "design-centered-review." The attached SRM provides staff direction on this issue.

This completes action on COMGBJ-06-0001.

Attachment:
As stated

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Lyons
EDO
OGC

April 14, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

FROM: Kenneth R. Hart, Acting Secretary /RA/

SUBJECT: STAFF REQUIREMENTS - COMGBJ-06-0001- ESTABLISHING A POLICY FOR THE REVIEW OF NEW POWER REACTOR COMBINED OPERATING LICENSES

The staff should provide an update on activities related to the design-centered-review process for combined license (COL) applications in the next semiannual status update on new reactor licensing. The staff should inform the Commission promptly of any impediments to conducting COL application reviews using the design-centered-review process.

The staff should continue to utilize the well established planning, budgeting, and performance management (PBPM) process to manage work in the area of new reactor licensing. The staff should ensure that its plans are updated as more detailed information becomes available regarding actual applications and coordination and standardization of these applications for review using the design-centered-review process.

At this time it is premature for the Commission to establish a prioritization scheme regarding new reactor applications. However, the staff should specifically address in the upcoming semiannual status update on new reactor licensing the policy issues related to new reactor applications, as well as the potential need for further identification of significant issues and associated resources, as appropriate. The staff should seek Commission direction on policy issues the staff identifies related to new reactor application reviews as early as possible.

The staff should continue to regularly and publicly meet and work together with all interested potential COL applicants to 1) achieve and clearly document (e.g., in a Regulatory Guide and other appropriate guidance) the scope, level of detail, and format needed for the COL applications to be effectively and efficiently reviewed by the staff, and 2) coordinate the timing and standardization of any future COL submittals to optimize the efficiency of the staff's design-centered-review approach.

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

POLICY ISSUE
(Information)

March 31, 2006 SECY-06-0078

FOR: The Commissioners

FROM: Luis A. Reyes
Executive Director for Operations /RA/

SUBJECT: STATUS OF RESOLUTION OF GSI-191, "ASSESSMENT OF [EFFECT OF] DEBRIS ACCUMULATION ON PWR SUMP PERFORMANCE"

PURPOSE:

The purpose of this paper is to inform the Commission of the status of the resolution of Generic Safety Issue (GSI) 191, "Assessment of [Effect of] Debris Accumulation on Pressurized-Water Reactor (PWR) Sump Performance." This paper does not address any new commitments.

SUMMARY:

The industry is making progress in developing plant-specific specifications for the type and size of PWR containment emergency core cooling system (ECCS) sump screens. Most PWR licensees intend to substantially enlarge the sump screens with passive designs, and the remaining licensees are planning to replace sump screens with active designs. As of March 24, 2006, the staff has received five requests to delay installation of sump screen modifications beyond December 31, 2007, and intends to evaluate those requests with the criteria provided in this paper.

This paper discusses (1) the status of the resolution of GSI-191, (2) staff plans for resolving GSI-191, and (3) staff activities for communicating Nuclear Regulatory Commission (NRC) expectations to licensees.

CONTACT: Hanry Wagage, NRR/DSS/SSIB
301-415-1840

BACKGROUND:

GSI-191 concerns the possibility that debris generated by a loss-of-coolant accident (LOCA) could accumulate on the ECCS sump screen, resulting in a loss of net positive suction head margin. Debris passing through the screen may degrade downstream components such as pumps, valves, and heat exchangers or plug or restrict heat exchanger or fuel flow channels. These phenomena may prevent the ECCS from meeting the criteria of Section 50.46 of Title 10 of the Code of Federal Regulations, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors." Section 50.46(b)(5) requires that licensees design their ECCS systems with capability for long-term cooling. After a successful system initiation, the ECCS must be able to provide cooling to maintain the core temperature at an acceptably low value for a sufficient duration.

During a high-energy primary coolant line break inside the containment of a PWR, energetic pressure waves and fluid jets would impinge on materials near the break such as thermal insulation, coatings, and concrete, damaging and dislodging them. In addition to debris generated by jet forces from the pipe rupture or latent debris in containment entrained by break or spray water flows, debris from unqualified or degraded qualified coatings could be generated as a result of the pressure, temperature, and humidity inside the containment. Finally, chemical products could be created by reactions between the materials in containment and the containment environment following a LOCA. A fraction of the generated debris, latent debris, disbonded coatings, and chemical products might be transported to the pool of water on the containment floor. For a number of postulated LOCA scenarios, the ECCS and containment spray system (CSS) pumps take suction from the recirculation sump, and the debris in the containment pool could subsequently accumulate on the sump screen or be transported into the ECCS and CSS.

The accumulation of the debris on the sump screen would create a debris bed, which would increase the head loss across the screen through a filtering action. If enough debris accumulated, the debris bed could reach a critical thickness such that the head loss through the screens would exceed the net positive suction head margin required to ensure the successful operation of the ECCS and CSS pumps in the recirculation mode. This sump screen clogging could result in severely degraded pump performance, eventual pump failure, and loss of the ECCS and CSS function.

Debris that passes through the sump screen could plug or cause excessive wear of close-tolerance components, deposit on surfaces within the ECCS or CSS systems, or impede coolant flow or cooling capability within the reactor vessel. These phenomena are referred to as "downstream effects." The plugging or wear might cause a component to degrade to the point where it could not perform its designated function (e.g., pump fluid, maintain system pressure, remove heat, or pass and control system flow). Debris that passes through the recirculation sump screen could lodge at a downstream flow restriction such as a high-pressure safety injection throttle valve or the fuel assembly inlet area. Debris blockage at such flow restrictions would impede or prevent the recirculation of coolant through the reactor core, leading to inadequate core cooling. Debris could also deposit on the reactor fuel, reducing the efficiency of decay heat transfer to the recirculating fluid. Debris blockage at flow restrictions in the CSS flowpath, such as containment spray nozzles, would impede or prevent CSS recirculation, leading to inadequate containment heat removal. Debris could also accumulate in close-tolerance subcomponents of pumps and valves. The effect could be either to plug the subcomponent, thereby rendering the component unable to perform its function, or to wear critical close-tolerance subcomponents to the point of degrading component or system function.

Containment coating debris is generated from destruction of coatings within the zone of influence and from postulated failure of degraded qualified coatings and unqualified coatings outside the zone of influence. The zone of influence is the volume of space affected by the impact of energetic pressure waves and fluid jets from a high-energy line break.

In 2004, the staff developed a generic letter (GL) to assist the resolution of GSI-191. In a June 1, 2004, letter commenting on a draft of the proposed GL, the Nuclear Energy Institute (NEI) requested a 5-month extension of the due date (from April 1, 2005, to September 1, 2005) for the requested information, in part to allow licensees time to address chemical effects (ML041550866). The NRC granted this request and changed the due date for licensees submitting the information requested in the GL to September 1, 2005.

On September 13, 2004, the NRC issued the final version of the GL as GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors (PWRs)." This GL requested all PWR licensees (1) to use an NRC-approved methodology to perform a mechanistic evaluation of the potential for post-accident debris blockage and operation with debris-laden fluids to impede or prevent the recirculation functions of ECCS and CSS following all postulated accidents for which these recirculation functions are required and (2) to implement plant modifications or other corrective actions which the evaluation identifies as necessary to ensure system functionality by December 31, 2007. If they could not complete all corrective actions by December 31, 2007, licensees were asked to describe how they would meet the applicable regulatory requirements referenced in the GL until the corrective actions were completed.

On December 6, 2004, the staff issued a safety evaluation of a May 28, 2004, NEI report (ML041550279) on "Pressurized Water Reactor Containment Sump Evaluation Methodology." The NEI report, in conjunction with the staff's safety evaluation (ML043280007 and ML043280008), provided a method acceptable to the staff for evaluating PWR sump performance as requested in the GL.

DISCUSSION:

Status of Licensee GL responses

The GL requested licensees to provide a significant amount of information by September 1, 2005, so that the NRC would have assurance that effective corrective actions were being taken. The September 2005 licensee GL responses revealed that licensees of 66 of the 69 PWRs committed to replace sump screens; the licensees of the remaining 3 PWRs had previously replaced their sump screens.

However, despite the fact that the staff gave the industry an additional 5 months to respond to the GL at the NEI's request, much of the information submitted by licensees was incomplete. For example, the September 2005 licensee GL responses revealed that licensees did not appear to have made significant progress in addressing chemical, downstream, and coatings issues. Two PWR licensees indicated that they would not meet the December 2007 deadline for completing sump modifications.

The staff recently sent requests for additional information to PWR licensees to supply the information missing from the September 2005 GL responses. In a letter dated February 28, 2006, NEI stated that the effort necessary to prepare responses to the requests for additional information on the originally requested schedule would divert resources and attention from the plant strainer modifications and jeopardize the current modification schedules. The staff agreed with the NEI proposal and licensees will now provide supplements to their GL responses by December 31, 2006, for licensees that complete sump screen modifications by that date; and within 90 days of outage completion (but not later than December 31, 2007) for licensees that complete sump screen modifications after 2006.

Criteria for Evaluating Delay of Hardware Changes

The industry has been developing an understanding of certain aspects of chemical and coatings behavior in a post-LOCA environment and establishing an acceptable methodology for evaluating downstream effects. We believe that some licensees may find it difficult to meet the December 2007 deadline for installing a sump screen system that fully demonstrates conformance with the functional requirements. Several licensees have either requested or are planning to request extensions to delay implementation of sump modifications to beyond December 31, 2007. Although the GL stated that the new mechanistic licensing basis should be effective on December 31, 2007, when modifications were to be completed, the staff will consider reasonable extension requests to delay the implementation of final hardware modifications. Provided such extension requests are found acceptable, the new mechanistic licensing basis will be invoked after the modifications are completed.

Proposed extensions to permit changes at the next outage of opportunity after December 2007 may be acceptable if, based on the licensee's request, the staff determines that:

- the licensee has a plant-specific technical/experimental plan with milestones and schedule to address outstanding technical issues with enough margin to account for uncertainties and

- the licensee identifies mitigative measures to be put in place prior to December 31, 2007, and adequately describes how these mitigative measures will minimize the risk of degraded ECCS and CSS functions during the extension period.

For proposed extensions beyond several months, a licensee's request will more likely be accepted if the proposed mitigative measures include temporary physical improvements to the ECCS sump or materials inside containment to better ensure a high level of ECCS sump performance.

Status of Research Activities

The NRC, in collaboration with the Electric Power Research Institute, conducted testing to address concerns about the formation of chemical reaction products in the ECCS containment pool. The testing showed that (1) chemical products/precipitates and gelatinous-like material can form under certain chemical environments in a PWR containment pool during the post-LOCA recirculation phase and (2) changes to important containment parameters such as pool temperature and pH, insulation debris type, and debris concentrations can affect the type and nature of chemical byproducts that form.

A series of NRC-sponsored head loss tests are underway to determine the potential for chemical products to increase the head loss associated with sump screen debris beds. The first test series is investigating the head loss caused by the formation of calcium phosphate due to the reaction of dissolved calcium in environments buffered with trisodium phosphate. Calcium silicate insulation is a prominent source of dissolved calcium in some plants, but other insulation materials and uncoated concrete are also sources. The objective of this test series is to realistically encompass expected post-LOCA containment pool conditions.

The initial results demonstrate that there could be a significant head loss contribution from chemical products associated with calcium silicate/trisodium phosphate containment pool environments. Testing of calcium silicate concentrations representative of reported plant conditions has resulted in significant head loss in these chemical environments due to the formation of chemical products. The NRC issued Information Notice 2005-26 (ML052570220) and follow-on Supplement 1 (ML060170102) to provide the results of head loss tests in an environment containing calcium silicate and trisodium phosphate. The staff is continuing head loss testing for chemical byproducts that developed in simulated PWR sump pool environments that use chemical species other than trisodium phosphate to buffer pH. This testing will be completed in spring 2006.

In addition to research on chemical effects, research is being conducted to address concerns about head loss from debris and about coatings transport. Testing is underway to evaluate head loss associated with standard PWR containment debris materials and to provide data that will be used to develop analytical head loss correlations. Testing is also underway to evaluate the transportability of coating chips to the sump screen and to understand chip characteristics that may affect transportability. Both testing activities will be completed in spring 2006.

Public Meetings With Licensees, Industry, and Advisory Committee on Reactor Safeguards (ACRS)

The staff has held regular public meetings with PWR licensees to share information on the NRC-sponsored test program and to provide feedback on related industry activities. Such meetings were held in December 2004 and January, April, June, September, and November 2005. On February 2, 2006, the staff met with the Palisades nuclear power plant licensee to discuss its plan to temporarily remove trisodium phosphate buffering agent from the containment. This change is intended to reduce uncertainty with regard to sump screen performance resulting from chemical effects associated with calcium silicate/trisodium phosphate until the licensee makes a permanent change to the screen, plant insulation material, and/or the buffering agent. During the meeting the staff asked questions regarding various technical aspects of the proposed change. The licensee stated that they would address these aspects in a license amendment request.

On February 9, 2006, the staff held a public meeting with the industry regarding the status of resolution of GSI-191. The staff expressed concerns with the September 2005 licensee responses to the GL and reemphasized the need for the licensees to meet the December 2007 date for completing sump performance modifications and corrective actions. The staff and industry discussed results from recent chemical effects head loss testing for calcium silicate/trisodium phosphate environments and the status of ongoing research. Also, the licensees of 6 PWRs using calcium silicate insulation and trisodium phosphate buffering agent inside containment discussed actions taken or planned relevant to the Information Notice 2005-26 Supplement. The staff considered the meeting effective in communicating the staff's concerns and expectations regarding licensee GL responses.

The staff continues to interact with the ACRS on GSI-191. In February 2006, the staff met with the ACRS Thermal-Hydraulics Phenomena Subcommittee to discuss progress and future plans on GSI-191 issues. The staff briefed the ACRS Full Committee on March 9, 2006. On March 24, 2006, the ACRS provided a letter to the Commission on GSI-191. In general, the letter supports the staff's emphasis on near-term improvements to containment sumps to reduce the risk of sump screen clogging. In addition, the ACRS calls for the staff to develop improved predictive methods and guidance in several technical areas related to GSI-191. The staff is reviewing the letter and developing a response.

Audits of Selected Licensees

The staff is conducting audits to verify the adequacy of sump modifications and corrective actions in response to the GL. The staff audited two volunteer pilot plants (Crystal River and Fort Calhoun) in 2005. The pilot audits were intended to contribute to the resolution of GSI-191 by providing timely feedback on the implementation of the NRC-approved methodology. For example, the first pilot audit conducted at Crystal River revealed that the licensee used an approach that seemed reasonable to design replacement sump screens. The staff is currently conducting two audits (at Oconee and Watts Bar) and plans to conduct six more audits during 2006 and 2007.

<u>RESOURCES</u>:

The staff plans for resolving GSI-191 discussed in this paper are not expected to require additional resources. The impact of the application of the revised GL extension criteria on NRC resources is expected to be minimal in FY 2007 and beyond.

COORDINATION:

The Office of the General Counsel has no legal objection to this paper.

/RA/

Luis A. Reyes
Executive Director
for Operations

STATEMENT SUBMITTED

BY THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

TO THE

COMMITTEE ON GOVERNMENT REFORM

SUBCOMMITTEE ON NATIONAL SECURITY, EMERGING THREATS AND

INTERNATIONAL RELATIONS

UNITED STATES HOUSE OF REPRESENTATIVES

CONCERNING

NUCLEAR SECURITY

PRESENTED BY

NILS J. DIAZ

CHAIRMAN

SUBMITTED: APRIL 4, 2006

Introduction

Mr. Chairman and Members of the Subcommittee, it is a pleasure to appear before you today to discuss the efforts and accomplishments by the U.S. Nuclear Regulatory Commission (NRC) and its licensees with respect to security at nuclear power plants. The NRC appreciated the opportunity to testify before this Subcommittee on September 14, 2004, regarding nuclear power plant security. The testimony today provides an update of our prior testimony, with a special focus on the Government Accountability Office's (GAO) recent report, GAO-06-388, "Nuclear Power Plants: Efforts Made to Upgrade Security, but the Nuclear Regulatory Commission's Design Basis Threat Process Should Be Improved."

Overview

The NRC's mission is to regulate the Nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of public health and safety, to promote the common defense and security and to protect the environment. On behalf of the entire U.S. Nuclear Regulatory Commission, I am pleased to report that the NRC continues to discharge its responsibilities well, ensuring that the commercial use of radioactive and nuclear materials including nuclear power plants remain safe and secure.

As we have previously reported, nuclear power plants have built-in features that strengthen their ability to withstand externally initiated events. They were designed to withstand catastrophic events including, but not limited to, fire, flood, earthquakes, and tornadoes. These plants were also designed to employ a defense-in-depth strategy, with redundant safety systems and are operated and protected by highly trained staff. Multiple barriers protect the nuclear fuel and the reactor and help prevent or mitigate off-site releases of radioactive materials. The original design features of the reactor facilities, as well as subsequent enhancements, provide substantial inherent protection against a malevolent attack. The NRC and its licensees continue to develop additional protective strategies necessary to complement the facilities capabilities to prevent, detect, and mitigate potential events.

Security at nuclear facilities across the country has long been the subject of NRC, and its predecessor, the Atomic Energy Commission (AEC), regulatory oversight. These security programs are designed, implemented and verified to defend against violent assaults by well-armed, well-trained adversaries. The sites employ sophisticated surveillance equipment, stringent access controls, physical barriers, and well-qualified and trained armed response forces to implement a site-specific defense strategy. Integrated with State, local and Federal law enforcement, we believe the sites are the best protected and tested commercial facilities in the Nation.

Summary of Security Performance

The NRC has a long history of ensuring the safety and security of civilian uses of nuclear power and materials. The NRC's process for reviewing and updating security requirements is based on decades of assessments and lessons learned. These have been integrated into a comprehensive protective scheme of regulatory requirements that are fully executed by our licensees; these requirements to be assessed, and when necessary enhanced.

Security, while clearly receiving added focus following the events of September 11, has long been an intrinsic component of NRC's regulatory framework and was originally addressed in the Atomic Energy Act of 1954, as amended. This Act created the AEC and outlined the essential requirements of a regulatory program to oversee the civilian use of nuclear material. It also provided the basis for regulations designed to guard against theft or diversion of special nuclear material, which included, but was not limited to, materials used in nuclear reactors. In the decade that followed its founding, the AEC required careful maintenance of inventories of special nuclear material and that specific consideration be given to the threat of theft or diversion when considering licensing approvals and actions.

In 1974, the Energy Reorganization Act established the NRC and addressed international terrorism and the need to secure increasing numbers of nuclear facilities and increasing inventories of potentially weapons-usable material in the private sector. The Act required the NRC to review all existing safeguards and security requirements and recommend upgrades where necessary.

During this same period, a Security Agency Study was undertaken. Completed in August 1976, the study focused on the possible establishment of a Federal protective security force to provide protection at commercial power reactors. The study found that the "...creation of a Federal guard force would not result in a higher degree of guard force effectiveness than can be achieved by the use of private guards, properly trained, qualified, trained and certified by the NRC." Shortly after September 11, this issue was again raised. The NRC continues to support the concept that a private security guard force, with special emphasis on performance-based training and full accountability, is the best approach to securing our Nation's commercial nuclear facilities.

In 1977, following the completion of multiple interagency working groups and fact-finding efforts, the NRC amended its regulations to specify physical security measures for nuclear power reactors and special nuclear material . By 1979, additional concerns arose regarding arms proliferation, industrial sabotage and global terrorism. In response, the Commission issued new regulations to incorporate a range of physical security upgrades, including finalizing the DBTs. The use and review of the DBT is an ongoing process; for example, in 1994, the NRC revised the DBT for radiological sabotage to incorporate threat lessons-learned from the 1993 World Trade Center bombing, the Three Mile Island vehicular intrusion in 1993, and terrorist attacks on a variety of foreign facilities. The NRC maintains a deliberate process for reviewing current threat information on an ongoing basis. For almost three decades, the NRC's threat assessment staff has reviewed domestic and international events on a daily basis to determine significance and appropriate NRC actions. Threat assessment and security staff from NRC Headquarters and Regions are available as part of the Information Assessment Team to conduct timely coordination with licensees, law enforcement and the intelligence community to respond to potential threats.

Nuclear Power Plant Defensive Strategies

While nuclear power plants have been required for decades to maintain physical security programs, the terrorist attacks on September 11, 2001, reaffirmed the need for additional collective vigilance, the need for enhanced security, and improved emergency preparedness and incident response capabilities across the Nation's critical infrastructure. As a result, the NRC conducted a comprehensive review of licensees' security programs and made further enhancements to security at a wide range of NRC-regulated activities and facilities.

Immediately following the September 11 attacks, the NRC placed nuclear power plants and other facilities at the highest level of alert using established procedures. On February 25, 2002, the NRC supplemented its security regulations through Orders to power reactor licensees imposing Interim Compensatory Measures, coordinating with law enforcement and intelligence agencies. These measures required power reactor licensees to enhance security and improve their capabilities to respond to a terrorist attack. These Orders constituted a de facto supplement to the DBT, by adding appropriate security enhancements that the NRC deemed necessary in light of the heightened threat environment. Many of these changes, arrived at with no industry input, were among the basis for the subsequent Orders. These enhancements to security included significantly increasing the number of dedicated security guards with threat response duties, increased vehicle standoff distances, consideration of water-borne threats, and improved coordination with law enforcement and intelligence communities, as well as strengthened safety-related mitigation procedures and strategies. Subsequently, on January 7, 2003, the NRC issued additional Orders to licensees to enhance background investigations of persons applying for and holding unescorted access to power reactor facilities.

Furthermore, on April 29, 2003, the NRC, after soliciting and receiving comments from appropriate Federal, State, and industry stakeholders, issued Orders supplementing the DBTs, providing additional details regarding specific adversary characteristics against which power reactors and Category I fuel cycle facilities (facilities that process highly enriched uranium), need to protect. While the specifics of these changes are sensitive or classified, in general these supplements to the existing threat resulted in enhancements such as increased patrols, augmented security forces and capabilities, additional security posts, additional physical barriers, vehicle checks at greater standoff distances, enhanced coordination with law enforcement and military authorities, augmented security and emergency response training, equipment, and communication, and more restrictive site access controls for personnel, including expanded, expedited, and more thorough employee background checks. Concurrently, additional Orders required nuclear power plant licensees to impose enforceable work-hour limits on security force personnel and procedures to evaluate security force fatigue and to enhance training and qualification programs to ensure that armed security personnel are fit, properly trained, and qualified.

The NRC's process for reviewing and updating the specific attributes of the design basis threat is deliberate, thorough, and well-informed. The NRC maintains a competent and dedicated staff that routinely interacts with the intelligence community to gather and review all relevant threat information. Thus, the Commission's decisions and direction to the staff regarding supplementing the DBT, the issuance of security-related Orders, and the subsequent follow-on rulemaking are informed by a variety of sources, including input from NRC staff and external stakeholders.

The NRC conducts security inspection programs to ensure compliance with its requirements, including a baseline inspection program and force-on-force exercises. The NRC conducted force-on-force testing at nuclear power plants since well before the events of September 11 and has since enhanced the program significantly. The NRC, nuclear industry, and certain other stakeholders have leveraged technology, increased funding, and committed additional personnel toward the continual improvement of this effort. The force-on-force exercises test a nuclear power plant's ability to meet requirements that the licensee must defend with a high degree of assurance.

The force-on-force program is a performance-based NRC program to physically test and evaluate the sites' defensive strategies concerning the DBT. The GAO report recognized its value to the continual improvement of security at NRC-regulated nuclear facilities. The NRC continues to enhance the program through the integration of lessons learned from previous exercises. Additionally, the NRC emphasizes use of advanced technology to minimize exercise artificialities, some of which have been identified in the report by GAO. The NRC concurs fully with the report's recommendation that "the NRC continue to evaluate and implement measures to further strengthen the force-on-force inspection program."

The force-on-force inspections at nuclear power plants involve significant preparation on the part of the NRC both in the weeks leading up to and during the on-site visit. NRC employs multiple mock adversary teams whose members possess comprehensive and complementary skill sets. Using proven operational security principles and state-of-the-art equipment, the teams develop, execute, and test threat scenarios through a series of exercises. As reflected in its report to the Committee, the GAO team observed a total of nine such exercises.

Safety is the NRC's first priority in the conduct of each force-on-force exercise. While every participant in the planning and execution of the exercise works to minimize the effects of necessary "artificialities", there are personnel and plant safety limits that must be maintained. Safety briefings and plant-wide notifications of the general schedule must be disclosed, and an increased presence of non-plant personnel will be evident. With that in mind, NRC staff and other participants are not allowed to share any information with the site regarding attack methodologies or tactics that will be employed during the exercise.

GAO Recommendations from its September 14, 2004 Testimony

I would like to take this opportunity to clarify the NRC's response to previous GAO recommendations on nuclear power plant security. GAO's September 2003 report and September 2004 testimony on nuclear power plant security made certain accountability-related recommendations. The first recommendation involved requiring inspectors to conduct follow-up visits to verify that corrective actions have been taken, even when a violation does not reach the threshold for being "cited." Licensees are required to address violations through their Corrective Action Program and the NRC does complete a follow-up visit on specific categories of cited violations.

GAO also recommended collecting and sharing lessons learned among the NRC Regions and licensees. As I have mentioned, there are multiple methods for collecting and sharing information. In addition to generic communications, such as the Regulatory Issue Summaries and Information Notices, the NRC headquarters security staff conducts weekly teleconferences with Regional Security Inspectors, Deputy Regional Administrators and Regional Inspectors. The NRC fully concurs that such communication and information sharing needs to be enhanced continually and is doing so. In addition, the NRC is committed to sharing security best practices among its licensees.

The last 2004 recommendation focused on ensuring the NRC's policy of submitting the results of force-on-force exercises within 45 calendar days was followed. The NRC agrees that reports need to be submitted in a timely manner. The NRC remains committed to improving in this area, as evidenced by a recent review indicating that of the seven most recent reports, only one went beyond the 45 day time line.

<u>GAO Report Regarding Nuclear Power Plant Security and the DBT Revision Process</u>

The GAO report indicates that it reviewed the NRC's documented findings from 27 baseline inspection and force-on-force reports. The findings identified by NRC were the result of good inspection practices on the part of NRC inspectors and good self-assessments by the licensees. In each case, the issue was identified and resolved. Depending on the severity of the finding, inspectors remained on-site until the licensee implemented appropriate compensatory measures. The NRC continues to inspect and licensees continue to be responsive when deficiencies are identified.

In its report, GAO recommended that "NRC improve its process for making changes to the DBT." Additionally, GAO recommended that the NRC should separate the responsibility of receiving and considering external stakeholder feedback from the process of developing the specific threat characteristics in the DBT.

With regard to improving the NRC decision-making process, GAO recommended that the Commission should develop explicit criteria for defining what is and is not reasonable for a private security force to defend against. As stated in our January 24 and February 23, 2006, letters to the GAO, the NRC rejects any implication that the Commissioners' decisions regarding final approval of the supplemented DBT were arbitrary. While additional delineation of relevant considerations might be useful in some circumstances, reasoned judgment within this and other areas of the Commission's statutory decision-making authority does not require, and in fact could be unduly restricted, by detailed prescriptive criteria. Moreover, consistent with governing statutes, the Commission utilized an appropriate decision-making process by providing for a majority Commission position on well-documented staff papers in order for actions to proceed, and documenting individual Commissioner views and proposed modifications for consideration by other Commissioners. The Commission's statutory authority under the Atomic Energy Act and the Energy Reorganization Act, coupled with broad, cross-cutting policy considerations, regular briefings, documented staff papers, and a detailed decision-making process provide the necessary and sufficient criteria for the Commission to make informed decisions regarding the DBT. Moreover, overly-detailed, prescriptive criteria could be detrimental to good governance.

GAO's second recommendation focused on the process used by the Commission to obtain external stakeholder input while developing the supplemented DBT in 2003. The Commission unanimously decided to seek input from all cleared stakeholders on the draft supplemental DBT in January 2003. As noted above, much of the staff's proposed draft DBT derived either explicitly or implicitly from the February 25, 2002 Order on which the Commission had consulted with law enforcement and intelligence agencies. Every State with an affected licensee, every Federal law enforcement, security and intelligence agency, and each affected licensee was asked to comment on the draft within a very short comment period for expeditious deliberations and implementation. Industry input was but one factor, and not a particularly significant one, in the Commission's ultimate decision on the supplemental DBT issued on April 2003. In any case, now that the NRC has returned to our normal DBT review process, wherein we sequentially develop a revision to the DBT then seek external stakeholder input, we believe most of GAO's concern will be alleviated regarding the appearance of undue influence by industry stakeholders.

Path Forward

As the Subcommittee may recall, in its September 2004 testimony, the NRC urged that specific legislative enhancements be enacted. Title VI of the Energy Policy Act of 2005 provides essentially all of these enhancements that collectively will provide additional protection to nuclear power plants. Provisions such as enhanced weaponry, broader fingerprinting and background checks, and criminal penalties for introduction of dangerous weapons and for sabotage of power plants were incorporated.

In addition to and consistent with Congress' legislative actions, the NRC initiated a rulemaking in which it proposed to update the DBT to reflect, among other things, the enhancements and supplementing requirements imposed in the Orders. For example, consideration of a broad range of DBT-related threat factors are explicitly included in NRC's current 10 CFR 73.1 rulemaking. Enhanced weaponry, more rigorous fingerprinting and background checks, and additional measures learned through the implementation of the post September 11 security Orders are also part of a separate 10 CFR 73.55 rulemaking.

Looking toward the future, the NRC recognizes that as the threat environment evolves, we must be positioned to respond decisively. Within the NRC, we must continue to attract and retain employees with the skill sets necessary to manage the challenge. The support of government agencies at the Federal, State and local levels, the legislative branch, and private sector stakeholders must continue to be leveraged to ensure continued success. We are confident that the NRC has the capability and commitment to continue our successful efforts in these areas.

Summary

GAO's audit of nuclear power plant security began in 2003. In the subsequent three years, GAO, the NRC, and multiple nuclear power plant licensees have expended significant resources to provide this Subcommittee and the American public with a greater understanding of the security structure in place to protect nuclear power plants against the potential impact of a terrorist attack. Because some security requirements have been imposed by the NRC through Orders and licensees' security plans, with related safeguards or classified information, cannot be shared in a public forum without compromising security, the GAO's public report should not be considered a full and complete accounting of the state of nuclear power plant security. The sum total of classified and unclassified security requirements provide a comprehensive and appropriate defense against potential terrorist attacks. We remain confident that nuclear power plant security plans are adequate to ensure the protection of the American people from malevolent attempts to damage vital plant equipment and release hazardous radioactive materials to the environment.

We appreciate the opportunity to appear before you today and look forward to answering any questions you might have.

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

April 17, 2006

EA-06-027

Florida Power and Light Company
ATTN: Mr. J. A. Stall, Senior Vice President
Nuclear and Chief Nuclear Officer
P. O. Box 14000
Juno Beach, FL 33408-0420

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (Turkey Point Nuclear Plant - NRC Inspection Report No. 05000250,251/200610)

Dear Mr. Stall:

The purpose of this letter is to provide you with the Nuclear Regulatory Commission's (NRC) final significance determination for a finding involving the B auxiliary feedwater (AFW) pump that was determined to be inoperable due to an incorrectly installed bearing. The finding was documented in NRC Inspection Report No. 05000250,251/2005005 dated January 27, 2006, and was assessed under the significance determination process as a preliminary White issue (i.e., an issue of low to moderate safety significance which may require additional NRC inspection). The cover letter to the inspection report informed Florida Power and Light Company (FPL) of the NRC's preliminary conclusion, provided FPL an opportunity to request a regulatory conference on this matter, and forwarded the details of the NRC's preliminary estimate of the change in core damage frequency for this finding.

In lieu of a regulatory conference, FPL provided a written response dated March 13, 2006. FPL's assessment identified several plant-specific factors beyond those used in the NRC's preliminary estimate and concluded that the finding was of very low risk significance (Green). In summary, the plant-specific factors included the time dependent degradation of the B AFW pump, more recent industry generic failure data, and additional and diverse plant specific features for secondary side heat removal.

After considering the information developed during the inspection and the information FPL provided in its written response, the NRC has concluded that the final inspection finding is appropriately characterized as White in the mitigating systems cornerstone. In summary, the NRC's risk assessment concluded that the factors identified in FPL's written response of March 13, 2006, were insufficient to warrant a significant change in our preliminary risk estimate. Additional information on the NRC's risk estimate, including the disposition of those factors identified in FPL's written response, are included as Enclosure 2 to this letter.

You have 10 business days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Supplement 3.

The NRC has also determined that the finding represents a violation of Technical Specification 3.7.1.2 and 10 CFR Part 50, Appendix B, Criterion XVI. In this case, the B AFW pump was inoperable from approximately September 10, 2003, until November 7, 2005, due to an incorrectly installed bearing. In addition, FPL failed to identify and correct the condition of the pump during this time period as required by 10 CFR Part 50, Appendix B, Criterion XVI, despite several indicators that the pump was degraded. The violation is cited in the attached Notice of Violation (Notice), and the circumstances surrounding it are described in detail in NRC Inspection Report No. 05000250,251/2005005. In accordance with the NRC Enforcement Policy, the Notice is considered escalated enforcement action because it is associated with a White finding.

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.

You are not required to respond to this letter unless the description herein does not accurately reflect your position, or if you choose to provide additional information. For administrative purposes, this letter is issued as a separate NRC Inspection Report (Nos. 05000250,251/2006010) and the above violation is identified as VIO 05000250,251/2006010-01, White Finding - AFW Pump B out of Service Greater than TS Allowed Due to Incorrect Bearing Installation. Accordingly, Apparent Violation (AV) 05000250,251/2005005-02 is closed.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response (should you choose to provide one) 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. Charles A. Casto, Director, Division of Reactor Projects, at (404)562-4500.

Sincerely,

/RA/

William D. Travers
Regional Administrator

Docket Nos. 50-250, 50-251
License Nos. DPR-31, DPR-41

Enclosures:
1. Notice of Violation
2. NRC Evaluation of Risk Significant Factors

cc w/encls: (See next page)

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OEMAIL
OEWEB

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ADAMS: ☐ Yes ACCESSION NUMBER:______________________

* See previous concurrences

OFFICE	RII:DRP	RII:ORA	RII:DRS	OE	NRR		
SIGNATURE	/RA/	/RA/	/RA/	/RA/	/RA/		
NAME	CCASTO	CEVANS	WROGERS	J Luehman *	R Pascarelli *		
DATE	04/03/06	04/03/06	04/03/06	04/12/06	04/06/06		
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: E:\Filenet\ML061090107.wpd

NOTICE OF VIOLATION

Florida Power and Light Company, Inc.
Turkey Point Nuclear Plant
Units 3 and 4

Docket No. 50-250, 50-251
License No. DPR-31, DPR-41
EA-06-027

During an NRC inspection completed on December 31, 2005, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

Technical Specification 3.7.1.2 requires two independent auxiliary feedwater trains including three pumps during plant operation. Action statement 3 states, in part, that with a single auxiliary feedwater pump inoperable, within 4 hours, verify operability of two independent auxiliary feedwater trains and restore the inoperable pump to operable status within 30 days or place the affected units in at least Hot Standby within the next 6 hours.

10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action, states, in part, that measures shall be established to assure that conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.

Contrary to the above, the licensee failed to restore the B auxiliary feedwater pump to operable status within 30 days and did not place the unit in at least Hot Standby during this time. In this case, the B auxiliary feedwater pump was placed in service on September 10, 2003, in an inoperable condition due to a misaligned radial bearing, and the inoperable condition was not identified until November 7, 2005. In addition, the licensee failed to identify and correct the condition adverse to quality during this time frame even though pump bearing vibration levels and oil samples provided indication of the adverse condition.

This violation is associated with a White significance determination process finding for Units 3 and 4 in the mitigating systems cornerstone.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved is already adequately addressed on the docket in the information provided by Florida Power and Light Company's written response of March 13, 2006, and in NRC Inspection Report No. 05000250,251/2005005. However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation - EA-06-027," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator, Region II, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

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.

If you choose to respond, your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), 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. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. 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 17th day of April 2006

NRC EVALUATION OF RISK SIGNIFICANT FACTORS

In lieu of a regulatory conference, Florida Power and Light Company (FPL) provided a written response (dated March 13, 2006) to support its determination of the risk significance of a finding involving an inoperable auxiliary feedwater (AFW) pump. Based on a review of this information, the following is provided as the basis for the NRC's final risk significance determination:

1. Licensee Input - There was a time dependent degradation of AFW pump B. Therefore, up until October 30, 2005, it can be assumed that the pump would have performed its intended function for at least 1 hour. This was based upon the completion of a 2.1-hour surveillance run on October 10, 2005.

   NRC Disposition - The historical information provided about AFW pump B supported the continuation of selecting the fail-to-run basic event surrogate for the performance deficiency. However, the information provided did not support that the surveillance test of October 10, 2005, indicated that the pump would operate for at least 1 hour up until October 30, 2005. From the completion the 2003 bearing mis-installation the pump's vibration was twice what it had been before bearing replacement and progressively worsened to five times its original value over the ensuing time period. Periodic oil samples taken since 2003 were also problematic, and on occasion, the bearing indicated high temperature. Upon disassembly there was grease caking, uneven tooth wear at the coupling, and flaking of the sleeve bearing babbit. The as-found condition clearly indicated the potential for imminent failure. How much earlier in the exposure period the pump would have failed cannot be exactly selected. Therefore, consistent with NRC Manual Chapter 0609, a t/2 correlation was used. All of these factors collectively indicated that pump failure could have happened anytime during the mission time. Recognizing that surveillance performance does not represent actual demand conditions (longer duration, higher pressure, and flow increasing shaft loading), no correlation for bearing performance between test operation and that which would be applicable for an actual demand was provided. Given the bearing's condition/possible pump failure mechanism, insufficient information was provided to support that bearing failure was a function of cumulative pump operation. However, for analysis purposes, the post-reactor trip performance of AFW pump B on March 22, 2005, was evaluated as the break point between failure in less than 1 hour or at greater than 1 hour. This is because the information provided indicated that the pump operated in a post-trip condition for greater than 1 hour. To simplify the analysis, only this exposure period of 63% of a year was quantified. Therefore, the input was partially included in the NRC's final significance determination.

2. Licensee Input - For the exposure time prior to October 30, 2005, with AFW pump B operating for 1 hour prior to failure, the decay heat within the reactor would be lower, allowing a longer time for operators to perform any actions. This would change the performance shaping factors for any human reliability failure probabilities from what was originally developed in the probabilistic risk assessments (PRA).

   NRC Disposition - It is true that the decay heat would be less with a subsequent lengthening of the time to core damage given a longer operation of AFW pump B. However, this does not automatically cause a shift in the human error probability (particularly an order of magnitude shift). Each basic event involving operator error must be evaluated since the performance shaping factor for time may already be a minor input or part of a dependency calculation that is insensitive to time. In addition, the attempted human action may have been a function of a particular setpoint or set of conditions which is not directly affected by the decay heat load. Finally, the quantified analysis was only for the time frame of less than 1 hour. Therefore, an alteration of the human error probabilities did not need to be considered in the quantification analysis.

3. Licensee Input - Due to the time dependent nature of AWF pump B's failure, additional offsite power recovery actions should be added to the PRA.

   NRC Disposition - There was no basis provided by the licensee to support this statement. Consequently, the specific actions could neither be identified nor considered in any quantification of the safety significance. In addition, the dominant accident sequences were not initiated by a loss of offsite power. Therefore, this input was not included in the final significance determination.

4. Licensee Input - Using a 24-hour mission time for the three AFW pumps and two standby steam generator pumps overestimates the probability of failure.

   NRC Disposition - This statement is true for all sequences that exclusively include "failure to run" basic events for any pumps including the pumps mentioned above. However, this is the standard methodology used in PRAs. To perform such an involved calculation would be very time consuming and of marginal value. The licensee did not provide any quantification as to the real effects of using this methodology on the PRA results. Therefore, this input was not included in the final significance determination.

5. Licensee Input - For that exposure time prior to November 30, 2005, with the lower decay heat level, the success criteria for feed and bleed can be modified from 2 of 3 reactor coolant system power operated relief valves to 1 of 3. The results of thermal-hydraulic computer simulations were provided to support this statement.

   NRC Disposition - Given the way in which the final significance determination was accomplished, the risk contribution in which the AFW pump B was postulated to operate for greater than 1 hour was not critical. Consequently, an extensive evaluation into possibly changing the success criteria was not conducted. The NRC recognizes that making changes in PRAs like this on time dependent failures is not atypical. However, the substantial information necessary to support such a change in success criteria was not provided. Therefore, this input was not included in the final significance determination.

6. Licensee Input - The Turkey Point probabilistic safety analysis (PSA) used generic data for basic events. When the new "generic" data from the draft mitigating systems performance indicator (MSPI) program was inserted into the PSA, comparable results with the simplified plant analysis risk (SPAR) were achieved.

   NRC Disposition - The input was included in the final significance determination.

7. Licensee Input - Turkey Point has additional and diverse plant specific secondary side heat removal features. Qualitatively, the risk impact of failing one AFW pump is minimal. Even with the failure of an AFW pump, Turkey Point still has the same degree of

defense-in-depth and margin of safety as a majority of pressurized water reactors. The number and diversity of secondary side heat removal systems provides a strong basis that the loss of one AFW pump is not risk significant.

NRC Disposition - All evaluations performed under Phase 2 and Phase 3 of the significance determination process (SDP) have recognized and included these diverse means in the analysis. The Phase 2 SDP worksheets and the computer models used incorporated the strengths and weaknesses associated with all the features mentioned in FPL's letter of March 13, 2006. Due to the nature of the initiating event or dependency involved with the failure of a particular basic event, all of these features are not available to provide secondary side heat removal. Without an informed understanding of these conditions, one cannot draw an accurate qualitative conclusion regarding how the failure of one AFW pump affects the risk significance. Risk insights gained from reviewing any PRA associated with Turkey Point clearly indicate (depending upon the nature of the failure, exposure time, and possibility for recovery) that such a failure can be of at least low to moderate safety significance.

8. Licensee Input - The fault exposure time does not reflect the time-dependent nature of AFW pump B's condition. The draft MSPI program eliminates fault exposure time. Based upon conservative weighting factors, 12-quarter performance data, and the Institute for Nuclear Power Operations MSPI calculator; the MSPI for AFW is Green.

   NRC Disposition - As has been discussed in numerous public forums, a correlation between MSPI results and SDP results is not appropriate because the two programs monitor two different aspects of performance. In addition, the basis that MSPI is only appropriate to deal with the time-dependent nature of this performance deficiency is not justified. The SDP is adequately suited to deal with this situation as exemplified by using the fail-to-run basic event surrogate in the SDP analyses. Therefore, this input was not included in the final safety significance determination.

9. Licensee Input - Using the Turkey Point PSA model was appropriate for Phase 2 SDP but was overly conservative for a Phase 3 SDP.

   NRC Disposition - Phase 2 SDPs are defined as the results obtained from the SDP Notebooks. When an alteration or amplification of methodologies beyond the notebook is used, it is a defacto Phase 3 analysis. Using PRA models is the normal protocol for Phase 3 analysis. In this particular case, two PRA models (SPAR and licensee full scope) were used – both of which indicated the performance deficiency was of low to moderate safety significance.

10. Licensee Input - Recovery of the AFW pump A is possible upon loss of the A direct current (DC) bus. The recovery activity is proceduralized and involves local operator actions to open valves and throttle flow. It has been quantified with a failure probability of 0.11.

    NRC Disposition - Although the actual methodology associated with acquiring this failure probability was not provided, the NRC examined this possibility and did not include it in the analysis. The procedure for responding to a loss of the A DC bus does not direct or provide a transition to the procedure that contained the instructions for operating AFW pump A locally. In addition, the procedure for responding to a loss of the A DC bus specifically stated that AFW pump A would be lost. Given a loss of the A DC bus and no specific cue, it is highly questionable whether operators would focus on these actions for recovering secondary side heat removal from a human reliability analysis perspective.

11. Licensee Input - FPL offered that, when the AFW pump B failed within 1 hour, the deadheading of the weak pump phenomena included in the PRA model would not occur.

    NRC Disposition - An NRC review of the procedures indicated that 20 minutes was a more appropriate time for possible failure of the weak pump. For ease of analysis, only 66% of the scenario time was quantified in the final significance determination. Therefore, this input was partially included in the final safety significance determination.

12. Licensee Input - The units were shutdown for select periods of time during the exposure time.

    NRC Disposition - At the inception of the SDP, all parties recognized the excessive burden associated with re-creating the actual plant conditions during any exposure time. Consequently, Phase 3 analyses use the averaged PRA model. This input was not included in the final safety significance determination.

13. Licensee Input - When the new basic event probabilities and the time-dependent nature of the AFW pump B failure were inserted into the model, the risk result was less than 1E-6.

    NRC Disposition - For the final significance determination, Sensitivity Cases 1 and 2 of your letter of March 13, 2006, were evaluated. Cutset No. 3 was revised to exclude use of AFW pump A following a loss of DC bus A. The resulting change in core damage frequency was reduced by 63% to account for the postulated pump failure within 1 hour. The result was subsequently reduced by 66% to account for only that period of time when the weak pump could fail after 20 minutes. The quantification was 1.2E-6 (low to moderate safety significance). This quantification did not include the accident sequences that were not dependent upon the weak pump phenomena to fail AFW pump B for this 63% of the exposure time or any of the cutsets associated with the other 37% of the exposure time. Alteration of the original SPAR model by (.63)(.66) produced comparable results. Therefore, the NRC reached a different conclusion than that proposed by FPL regarding this performance deficiency.

In conclusion, after considering the information developed during the inspection and the information FPL provided in its written response, the NRC has concluded that the final inspection finding is appropriately characterized as White in the mitigating systems cornerstone. In summary, the NRC's risk assessment concluded that the factors identified in FPL's written response of March 13, 2006, were insufficient to warrant a significant change in our preliminary risk estimate.

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

April 7, 2006

EA-05-169

Tennessee Valley Authority
ATTN: Mr. K. W. Singer
Chief Nuclear Officer and
Executive Vice President
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (Watts Bar Nuclear Power Plant - NRC Inspection Report No. 05000390/2006007)

Dear Mr. Singer:

The purpose of this letter is to provide you with the Nuclear Regulatory Commission's (NRC) final significance determination for a finding involving a challenge to reactor coolant system (RCS) integrity by multiple pressurizer power-operated relief valve (PORV) actuations and a challenge to RCS inventory control by the loss of RCS coolant via the open PORVs which occurred on February 22, 2005, during transition to solid plant operations.

The finding was documented in NRC Inspection Report No. 05000390/2005013, dated September 7, 2005, and was assessed under the significance determination (SDP) process as a preliminary "greater than Green" issue (i.e., an issue of at least low to moderate safety significance which may require additional NRC inspection). The cover letter to the inspection report informed the Tennessee Valley Authority (TVA) of the NRC's preliminary conclusion, provided TVA with an opportunity to request a regulatory conference on this matter, and forwarded the details of the NRC's preliminary estimate of the change in core damage frequency (CDF) for this finding. At TVA's request, an open regulatory conference was conducted on October 25, 2005, to discuss TVA's position on this issue. The enclosures to this letter include the list of attendees at the regulatory conference and material presented by TVA.

During the conference, TVA presented the results of its estimate of the increase in CDF due to the performance deficiency, including influential assumptions and its analysis methodology. TVA concluded that the finding was of very low safety significance (Green). TVA's analysis included the following five key differences between its evaluation and the NRC's preliminary evaluation: (1) the number of PORV lifts totaled five instead of the seven lifts used in the NRC's event tree, (2) TVA concluded that a more rigorous mathematical treatment of each successive PORV lift was warranted, (3) the Residual Heat Removal (RHR) suction relief valve reliability to open was greater than that assumed in the NRC's preliminary estimate, (4) TVA's analysis included two additional RHR discharge relief valves to relieve increasing RCS pressure which were not included in the NRC's evaluation, and (5) TVA contended that secondary plant cooling was available to prevent core damage. TVA agreed with the NRC's characterization of the finding as a violation of plant procedures. At the regulatory conference, the NRC requested that TVA provide the basis for its RHR system relief valve reliability, the availability of emergency core cooling system sump during the event, and the basis for its human error probabilities used in risk calculations for the event. In addition, TVA agreed to perform a simulator run to assess the likelihood of success by operators in establishing secondary plant cooling with the RHR system isolated and the RCS closed without exceeding the pressure and temperature limits report (PTLR). TVA also agreed to provide the results of this activity to the NRC. This information was subsequently transmitted to the NRC by TVA's letter dated December 27, 2005.

The information provided by TVA caused the NRC to change the event tree that described the finding. TVA's simulator results indicated that when the operators isolated the RHR system to stop the leak from a stuck open RHR relief valve, an over pressurization event will occur. In addition, this resulted in a reduction in the importance of the five key differences that TVA presented at the conference as they no longer have a major impact on the dominant risk sequence and the NRC's final risk characterization.

After considering the information developed during the inspection, the information TVA provided at the conference, and supplemental information as discussed above, the NRC has concluded that the final inspection finding is appropriately characterized as White in the barrier integrity cornerstone.

As part of the NRC's final risk characterization, the dominant risk sequence included an assumption that plant procedures required isolation of the RHR system in response to an RHR relief valve that fails to close during the scenario. The NRC's risk characterization also considered 12 challenges to the PORVs. This value is based on the total demands seen by the PORV circuitry during the scenario. The NRC's evaluation also assumed both PORV block valves to be open instead of the actual condition that existed during the event. This assumption is consistent with the NRC's SDP methodology in which the failure probability of mitigating equipment is determined based on the average condition of the equipment.

These assumptions resulted in a dominant risk sequence that involves the over pressurization of the RCS. The dominant risk sequence would progress with the subsequent unavailability of the PORVs to relieve pressure in the low temperature overpressure protection mode, the subsequent challenge of the RHR relief valves, and the failure of the relief valves to reclose. Subsequent to this, the sequence would progress with successful operator action to isolate the RHR system in accordance with plant procedures which would cause a pressure spike that would exceed the reactor vessel's material limits as specified in the PTLR. This could induce a consequential reactor vessel failure from brittle fracture resulting in subsequent core damage. The staff's preliminary risk assessment assumed the failure probability of the reactor vessel, given this scenario, to be 1.0 which resulted in an estimated delta CDF of approximately 7E-5 per year (Yellow). The staff recognized that the reactor vessel failure probability of 1.0 was based on a conservative assumption and conducted a reassessment using multiple approaches, both quantitative and qualitative, to assess the importance of the low temperature overpressure sequence.

The quantitative results were used as inputs to a qualitative risk evaluation. This evaluation also considered defense-in-depth concepts and the uncertainties of the different numerical methods used in the sensitivity screening analysis. The qualitative analysis was used for the final risk determination reassessment.

During the reassessment, the NRC used multiple approaches to assess the importance of the low temperature overpressure sequence. Low temperature overpressure prevention and mitigation is most critical during RCS water-solid conditions which correlates to the plant conditions of the Watts Bar performance deficiency. Quantitative delta CDF results were within the range of 1E-6 to 1E-5 per year. For the quantitative assessment, the staff performed several sensitivity cases. These cases included application of a vessel failure probability screening value supported by engineering expert opinion regarding the vessel's robustness and, separately, use of previous agency regulatory work for resolution of Generic Safety Issue 94, Additional Low Temperature Overpressure Protection Requirements.

The NRC's preliminary significance determination for the performance deficiency, as transmitted in our letter of September 7, 2005, did not specifically address the change in large early release frequency (LERF), in part, because of the complexities and rigor that would be necessary to quantify an estimate. However, the staff has subsequently conducted a qualitative assessment of the change in LERF and considers this to be non-trivial due, in part, to the potential for the containment to be open to support outage work and the relevant mode of operations. From a defense-in-depth perspective, unlike most other accident initiators that can lead to core damage, a low temperature overpressure event could result in the reactor pressure vessel being unavailable for either subsequent recovery of the reactor core or as an additional barrier for fission product retention. The consequences of such an event can be significant as a result of containment bypass or failure of containment isolation following vessel failure.

Overall, given the above considerations taken in the aggregate, the staff concluded that the finding should be characterized as White in the barrier integrity cornerstone.

You have 10 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 determined that a violation of Technical Specification 5.7.1.1 and Procedure GO-6, Unit Shutdown from Hot Standby to Cold Shutdown, occurred in that TVA personnel failed to slowly raise charging flow to fill the pressure at less than 30 gallons per minute as required by procedure. The violation is set forth in the enclosed Notice of Violation.

You are not required to respond to this letter unless the description herein does not accurately reflect your position or if you choose to provide additional information. For administrative purposes, this letter is issued as a separate NRC Inspection Report (No. 05000390/2006007) and the above violation is identified as VIO 05000390/2006007-01, White Finding - Failure to Implement Shutdown Procedures which Resulted in Pressurizer PORV Actuations. Accordingly, Apparent Violation (AV) 05000390/2005013-01, Failure to Implement and Maintain Shutdown Procedures which Resulted in Pressurizer PORV Actuations, is closed.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response (should you choose to provide one) 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. Charles A. Casto, Director, Division of Reactor Projects, at (404)562-4500.

Sincerely,

/RA/

William D. Travers
Regional Administrator

Docket No.: 50-390
License No.: NPF-90

Enclosures:
1. Notice of Violation
2. List of Attendees
3. Material presented by TVA

cc w/encl: (See next page)

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NAME	CEVANS	CCASTO	RPASCARELLI	CNOLAN		
DATE	3/24/06	3/24/06	03/30/06	04/04/06		
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NOTICE OF VIOLATION

Tennessee Valley Authority
Watts Bar Nuclear Plant
Unit 1

Docket No. 50-390
License No. NPF-90
EA-05-169

During an NRC inspection completed on April 6, 2005, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> Technical Specification 5.7.1.1 requires that written procedures be implemented and maintained covering the activities in the applicable procedures recommended by Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, of which Part 2.j requires a procedure for hot standby to cold shutdown. Procedure GO-6, Unit Shutdown from Hot Standby to Cold Shutdown, Section 5.5, Step [1] [e] states, "Slowly RAISE charging to fill Pressurizer at less than 30 gpm."
>
> Contrary to the above, on February 22, 2005, the licensee failed to follow procedure GO-6, Section 5.5, Step [1] [e], in that net charging flow was raised to a rate that exceeded the 30 gpm procedural specification.
>
> This violation is associated with a White significance determination process finding for Unit 1 in the barrier integrity cornerstone.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved is already adequately addressed on the docket and in the information provided by TVA at the conference (Enclosure 3). However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation - EA-05-169," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator, Region II, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

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.

If you choose to respond, your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS). 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. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. 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 7th day of April 2006

Enclosure 1

LIST OF ATTENDEES

Nuclear Regulatory Commission:
W. Travers, Region II (RII)
J. Shea, Deputy Director, Division of Reactor Projects (DRP), RII
C. Christensen, Deputy Director, Division of Reactor Safety (DRS), RII
S. Cahill, Branch Chief, DRP, RII
J. Bartley, Senior Resident Inspector, DRP, RII
R. Bernhard, Senior Risk Analyst, DRS, RII
L. Trocine, Senior Enforcement Specialist, Office of Enforcement
C. Evans, Regional Attorney and Enforcement Officer, RII
S. Sparks, Senior Enforcement Specialist, RII
M. Reinhart, Office of Reactor Regulation (telecon)
M. Pohida, Office of Reactor Regulation (telecon)
F. Bonnett, Office of Reactor Regulation (telecon)

Tennessee Valley Authority:
M. Skaggs, Site Vice President
D. White, Operations Manager
F. Koontz, Engineering Specialist
P. Pace, Licensing and Industrial Affairs Manager
J. Smith, Sequoyah Licensing Manager
T. Langley, Browns Ferry Licensing Manager
J. Mayo, Watts Bar Shift Manager
C. Borrelli, TVA PSA Engineer
S. Roa, Director of Risk Management Solutions, ABS Consulting

Enclosure 2

April 20, 2006

Mr. James H. Lash
Site Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Post Office Box 4
Shippingport, PA 15077

SUBJECT: NOTICE OF ENFORCEMENT DISCRETION FOR FIRST ENERGY NUCLEAR OPERATING COMPANY REGARDING BEAVER VALLEY POWER STATION UNIT 2 (NOED NO. 06-1-01)

Dear Mr. Lash:

By letter dated April 13, 2006, you requested that the NRC exercise discretion to not enforce compliance with the actions required in the Technical Specifications (TS) for Beaver Valley Power Station Unit 2 (BVPS-2). Specifically, you requested that NRC not enforce the requirements of your plant Technical Specification 3.0.3. Your letter documented information previously discussed with the NRC in a telephone conference on April 11, 2006, at 2:00 p.m. The principal NRC staff members who participated in that telephone conference included:

NRC Region I Staff

- Brian E. Holian, Director, Division of Reactor Projects
- Lawrence T. Doerflein, Acting Director, Division of Reactor Safety (DRS)
- Christopher G. Cahill, Senior Reactor Analyst, DRS
- Roy L. Fuhrmeister, Senior Projects Engineer, DRP
- Paul C. Cataldo, Senior Resident Inspector, Beaver Valley Power Station
- Galen D. Smith, Resident Inspector, Beaver Valley Power Station

NRC Headquarters Staff

- Cornelius F. Holden, Deputy Director, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation (NRR)
- Richard J. Laufer, Chief, Plant Licensing Branch I-1, NRR
- Mike Franovich, Chief, PRA Operational Support and Maintenance Branch, Division of Risk Assessment, NRR
- Margaret A. Kotzalas, Chief, Accident Dose Branch, Division of Risk Assessment, NRR
- Jay Y. Lee, Technical Reviewer, Division of Risk Assessment, NRR
- Harold Walker, Technical Reviewer, Division of Risk Assessment, NRR
- Harold Chernoff, Senior Project Manager, Generic Communications and Power Uprates Branch, Division of Policy and Rulemaking, NRR
- Tim Colburn, Project Manager for Beaver Valley, Division of Operating Reactor Licensing, NRR
- See-Meng Wong, Senior Reactor Analyst, Division of Risk Assessment, NRR

You stated that on April 10, 2006, at 4:36 a.m., you entered TS Limiting Condition for Operation (LCO) 3.7.8.1, when the "A" train of the Supplemental Leak Collection and Release System (SLCRS) was removed from service for routine maintenance and testing. An inadvertent actuation of fire protection deluge systems at 9:24 a.m. on April 11, 2006, caused wetting of the charcoal adsorbers in both trains of SLCRS, rendering both trains of the system inoperable. On April 11, 2006, at 9:24 a.m., you entered TS LCO 3.0.3 due to non-compliance with LCO 3.7.8.1. LCO 3.0.3 would have required bringing the plant to Hot Standby by 4:24 p.m. that afternoon. You requested that a Notice of Enforcement Discretion (NOED) be granted pursuant to the NRC's policy regarding exercise of discretion for an operating facility, set forth in Section VII.C, of the "General Statement of Policy and Procedures for the NRC Enforcement Actions" (Enforcement Policy), NUREG-1600, and be effective for a period of 48 hours. This letter documents our telephone conversations on April 11, 2006, which culminated when we orally issued this NOED at 3:20 p.m. We understand that the condition causing the need for this NOED was corrected by you on April 13, 2006, at 9:45 a.m. (approximately 42 hours into the 48 hour extension), and you re-entered LCO 3.7.8.1 at that time.

Your staff requested this NOED after the inadvertent discharge of multiple fire protection deluge systems at 9:24 a.m. on April 11, 2006. The inadvertent discharges caused wetting of both trains of charcoal filters in the SLCRS system, causing a loss of filtration capability, and resulting in both trains of SLCRS being declared inoperable (the "A" train had previously been declared inoperable and removed from service for maintenance and testing). Having both trains of SLCRS out of service caused a non-compliance with LCO 3.7.8.1, necessitating entry into LCO 3.0.3, and taking action (within one hour) to place the unit in Hot Standby within the next six hours (by 4:24 p.m.), and in Hot Shutdown within the following six hours (by 10:24 p.m.), and Cold Shutdown within the subsequent 24 hours.

During the NOED discussion, you referenced that the SLCRS system was originally credited in the Final Safety Analysis Report (FSAR) accident analysis for iodine removal after a Loss of Coolant Accident (LOCA). License Amendment 139, issued September 10, 2003, adopted alternate control room habitability criteria and selected alternate source term criteria. As a result, SLCRS is no longer credited in the accident analysis for LOCA, and is now only credited for fuel handling accidents with recently irradiated fuel. This change will be reflected in the adoption of Improved Technical Specifications (ITS) for Beaver Valley Unit 2, pending approval by the NRC. The ITS amendment was submitted by FENOC in a letter dated February 25, 2005, and completion of the review of the ITS amendment request is scheduled for late 2006. You further stated that sufficient cooling was available for the Emergency Core Cooling System pumps in the Primary Auxiliary Building through use of the alternate auxiliary building exhaust fans, which are powered from the Class 1E distribution system. The 48 hour NOED extension was requested to allow for replacing the high efficiency particulate filters and charcoal adsorbers in one train of SLCRS to restore compliance with the Technical Specifications. Your staff determined that, based upon prior experience, 48 hours provided sufficient time to complete the replacement and post-maintenance testing.

Regarding the root cause that necessitated the NOED, you stated that the fire protection deluge valves were isolated to prevent recurrence of wetting the filters, and compensatory measures had been implemented in accordance with your fire protection program. Regarding a discussion of equipment availability, you stated that all TS equipment necessary for plant shutdown was available; and further, you would not deliberately remove any safety-related equipment from service, or perform any work which would challenge off-site power during the period of the NOED.

During the NOED discussion on April 11, 2006, your staff provided a qualitative assessment that the criteria of Section D.4 of the NOED guidance were met. The BVPS-2 SLCRS is not credited in the Probabilistic Risk Assessment (PRA) for either its ventilation or filtration functions and therefore did not impact your risk calculations. The small risk increase was due to the swing charging/high head safety injection pump (2CHS-P21C) and the "A" containment instrument air compressor (2IAC-C21A) being out-of-service. In order to provide additional assurance that other critical systems would not be impacted and to enhance the availability of selected systems, you proposed compensatory measures. These compensatory measures included: (1) no other safety related Technical Specification or PRA/safety monitor modeled equipment will be intentionally removed from service for surveillances or maintenance activities during the discretionary period; (2) no discretionary switchyard activities will be allowed during the NOED time period; and (3) the fire protection deluge system water supply to the affected plant areas would be isolated and fire watch tours would be in place during the NOED time frame. Therefore, after review by your Station Operating Review Committee, you concluded that this NOED involved no net increase in radiological risk. The NRC reviewed your assessment and determined it provided an adequate technical basis for your conclusion. Additionally, your April 13, 2006, evaluation concluded that the Incremental Conditional Core Damage Probability (ICCDP) for the equipment out-of-service over a 48-hour duration was 5.5E-10, and the Incremental Conditional Large Early Release Probability (ICLERP) for this duration was below NRC guidance thresholds.

You determined that Section B.2.1, Criterion 1.a and all applicable criteria in Section D to the NRC Inspection Manual, Part 9900: Technical Guidance, "Operations - Notices Of Enforcement Discretion," dated February 7, 2005, have been met. You stated that this NOED is intended to avoid an unnecessary transient which would result from compliance with the Technical Specification Limiting Condition for Operation, in order to minimize potential safety consequences and operational risks. In addition, you determined that a license amendment request to remove the SLCRS requirements during normal power operation had previously been submitted on February 25, 2005, and was currently under review by NRC.

On the basis of the staff's evaluation of your request, we have concluded that granting this NOED is consistent with the Enforcement Policy and staff guidance, and has no adverse impact on the public health and safety or the environment. Therefore it is our intention to not enforce compliance with Technical Specification Limiting Condition for Operation 3.0.3 for the period from 3:20 p.m. on April 11, 2006, until 3:20 p.m. on April 13, 2006. Additionally, the NRC resident inspectors assigned to the Beaver Valley Power Station independently verified that the compensatory actions described above were implemented.

As stated in the Enforcement Policy, action will be taken, to the extent that violations were involved, for the root cause that led to the noncompliance for which this NOED was necessary.

If you have any questions about this matter, please contact Dr. Ronald R. Bellamy, Projects Branch Chief for Beaver Valley, at 610-337-5200.

Sincerely,

/RA/

Brian E. Holian, Director
Division of Reactor Projects

Enclosure: Request for Regional Enforcement Discretion dated April 13, 2006

Docket No. 50-412
license No. NPF-73

cc:
R. Mende, Director - Work Management
T. Cosgrove, Director, Maintenance
P. Sena, Director, Engineering
L. Freeland, Director, Site Performance Improvement and Manager, Regulatory Compliance
D. Jenkins, Attorney, FENOC
B. Sepelak, Supervisor, Nuclear Compliance
M. Clancy, Mayor, Shippingport, PA
D. Allard, PADEP
C. O'Claire, State Liaison to the NRC, State of Ohio
Z. Clayton, EPA-DERR, State of Ohio
Director, Utilities Department, Public Utilities Commission, State of Ohio
D. Hill, Chief, Radiological Health Program, State of West Virginia
J. Lewis, Commissioner, Division of Labor, State of West Virginia
W. Hill, Beaver County Emergency Management Agency
J. Johnsrud, National Energy Committee, Sierra Club

If you have any questions about this matter, please contact Dr. Ronald R. Bellamy, Projects Branch Chief for Beaver Valley, at 610-337-5200.

Sincerely,
/RA/

Brian E. Holian, Director
Division of Reactor Projects

Enclosure: Request for Regional Enforcement Discretion dated April 13, 2006

Docket No. 50-412
license No. NPF-73

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R. Mende, Director - Work Management
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D. Jenkins, Attorney, FENOC
B. Sepelak, Supervisor, Nuclear Compliance
M. Clancy, Mayor, Shippingport, PA
D. Allard, PADEP
C. O'Claire, State Liaison to the NRC, State of Ohio
Z. Clayton, EPA-DERR, State of Ohio
Director, Utilities Department, Public Utilities Commission, State of Ohio
D. Hill, Chief, Radiological Health Program, State of West Virginia
J. Lewis, Commissioner, Division of Labor, State of West Virginia
W. Hill, Beaver County Emergency Management Agency
J. Johnsrud, National Energy Committee, Sierra Club

Distribution w/encl:
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D. Lew, DRP
R. Bellamy, DRP
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A. Rosebrook, DRP
L. Doerflein, DRS
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H. Chernoff, NRR
M. Franovich, NRR
S. Wong, NRR
Region I Docket Room (with concurrences)

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SISP Review Complete: __RRB______ (Reviewer's Initials)

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NAME	RFuhrmeister		RBellamy		LDoerflein		CHolden	
DATE	04/18/06		04/18/06		04/20/06		04/20/06	

OFFICE	RI/DRP	
NAME	BHolian	
DATE	04/20/06	

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 23, 2006

NRC INFORMATION NOTICE 2006-10: USE OF CONCENTRATION CONTROL FOR CRITICALITY SAFETY

ADDRESSEES

All licensees authorized to possess a critical mass of special nuclear material.

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to inform addressees of a concern about the use of concentration control for criticality safety as the primary nuclear criticality safety (NCS) control for unsafe-geometry vessels. It is expected that licensees will review this information and consider actions, as appropriate, to avoid similar problems. Suggestions contained in this IN are not NRC requirements; therefore, no specific action nor written response is required.

DESCRIPTION OF CIRCUMSTANCES

Under 10 CFR Parts 70 and 76, certain licensees processing, storing, or handling critical masses of fissile material are required to identify accident scenarios leading to criticality and develop, implement, and maintain reliable controls to ensure that inadvertent criticality is highly unlikely. Typical criticality safety analyses identify credible accident sequences leading to criticality; identify bounding assumptions related to the processes, equipment, or material being analyzed; and establish limits or boundaries of processes, equipment, or material that comply with corresponding bounding assumptions. Criticality may be deemed not credible when inherent features of the process, equipment, or material in a specific accident sequence leading to criticality can be shown to constrain the reactivity of fissile material within subcritical limits. The safety concern arises when accident scenarios leading to criticality are deemed not credible, based on bounding assumptions that are less than optimal for the system involved.

During a recent review of criticality safety analyses at a fuel cycle licensee facility, NRC inspectors noted routine sampling results showing concentrations near a licensee-proposed bounding concentration value in an unsafe-geometry tank. The fuel cycle licensee relied solely on concentration control to maintain safety in an unsafe geometry tank. The licensee asserted that the NCS method for controlling concentration in the tank was by limiting the concentration in the waste stream leading into the tank. The licensee stated that the waste stream solution was uniform on entry to the tank, and that settling could not result in an unsafe concentration. The analysis demonstrated that by regulating the waste stream concentration to 0.06 grams uranium-235 (U^{235})/liter (0.227 grams U^{235}/gallon), the overall concentration in the tank was guaranteed to remain below the maximum-assumed concentration of 8 grams U^{235}/liter (30.28 grams U^{235}/gallon).

ML060880311

However, the licensee performed chemical analysis on settled solids in the tank and determined that the solids contained fissile material near 8 grams U^{235}/liter. As part of routine sampling, the licensee found a sample with a concentration of 7.74 grams U^{235}/liter (29.30 grams U^{235}/gallon).

The licensee sparged the tank, but only in instances where a sample was to be extracted from the waste solution. The sparging was not credited with, nor used to maintain uniformity in, the tank.

DISCUSSION

The effective use of concentration control requires a system in which concentration changes are well-understood and controlled. NRC is concerned that, in this instance, the licensee maintained the use of concentration control as the single parameter for assuring criticality safety without adequately maintaining a uniform solution and without treating settling in the tank as an upset condition. In this case, the U^{235} concentration limit was chosen from expected concentrations in the tank as a result of limiting inlet waste stream concentrations. The licensee determined that 8 grams U^{235}/liter (30.28 grams U^{235}/gallon) would bound all known U^{235} concentrations in the unfavorable tanks. Without ensuring uniformity within the tanks, it is credible for settling to occur in the solution. The idea of settling within the tank was not considered as an upset condition in this case. Had possible accumulations of settled solids been further evaluated, it may have been shown to be credible for fissile material concentrations in settled solids to exceed the 8-gram (0.018-pound) limit.

An inappropriate use of concentration control was highlighted in an earlier notice (IN-2004-14), on use of a limit on uranium concentration that was less than bounding for the process in which it was applied. A licensee determined that mass controls would limit the uranium concentration in the incinerator ash to less than 21.6 percent throughout the incinerator system. However, material control and accountability (MC&A) sampling data showed concentration levels above 21.6 percent uranium in some parts of the incinerator system. Although the IN focused on the need to establish appropriate interactions between criticality safety and MC&A staff, it also provides another case which exemplifies the need, when using concentration control, for licensees to ensure that they adequately capture all credible bounding scenarios which could potentially impact their system.

Licensee NCS staff should fully understand their systems and all changes that could upset concentration control in the system. Staff should also ensure that all credible scenarios are addressed, and that analyses governing the process bound all such scenarios. During future inspections, NRC inspectors will review systems using this control to ensure that proper controls are in place and that they are properly implemented.

CONTACT

This IN requires no specific action nor written response. If you have any questions about the information in this notice, please contact the technical contact listed below.

/RA/

Robert C. Pierson, Director
Division of Fuel Cycle Safety
and Safeguards
Office of Nuclear Material Safety
and Safeguards

Technical Contact: Natreon Jordan, NMSS
301-415-7648
E-mail: njj@nrc.gov

Enclosure:
List of Recently Issued NMSS Generic Communications

Recently Issued NMSS Generic Communications

Date	GC No.	Subject	Addressees
01/26/06	RIS-02-15, Rev. 1	NRC Approval of Commercial Data Encryption Products For the Electronic Transmission Of Safeguards Information	All authorized recipients and holders of sensitive unclassified safeguards information (SGI).
01/24/06	RIS-06-01	Expiration Date for NRC-Approved Spent Fuel Transportation Routes	The U.S. Nuclear Regulatory Commission (NRC) licensees who transport, or deliver to a carrier for transport, irradiated reactor fuel (spent nuclear fuel (SNF)).
01/13/06	RIS-05-27, Rev. 1	NRC Regulatory Issue Summary 2005-27, Rev. 1, NRC Timeliness Goals, Prioritization of Incoming License Applications and Voluntary Submittal of Schedule for Future Actions for NRC Review	All 10 CFR Parts 71 and 72 licensees and certificate holders.
03/21/06	IN-02-23, Supl. 1	Unauthorized Administration of Byproduct Material for Medical Use	All medical licensees.
01/19/06	IN-06-02	Use of Galvanized Supports and Cable Trays with Meggitt Si 2400 Stainless- Steel-jacketed Electrical Cables	All holders of operating licenses for nuclear reactors except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel; and fuel cycle licensees and certificate holders.

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY REGULATION
WASHINGTON, D.C. 20555-0001

April 11, 2006

NRC INFORMATION NOTICE 2006-09: PERFORMANCE OF NRC-LICENSED INDIVIDUALS WHILE ON DUTY WITH RESPECT TO CONTROL ROOM ATTENTIVENESS

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 instances in which on-duty control room operators were inattentive. This IN serves to reaffirm the necessity for high standards of control room professionalism and operator attentiveness to ensure 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 information notice are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

Recent NRC staff investigations at certain plants identified multiple examples of on-duty (i.e., on-shift) control room operators inattentive to licensed duties. In one case, the NRC determined that an on-duty licensed senior operator was asleep for approximately 4 minutes in the control room and was neither alert nor attentive to duties. This particular issue was further compounded when other crew members deliberately failed to take immediate action to wake the sleeping operator and implement procedural requirements to notify station management of the occurrence and complete a fitness-for-duty (FFD) evaluation.

In a separate case, the NRC determined that several on-duty licensed operators at another facility were inattentive when engaged in the non-business-related use of control room computers. This distracting activity could compromise their ability to monitor and respond to plant indications even though another reactor operator was assigned control panel monitoring duties, generally termed "at-the-controls." The operators engaged in this distracting activity at various times during their shifts.

In each of these examples, the plant licensees determined that the inattentiveness of the on-duty licensed operators was unacceptable control room behavior. The NRC and the licensees took prompt actions to address each occurrence.

ML060110024

BACKGROUND

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.54, "Conditions of Licenses," requires that an operator or senior operator licensed pursuant to 10 CFR Part 55, "Operators' Licenses," be present at the controls at all times during the operation of a facility. The operator at the controls of a nuclear power unit has many responsibilities that include, but are not limited to: (1) adhering to the unit's technical specifications, plant operating procedures, and NRC regulations; (2) reviewing operating data, including data logging and review, in order to ensure safe operation of the unit; and (3) being able to manually initiate engineered safety features during various transient and accident conditions. In order for the operator at the controls of a nuclear power unit to be able to carry out these and other responsibilities in a timely fashion, the operator's attention must be given to the condition of the unit at all times. The operator must be alert to ensure that the unit is operating safely and must be capable of taking action to prevent any progression toward a condition that may be unsafe.

Additionally, 10 CFR 50.54 requires that a senior operator be present in the control room at all times when a nuclear power unit is in an operational mode other than cold shutdown or refueling as defined by the unit's technical specifications. The staffing rule requires the continuous presence of a senior operator in the control room to ensure that (1) an individual is available who can provide the oversight function of the supervisor so that the probability of correctly detecting abnormal events early enough to mitigate potential adverse consequences is increased; (2) the senior operator in the control room is aware of plant conditions prior to and resulting from an abnormal event so that the senior operator's extra experience, training, and knowledge can be used to act promptly to mitigate that event; and (3) the operator at the controls is able to direct attention to performing the immediate actions necessary to mitigate an event, rather than having to brief the senior operator about the background of that event, if the senior operator had been absent from the control room. In order to fulfill these responsibilities, the senior operator must be attentive and alert.

The NRC has previously issued the following generic communications involving inattentive on-duty control room operators:

- NRC Office of Inspection and Enforcement (IE) IN 79-20, Revision 1, "NRC Enforcement Policy - NRC Licensed Individuals" (Agencywide Document Access and Management System (ADAMS) Accession Number ML031180160)
- NRC IE Circular 81-02, "Performance of NRC-Licensed Individuals While on Duty" (ADAMS Accession Number ML031220537)
- NRC IE IN 85-53, "Performance of NRC-Licensed Individuals While on Duty" (ADAMS Accession Number ML031180229)
- NRC IN 87-21, "Shutdown Order Issued Because Licensed Operators Asleep While on Duty" (ADAMS Accession Number ML031180011)

DISCUSSION

As described above, 10 CFR 50.54 states that an operator or senior operator licensed pursuant to 10 CFR Part 55 shall be present at the controls at all times during the operation of the facility and that the continuous presence of a senior operator in the control room is required to ensure that the operator at the controls is able to perform the actions necessary to prevent or mitigate an accident. It is essential that control room operators are (1) highly trained and qualified, (2) physically and mentally fit to carry out their duties, and (3) attentive to plant status relevant to their responsibilities to ensure the continued safe operation of nuclear facilities.

A positive relationship exists between the professionalism of operating personnel at a nuclear power plant and the degree to which the health and safety of the public are protected. Nuclear power plant operators have a professional responsibility to ensure that the facility is operated safely and within the requirements of the facility's license, including its technical specifications and the regulations and orders of the NRC. Mechanical and electrical systems and components required for safety can and do fail. However, the automated safety features of the plant, together with the operator, can identify at an early stage degradation in plant systems that could affect reactor safety. The operator can take action to mitigate the situation. Therefore, nuclear power plant operators on each shift should have knowledge of those aspects of plant status relevant to their responsibilities, should maintain their working environment free of distractions, and should be alert to prevent or mitigate any operational problems. Any behavior, condition, or use of materials that distracts a control room operator from performing assigned duties and responsibilities would cause them to be inattentive to duty.

Instances of on-duty licensed operators sleeping are of particular concern not only because this behavior is in violation of required licensee procedures but it may also represent a failure to recognize the responsibility to operate in a manner that merits public confidence. The NRC Enforcement Policy, Supplement I.C.3, lists inattentiveness to duty on the part of licensed personnel as an example of a Severity Level III violation. The deliberate failure to take immediate corrective actions to awaken a sleeping on-duty licensed operator and immediately relieve the operator of their duties is also a violation involving unacceptable behavior.

Although licensees have established policies to prohibit or minimize distracting activities, the following measures could reduce the possibility of such occurrences: (1) review and revise, as necessary, administrative controls regarding operator performance to ensure that these documents clearly define acceptable standards of operation and provide specific examples of activities that are prohibited while licensed personnel are on duty; (2) discuss these recent incidents with their operations staff to emphasize the importance of alertness, attentiveness, peer checking, and FFD matters; (3) consider applying the on-duty operator administrative controls to other plant personnel; and (4) confirm that the on-duty operator administrative controls are not compromised by other corporate policies.

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 Regulation (NRR) project manager.

/RA/
Christopher I. Grimes, Director
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Office of Nuclear Reactor Regulation

Technical Contacts: Steven Dennis, NRR
301-415-1349
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301-415-1043
E-mail: drd@nrc.gov

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

NRC Yellow Announcement

UNITED STATES NUCLEAR REGULATORY COMMISSION

Announcement No. 024

Date: April 21, 2006

To: **All NRC Employees**

SUBJECT: **APPOINTMENT OF EDWIN M. HACKETT AS DEPUTY DIRECTOR, TECHNICAL REVIEW DIRECTORATE, SPENT FUEL PROJECT OFFICE, OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS**

I am pleased to announce the selection of Edwin M. Hackett as the Deputy Director, Technical Review Directorate, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards. His appointment is effective on May 14, 2006.

Dr. Hackett joined the NRC in 1991 as a Materials Engineer in the Office of Nuclear Regulatory Research (RES). He has since h[illegible]ogressively more responsible positions at NRC, including Senior Materials Engineer in the Office of Nuclear Reactor [illegible]tion (NRR) (1993 -1996), and Section Chief and Assistant Branch Chief (1997-2003) in RES. During his NRC career, Dr. Hackett has worked primarily on nuclear plant primary system structural integrity and also on the structural integrity of dry cask storage systems for spent nuclear fuel. In 2002, he served as Assistant Team Leader for the NRC Davis-Besse Lessons-Learned Task Force. In May 2003, Dr. Hackett was appointed as the Project Director for Project Directorate II in the NRR Division of Licensing Project Management (DLPM). Since October 2005, Dr. Hackett has served as the Deputy Director for the Division of Operating Reactor Licensing (DORL) in NRR.

Prior to joining the NRC, Dr. Hackett worked as a Materials Researcher for the U.S. Navy's David Taylor Research Center in Annapolis, MD (1982 -1991). His work there was largely focused on fatigue and fracture behavior of U.S. Navy structural materials and development of high strength low alloy steels for ship construction. Previously, Dr. Hackett worked for the Ingersoll-Rand Company (1980-1982), and for the Babcock and Wilcox Company's Nuclear Power Generation Division (1976-1978). Dr. Hackett holds a B.S. degree in Materials Engineering from Virginia Tech (1980) and M.S. (1985) and Ph.D. (1989) degrees in Materials Science and Engineering from the Johns Hopkins University.

Please join me in congratulating Ed on his new assignment.

/RA/

Jack R. Strosnider, Director
Office of Nuclear Material Safety
and Safeguards

NF[illegible]low Announcements Index

power & industrial

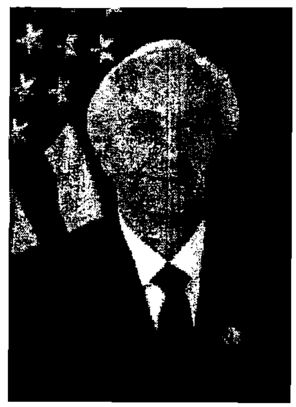
FEDERAL AGENICES

Bush Picks DOD Official to Chair NRC 5/1/2006

By Tom Ichniowski

President Bush has selected Dale Klein, an aide to Defense Secretary Donald Rumsfeld, to be the new chairman of the Nuclear Regulatory Commission. The White House said on April 27 that Bush planned to nominate Klein to a five-year term as an NRC commissioner and also designate him to be NRC Chairman. The NRC position requires Senate confirmation.

If approved, Klein would succeed Nils J. Dias as NRC chairman.



Klein is aide to Rumsfeld for nuclear, chemical and biological programs (Photo Credit: DOD photo by Scott Davis)

Since November 2001, Klein has been Rumsfeld's assistant for nuclear, chemical and biological programs. His duties include policy and planning for nuclear weapons, including safety and security, as well as chemical weapons demilitarization.

Before coming to the Pentagon, Klein was The University of Texas System's vice chancellor for special engineering programs and a professor in the University of Texas at Austin's mechanical engineering department. He also was executive director and chairman of the Amarillo National Research Center.

He has bachelor's and masters degrees and a Ph.D. from the University of Missouri at Columbia.

Inside NRC

Volume 28 / Number 9 / May 1, 2006

NRC will soon re-issue 50.69 regulatory guide for trial use

The NRC has nearly completed a redrafting of Regulatory Guide 1.201 and plans to issue it for trial use within a few weeks, agency staff said at a recent meeting with industry representatives.

The reg guide, RG 1.201, provides guidelines for categorizing structures, systems and components, or SSCs, in nuclear power reactors according to their safety significance. It is intended for use by licensees that plan to implement a new voluntary rule, 10 CFR 50.69, which risk informs NRC's so-called special treatment requirements. The industry would like to use that rule to reduce its procurement costs for safety-related, but low safety significant SSCs, known as "RISC-3" SSCs. But it has expressed disappointment and frustration with a version of RG 1.201 issued earlier this year (INRC, 6 Feb., 16).

The Nuclear Energy Institute and industry's 50.69 task force reviewed RG 1.201 and provided NRC staff with "suggested revisions and comments" in a March 21 letter from Anthony Pietrangelo, NEI's senior director for risk regulation. These comments focused particularly on treatment of RISC-3 SSCs, and were discussed at an April 19 meeting at NRC headquarters.

Donnie Harrison of NRC's probabilistic safety assessment branch gave a presentation at the meeting on changes staff had made to RG 1.201 based in part on industry's comments. The guide's requirements regarding scope of probabilistic risk assessments and the need to conduct uncertainty analyses were "clarified" in the staff's revision, reducing what industry considered unnecessary additional requirements.

Revisions were also made to language in RG 1.201, which described industry's SSC categorization guidance, NEI 00-04, in terms that industry's comments described as "unnecessarily negative" (INRC, 3 April, 16), and language added indicating that industry's guidance "represents an acceptable approach," Harrison said.

NEI will continue to work with NRC staff to "recraft" guidance in Section 12 of the reg guide to more accurately reflect the industry guidance in NEI 00-04, Pietrangelo said at the meeting.

In the March 21 letter, NEI urged that RG 1.201 be issued in final form, rather than for "trial use," because it has already been adequately tested at the 50.69 pilot plants, and because that terminology "implies potential for future changes and will impact the perceived stability and desirability of this rulemaking. If changes are needed, the regulatory guide can simply be revised."

At the meeting, Harrison said NRC staff "disagrees" with this comment and RG 1.201 "still needs to go out for trial use."

The revised reg guide will be issued "as soon as possible" after further revisions, probably within a matter of "weeks," Harrison said.—*Steven Dolley, Washington*

Inside NRC

Volume 28 / Number 8 / April 17, 2006

French ponder organization of new nuclear safety authority

As a bill establishing an independent nuclear regulatory body makes its way through the French parliament, authorities are turning to the practicalities of how the new body might be set up and operate.

Among the questions are how administration employees will be transferred to the new agency's staff, as is foreseen, and how the five-member regulatory commission will interact with that staff.

The Senate and National Assembly have both passed the Nuclear Transparency and Safety Act in a first reading; it will come back for a second reading after the current vacation which ends April 30. If all goes smoothly, the bill could be passed by June. The texts passed by the legislators last month give general instructions for setting up the new Nuclear Safety Authority, or ASN (INRC, 3 April, 10). Details are left to executive decrees that are now being worked on in the administration, notably at the General Directorate for Nuclear Safety and Radiological Protection, or DGSNR.
At a press conference April 4 in Paris to present the French regulatory agency's 2005 report, DGSNR Director General Andre-Claude Lacoste said the legislation makes clear what decisions will be made by the government and what will be left to the new ASN. Regulations must be signed by ministers, who will also have responsibility for important decisions that are now the object of decrees, such as licenses, Lacoste said. The new authority, for its part, will have the power to make "day-to-day decisions" and to impose sanctions, including fines, against licensees who violate regulations.

The future ASN will also have a major mission to inform the public, Lacoste said, adding that the agency "will publish an official bulletin."

Lacoste also addressed the issue of how the ministers with jurisdiction over nuclear safety and radiological protection — those responsible for industry, environment and health — will continue to exercise ultimate governmental authority over nuclear regulation once the new agency is set up, as the French constitution requires. He said options were to set up a single office in the administration serving all three ministers, or three separate services. What's important, he said, was to create the new oversight office(s) before government personnel are transferred to the independent authority.

The legislation provides for transfer of DGSNR's headquarters personnel and the regional nuclear and rad protection inspectorates to the new ASN.

Lacoste said that although ASN will "not be under the authority of any minister," it nevertheless must respond to requests made by the ministers.

As for accountability of the new ASN, Lacoste said that under the bill, "the parliament has the right to ask for testimony from the commission chairman." But he did not envisage a process of regular public hearings with sworn testimony, which would be unusual under the French legislative system.

Lacoste argued that "in the end, the real power (of the parliament) remains the (agency's) budget," which legislators will have to approve every year as part of the government's authorization bill.

The DGSNR chief said that his agency was beginning to study the workings of regulatory commissions both in France and abroad to make suggestions for organizing the new ASN. He said the investigation would consider both the composition of the regulatory commissions — the professional profiles of commissioners — and how the commissions and the staffs interact. There are some 30 "independent administrative authorities" in France, regulating everything from telecommunications to financial markets.

Lacoste told Platts he and colleagues would look at the workings of the US NRC and Canada's Nuclear Safety Commission, in particular as concerns their public information policies.

Under Lacoste, DGSNR has gone much further than is traditional for an office of the French administration in providing public information on its activities, without a clear legal basis to do so. DGSNR has also set up a quasi-formal system of regulatory decisions that was also without legal foundation; the new law is designed to fill that legal void. As an independent agency, ASN will be able to go further still, probably releasing information that comes from licensees. But Lacoste did not appear to expect the meetings of the new commission to be public.

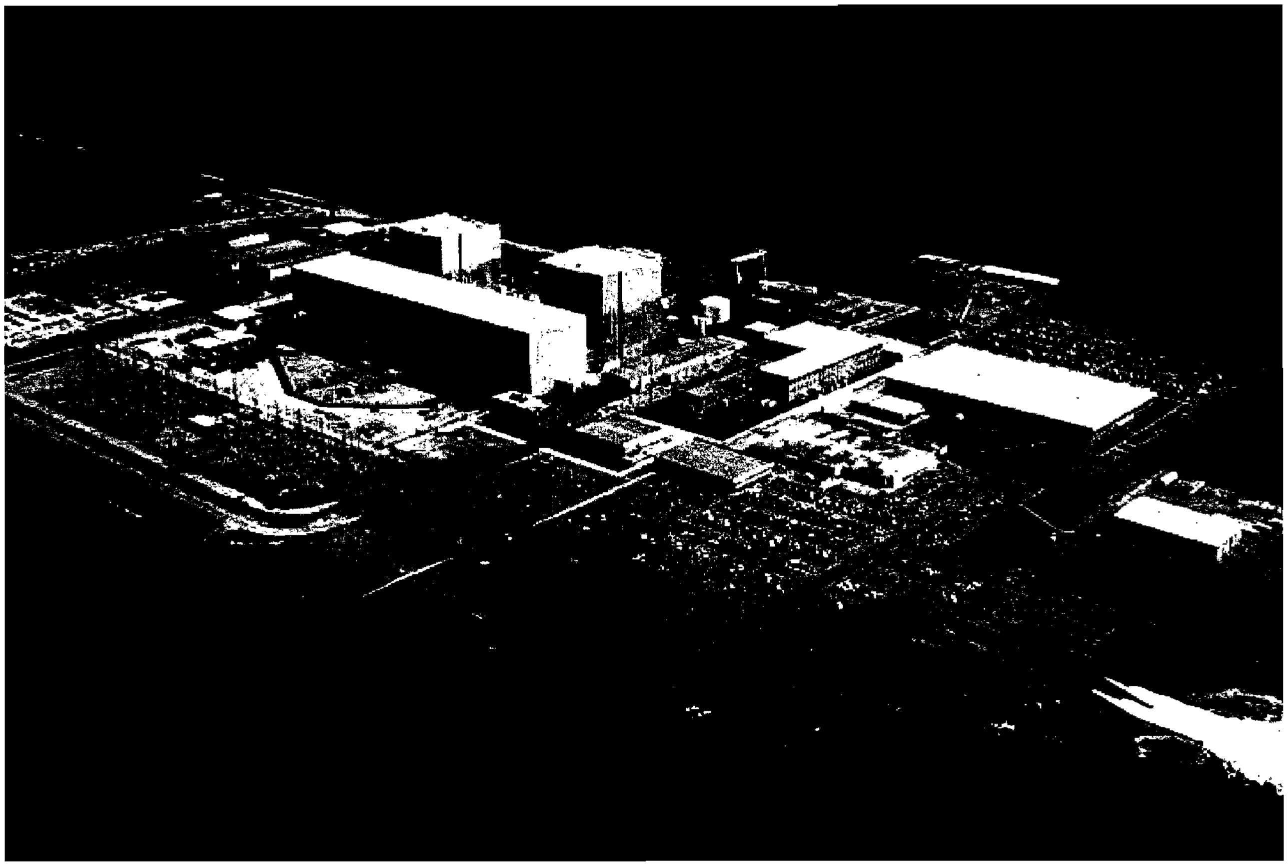
Lacoste also said the administration was "thinking about how the commission could be composed." He suggested that its members could be chosen to represent several different skills, from safety and radiation protection to legal, industrial and "social or sociological" competencies.

Three of the future ASN commissioners are to be named by the President of the Republic, and one each by the presidents of the Senate and National Assembly.

Lacoste deflected a question about whether he expected or wanted to be named to the new agency. He had earlier told journalists that it would be his last annual press conference as head of DGSNR. Lacoste automatically becomes ineligible for administration service in November, when he turns 65. The Transparency and Safety bill as voted out of Assembly requires that members of the ASN be named before their 65th birthday. That means that if Lacoste is to have the job, the government and parliament have to work quickly after passage of the act to select five candidates and promulgate the commission's nomination by decree. This provision would also disqualify for the top ASN job Jean Syrota, an important figure in French energy circles who has been rumored to be in line for the ASN chairmanship but who just left the chairmanship of the Energy Regulatory Commission because of the age-65 rule.

Among others who were rumored to be under consideration for the ASN was Anne Lauvergeon, current chairman of state-owned Areva. But that smacks more of political maneuvering by those who want her dethroned, observers said.
Lauvergeon is hoping for renomination when her first Areva terms expires in June.
Several other potential candidates who are now in the administration or head state-owned companies were cited by the financial daily Les Echos several weeks ago.
—Ann MacLachlan, Paris

Brunswick Steam Electric Plant
Units 1 and 2



INDEX

(1)

Brunswick Steam Electric Plant
Units 1 and 2

License Renewal Presentation to ACRS

NGG Nuclear Generation Group

Progress Energy

Agenda

- A. Overview of License Renewal Application
- B. Operating Experience
  - a. Drywell Liner
  - b. EPU Vibration
- C. Major Equipment Replacements/Repairs
- D. Major Exceptions to the GALL Report
- E. Commitment Tracking

INDEX

Description of BSEP

- Located in Southport, NC
- Cape Fear River is Ultimate Heat Sink
- Dual unit GE BWR 4 with Mark I Reinforced Concrete Containment
- Both units have achieved 120% power uprate
- Current License Expiration
  - Unit 1 September 2016
  - Unit 2 December 2014

INDEX

Application Background

- LRA used Class of 2003 Format – May 2003
- Information in the LRA was developed in plant calculations
- Addressed ISGs 1 through 20
- 34 Aging Management Programs Identified
- No Open Items or Confirmatory Items

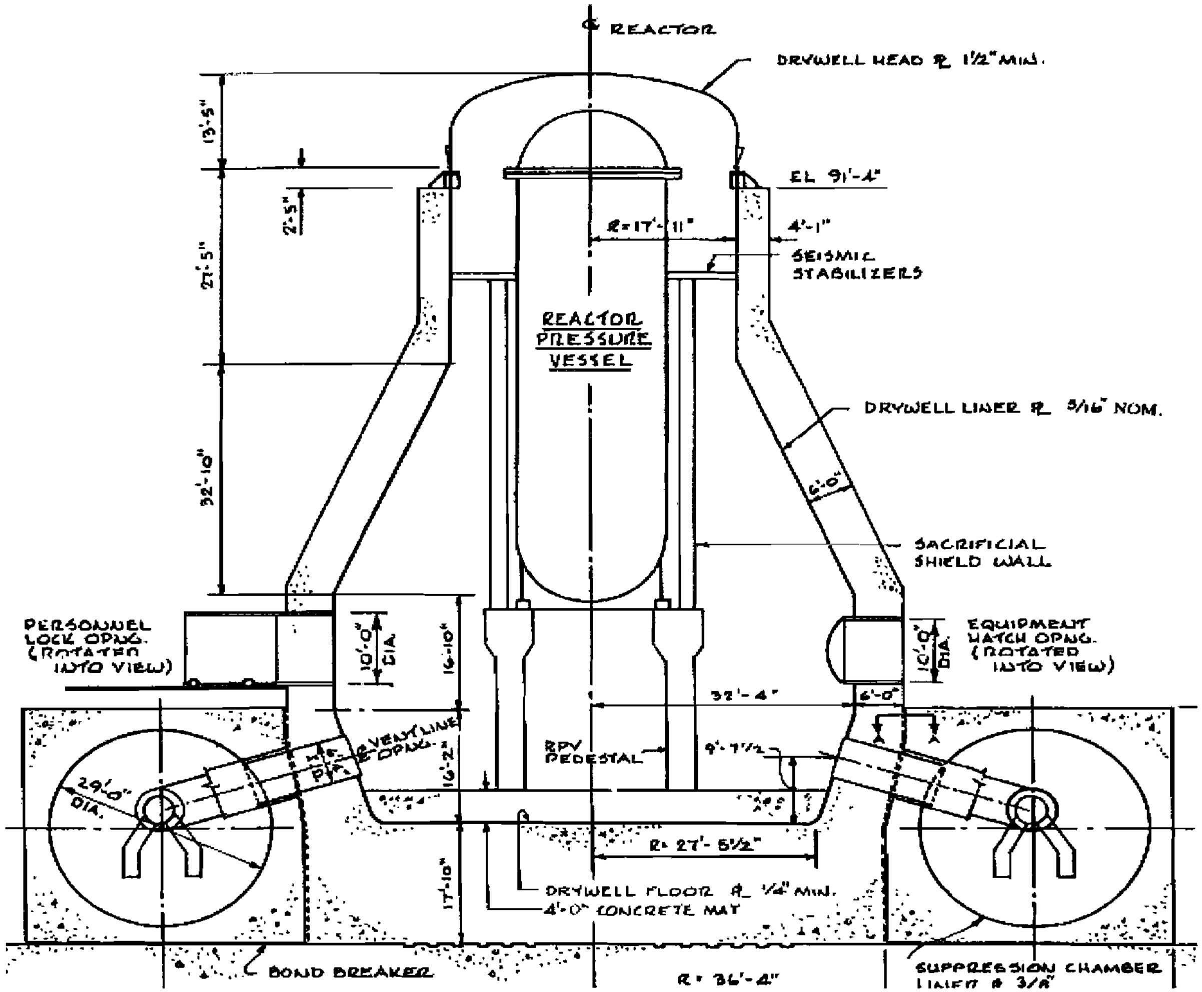
Drywell Liner Operating Experience

Tom Overton

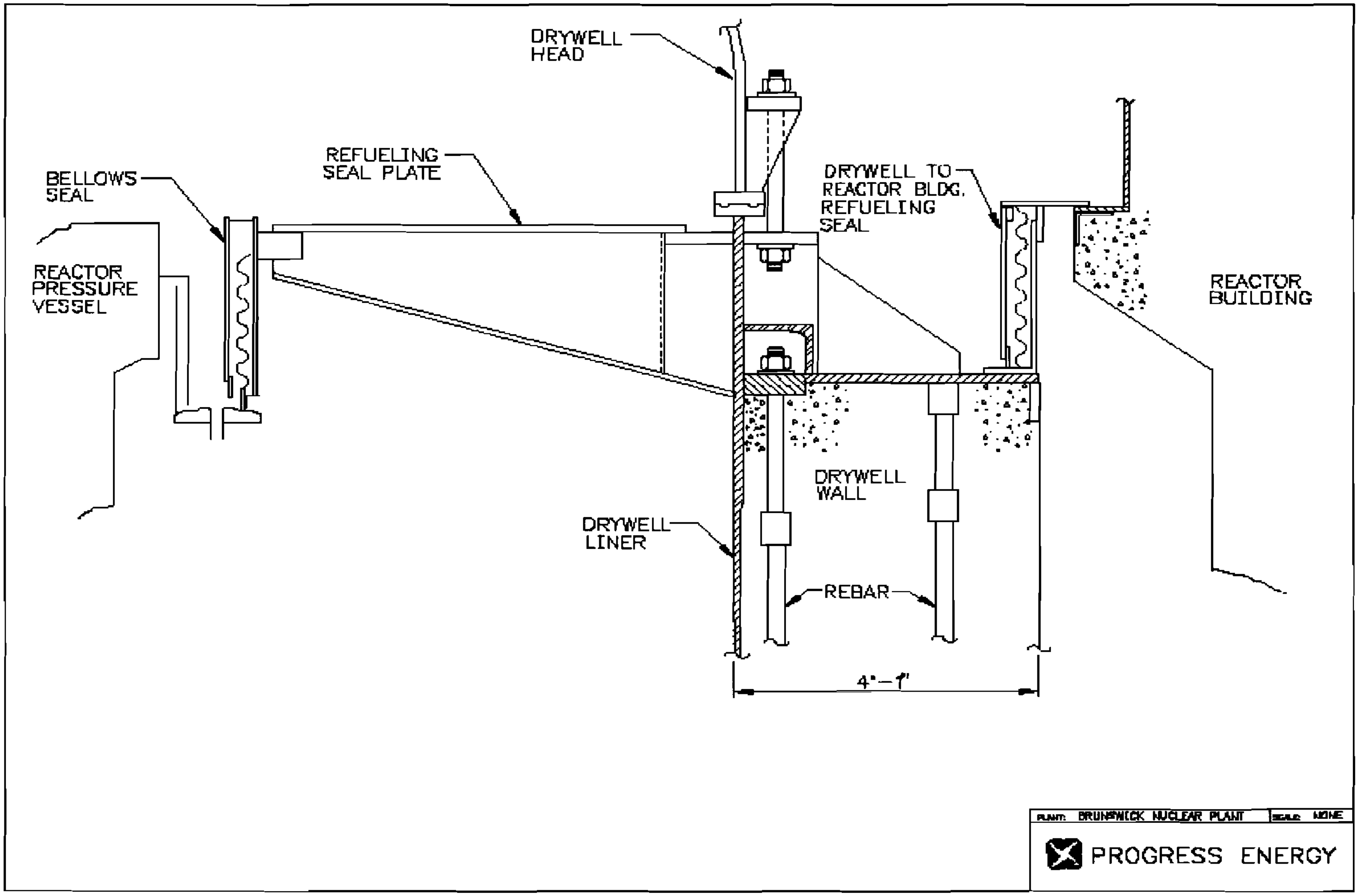
BWR Mark I Steel Lined Reinforced Concrete Containment

- Only BWR Mark I steel lined, reinforced concrete containment
  - No annular space between the metallic liner and the reinforced concrete
  - No sand bed region

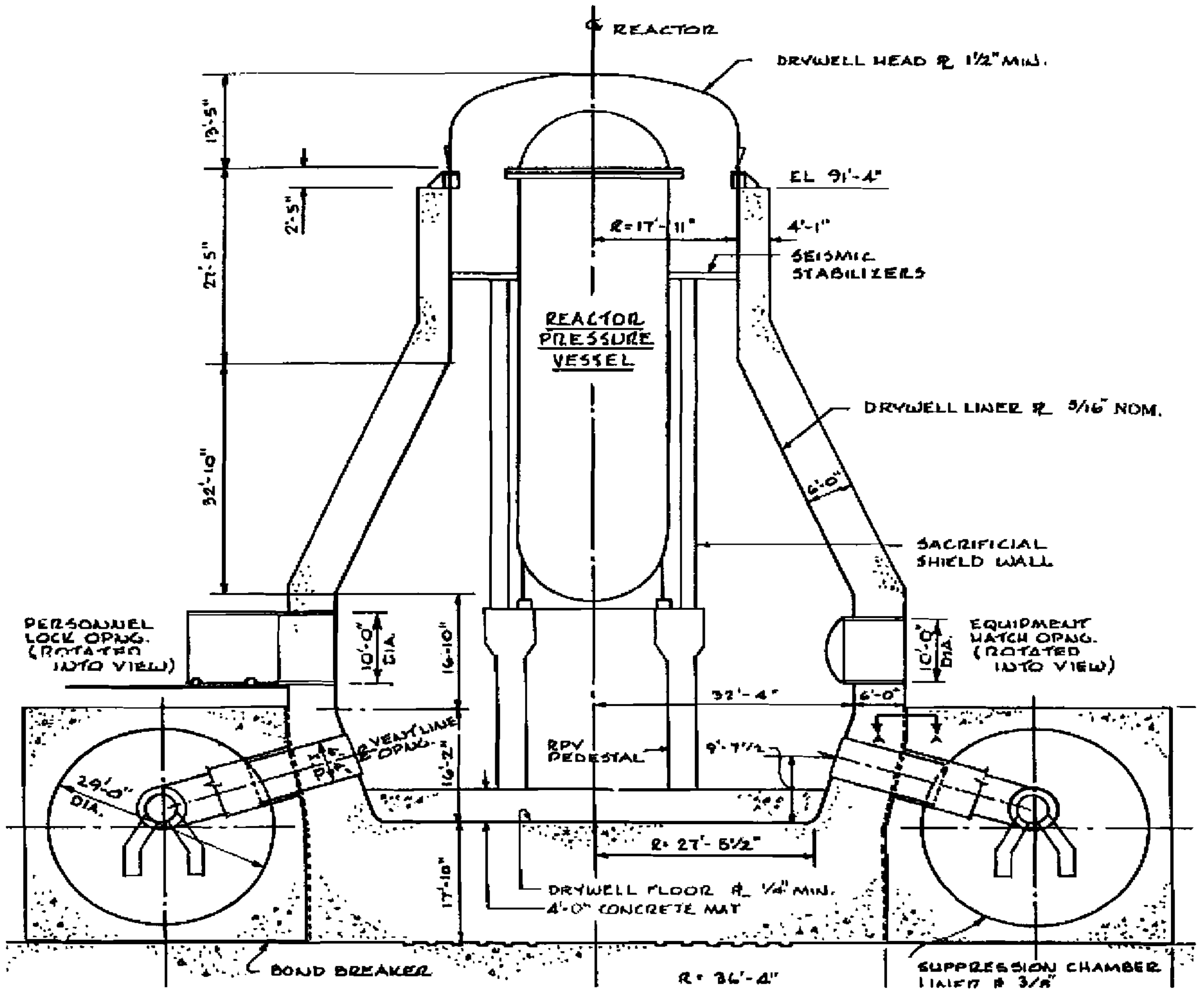
Brunswick
Mark I Steel Lined Reinforced Concrete Containment



Refueling Bellows



Brunswick
Mark I Steel Lined Reinforced Concrete Containment



Power Uprate Vibration Operating Experience

Mark Grantham

EPU Vibration Experience

- Main Steam and FW Vibration Monitoring
  - Based on ASME/ANSI OM Part 3
  - Modal analysis performed to determine sensor locations
  - Vibration levels increased as part of EPU implementation, but remain well below code allowable stresses

EPU Vibration Experience
Main Steam Line Piping

- Acceleration Study for Unit 1 Main Steam Node 26

Channel Number	Measured Acceleration at EPU (g)	Allowable Acceleration (g)	Percent of Allowable (%)
11	0.126	1.014	12.4
12	0.108	0.698	15.5

EPU Vibration Experience
Feedwater Piping

- Acceleration Study for Unit 1 Feedwater Node 37

[redacted]			
27	0.020	2.155	1.0
28	0.021	2.364	1.0

EPU Vibration Experience
BOP Piping

- Fatigue failure of EHC return line for main turbine control valves
  - Interim power level was likely a contributor
  - Industry OE with these types of failure exists
  - Piping modified to a flexible connection
- Socket welded drain line failures
  - Previous industry and BSEP OE with these types of failures
  - Changed socket weld configurations to a more fatigue tolerant design

EPU Vibration Experience
BOP Piping

- Rod Hung BOP Piping
  - Low frequency vibration
  - Modified to add lateral supports

Major Equipment Replaced or Repaired

Mark Grantham

Major Equipment Replacement/Repair

- Replaced Power Range Neutron Monitoring System
- Replaced Main Power Transformers
- Replaced High Pressure Turbines
- Rewound Main Generator Stators
- Replaced FW Heaters
  - Unit 1 - 5 FW Heaters
  - Unit 2 - 1 FW Heater
- Replaced Reactor Feed Pump Turbines, Governor, and pump rotating assemblies

Major Equipment Replacement/Repair

- Replaced Condensate Pumps and Motors
- Replaced Isophase Bus Cooling Units
- Fire Detection System (in progress)

Major Exceptions to GALL

Mike Heath

Major Exceptions to GALL

Fire Protection Program

NUREG 1801:

- Visual Inspection of 10% of Each Type Penetration Once Every Refueling Outage.

BSEP:

- Visual Inspection of a Statistical Sample Once Every 18 Months.

Major Exceptions to GALL

Fire Protection Program – continued

NUREG 1801:

- Test Halon/CO2 Every 6 Months.

BSEP:

- Test Halon Annually/Test CO2 Every 18 Months.

Major Exceptions to GALL

<u>Fuel Oil Chemistry Program</u>

NUREG 1801:

- Internal Surfaces of Tanks are Cleaned and Inspected.

BSEP:

- Only Main Fuel Oil Tank Internal Surface is Inspected and Cleaned if Needed. Smaller Tanks Have External UT of Tank Bottom.

Commitment Tracking

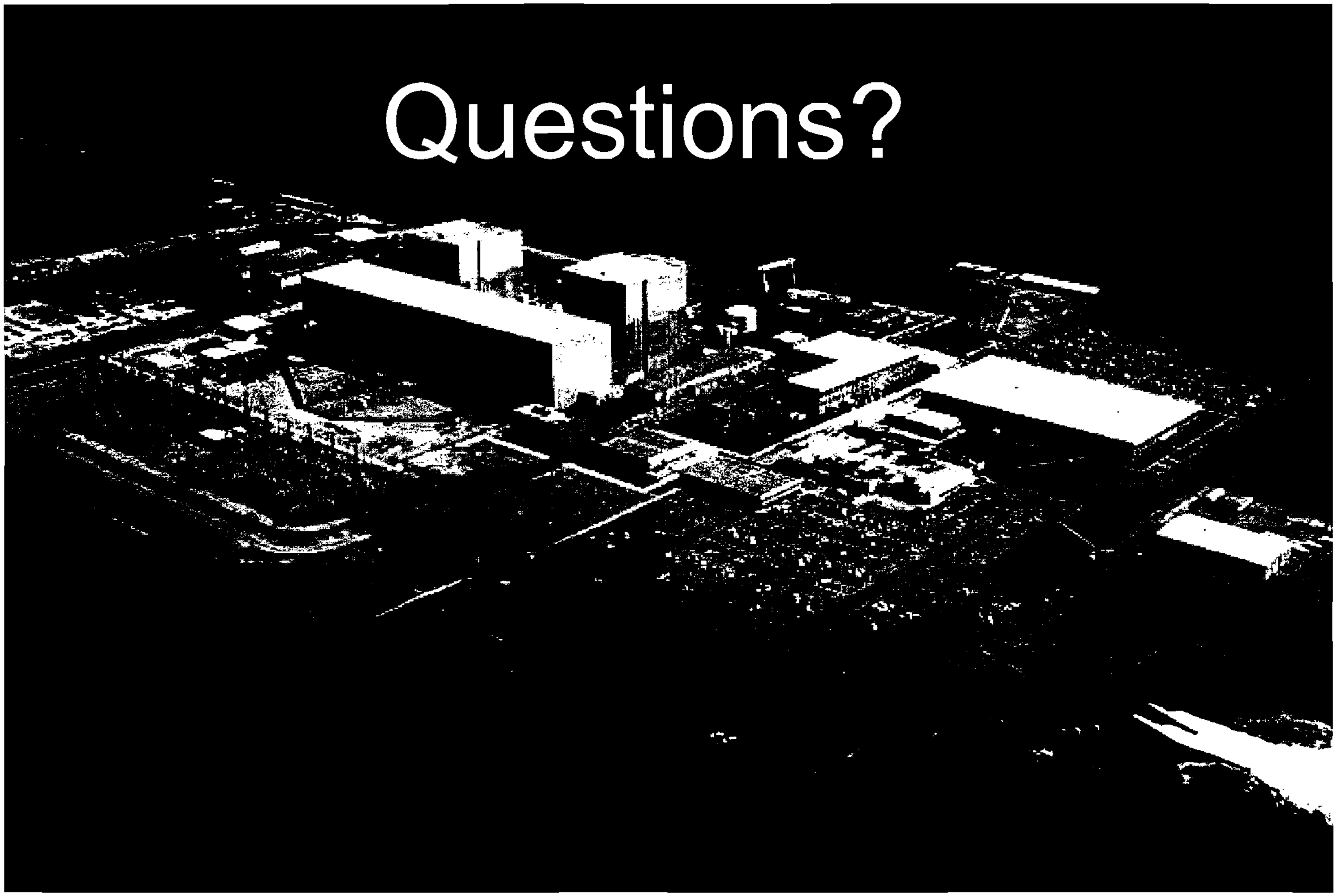
Mike Heath

Commitment Tracking

- All Commitments are Tracked by the BSEP Corrective Action Program (CAP)
- Each Commitment Has an Implementation Plan
  - Each Implementation Plan Identifies all required actions
  - All actions are linked to the CAP
  - All actions have a due date and owner
- LR Program Procedure Tracks LR Activities
- Most Document Updates Scheduled for 2006

Conclusion

- The New Audit Process Effective
- Early Identification of Concerns Allowed Early Resolution


Questions?

Brunswick Steam Electric Plant (BSEP) Units 1 and 2 License Renewal Final Safety Evaluation Report

Staff Presentation to the ACRS Full Committee

Sikhindra (SK) Mitra, Project Manager
Maurice Heath, Project Manager
Office of Nuclear Reactor Regulation

May 4, 2006

2

Introduction

- Overview
- Highlights of the Review
- Time-Limited Aging Analyses (TLAAs)
- Conclusion

Overview

- LRA submitted by letter dated October 18, 2004
- GE Boiling Water Reactors, Mark 1 design containments
- BSEP located at the mouth of Cape Fear River in Brunswick County, NC, two miles north of Southport, NC
- Unit 1 expires September 8, 2016, Unit 2 expires on December 27, 2014
- Request operating license extensions 20 years beyond the current expiration dates

Overview (continued)

- Each unit generates 2923 MW thermal, 1007 MW electrical – Include 20% Extended Power Uprate (EPU)
- Applicant committed to review plant and industry operating experience, relevant aging effects caused by operation at power uprate. The evaluation will be submitted for NRC review one year prior to period of extended operation (Commitment # 31)

Overview (continued)

- SER issued on December 20, 2005
  - No Open or Confirmatory Items
- FSER issued on March 31, 2006
  - Staff Conclusion: BSEP LRA has met the requirements of 10CFR Part 54

Highlights of Review

- Three (3) license conditions
  - FSAR update following the issuance of renewed license
  - Commitments completed in accordance with schedule
  - Reactor Vessel Surveillance Program
    - Implement Staff approved BWRVIP Integrated Surveillance Programs (ISP)
    - Obtain NRC staff review and approval for any changes to the capsule withdrawal schedule

Highlights of Review

- Items Brought into scope and subject to AMR
  - Switchyard Breakers
  - Service Water Intake structure fan, dampers, bird screen
  - Condensate Storage Tank Piping Credited for SBO

Highlights of Review

- Tier 1: Screen, Review (LRA, FSAR), Identify Systems for Inspections
- Tier 2: Review (Boundary Drawings, and Other Licensing Basis Documents in Addition to LRA, FSAR)
- 39 out of 62 Mechanical Systems are BOP (Most Auxiliary and Steam and Power Conversion Systems)
- 15 BOP Systems Selected for Tier 1 Review
- 24 BOP Systems Selected for Tier 2 Review

Highlights of Review

- Two – Tier Scoping Review Based on Screening Criteria
  - Safety Importance/Risk significance
  - Systems Susceptible to Common Cause Failure of Redundant Trains
  - Operating Experience Indicating Likely Passive Failures
  - Previous LRA Review Experience of Omissions

- 8 Total Electrical Systems and Structures Continue to Receive Tier 2 review

Highlights of Review

	Aggressive Limit	BSEP
pH	<5.5	6.4 – 7.5
Chlorides	>500 ppm	11 – 49 ppm
Sulfates	>1500 ppm	2 – 66 ppm

- Ground water phosphate level at 0.12 ppm
- Below grade environment is non-aggressive
- Annual groundwater monitoring frequency for concrete structures

Highlights of Review

- Commitment # 22 defines which BWRVIP reports are included in the scope of the Reactor Vessel and Internals Structural Integrity Program (RV&ISIP) and additional specific augmented activities that will be taken by the applicant
- Added sample size of the augmented inspection for top guide that will focus on the high fluence region

Highlights of Review

- BSEP is Mark I Steel Lined Reinforced Concrete Containment
- BSEP Credits ASME Section XI, Subsection IWE and 10 CFR Part 50, Appendix J for management of Drywell Liner
- Both IWE and Appendix J requires 100% inspection per period, there are 3 periods per interval, and each interval is ten years.

TLAA - Reactor Vessel (RV) Upper Shelf Energy (USE)

RV Beltline Component	Acceptance Criterion for USE	Component Value for 54 EFPY	Acceptable (Y/N)
Brunswick 1 Lower Intermediate Shell Plate (Heat No. B8946-1)	Percent Drop <23.5 percent drop in the USE ft-lb value	21.0 Percent Drop in USE ft-lb	Yes [TLAA satisfies 54.21(c)(1)(ii)]
Brunswick 1 Circumferential Weld FG (Heat No. 1P4218)	Percent Drop <39.0 percent drop in the USE ft-lb value	14.1 Percent Drop in USE ft-lb	Yes [TLAA satisfies 54.21(c)(1)(ii)]
Brunswick 2 Lower Shell Plate (Heat No. C4500-2)	Percent Drop <23.5 percent drop in the USE ft-lb value	17.0 Percent Drop in USE ft-lb	Yes [TLAA satisfies 54.21 (c)(1)(ii)]
Brunswick 2 Circumferential Weld FG (Heat No. S3986)	Percent Drop <39.0 percent drop in the USE ft-lb value	13.3 Percent Drop in USE ft-lb	Yes [TLAA satisfies 54.21 (c)(1)(ii)]

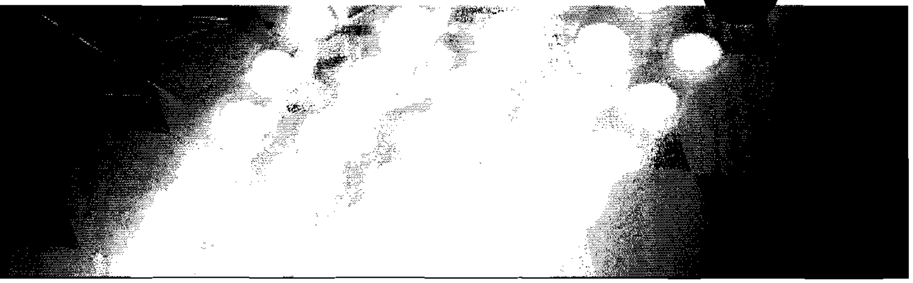
TLAA - Reactor Vessel (RV) Upper Shelf Energy (USE)

RV Beltline Component	Acceptance Criterion for USE	Component Value for 54 EFPY	Acceptable (Y/N)
Brunswick 1 and 2 N-16 Instrument Nozzle Forgings	Neutron Fluence $<1.6 \times 10^{18}$ n/cm^2 (E>1.0 MeV)	Neutron Fluence $= 1.38 \times 10^{18}$ n/cm^2 (E>1.0 MeV)	Yes [TLAA satisfies 54.21 (c)(1)(ii)]
Brunswick 1 and 2 N-16 Instrument Nozzle Welds	Percent Drop <35.0 percent drop in the USE ft-lb value	12.0 Percent Drop in USE ft-lb	Yes [TLAA satisfies 54.21 (c)(1)(ii)]

Conclusion

- On the basis of its evaluation of the license renewal application, the NRC staff concluded that the requirements of 10 CFR 54.29(a) have been met

Constellation Energy

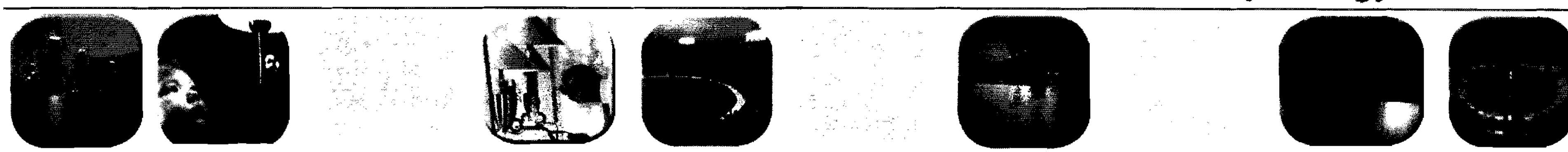


Ginna Extended Power Uprate

ACRS Full Committee Meeting

May 4, 2006

The way energy works.™



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Ginna Extended Power Uprate

Dave Holm
Ginna Plant Manager
Introduction/Agenda Review

Agenda

• Introduction	Dave Holm
• Plant Changes	Mark Finley
• Safety Analysis	Mark Finley
• Mechanical Impacts	Jim Dunne
• PRA	Rob Cavedo
• Conclusion	Dave Holm

Introduction - Agenda

- Design and Operating History
- Preparations for Uprate

Introduction – Design and History

- Westinghouse two-loop 1520 MWt NSSS design
- Commercial operation in 1970
- 1300 MWt original licensed power
- 1520 MWt licensed in 1972
- 1775 MWt Extended Power Uprate (1)

(1) Kewaunee is operating at 1772 MWt

Introduction – Preparations for Uprate

- Replaced steam generators 1996
- Replaced reactor vessel head 2003
- Experienced project team:
  Westinghouse, Stone & Webster, Siemens
- Executive oversight:
  corporate, vendor, industry experts

Ginna Extended Power Uprate

Mark Finley

Project Director

Plant Changes

Plant Changes-Agenda

- Operating Parameters
- Major Modifications
- License Amendments

Plant Changes-Operating Parameters

	EPU		Pre-EPU		
	Condition	Enthalpy	Condition	Enthalpy	Change
Core Power (MWt)	1775		1520		+16.8%
Taverage	574°F		561°F (1)		+13°F
Tcold / h cold (BTU/lb)	541°F	536.1	532°F	525.1	+9°F
Delta T	66°F		58°F		+8°F
Delta h		87.1		74.0	+17.5%
Thot / h hot (BTU/lb)	607°F	623.1	590°F	599.1	+17°F
Coolant Mass Flow (lb/hr)	6.96E+07		7.01E+07		-0.7%
Pressurizer Pressure	2250 psia		2250 psia		
SG Power (MWt)	1781		1526		+16.8%
FW In / h in (BTU/lb)	432°F	410.5	425°F	402.9	+7°F
Delta h		788.8		797.2	-1.2%
Stm Out / h out (BTU/lb)	798 psia	1199.4	770 psia	1200.1	+28 psia
Stm Mass Flow (lb/hr)	7.71E+06		6.53E+06		+18.0%

(1) Taverage was 573.5°F prior to SG replacement in 1996

Plant Changes – Major Modifications

- Fuel assembly
- Feed isolation valve actuators
- High pressure turbine and turbine control valves
- Main feedwater and booster pumps, feed regulating and bypass valves
- Cooling for main generator, step-up transformer, isophase ducts and underground oil cables
- Moisture Separator Reheater relief system
- Risk beneficial modifications: charging pump backup air, charging and TD AFW controls

Plant Changes – License Amendments

Change	EPU	Current
Core Thermal Power	1775 MWt	1520 MWt
LOCA Methods	BE LOCA/ASTRUM	BE LOCA/SECY-83-472
Axial Offset Control	RAOC (Relaxed)	CAOC (Constant)
Max Boron - Accumulator / RWST	3050 ppm	2600 ppm
Min Volume – Accumulator	1090 ft^3	1111 ft^3
Min Volume - Condensate Storage Tank	24350 gal	22500 gal
Feed Isolation Valve (Back-up Valve Stroke Time)	30 sec	60 sec
Safety Setpoints	Later in 'Safety Analysis'	Later in 'Safety Analysis'

Ginna Extended Power Uprate

Mark Finley
Project Director
Safety Analysis

Safety Analysis-Agenda

- Safety Setpoints
- Control Settings
- Methods
- Non-LOCA
- LOCA
- LTC
- Conclusion

Safety Analysis-Safety Setpoints (Analytical)

Setpoint	EPU	Current
High Flux Trip	$\leq$ 115%	$\leq$ 118%
Steam Line Isolation Hi-Hi	$\leq 5.97 \times 10^6$ lbm/hr	$\leq 3.70 \times 10^6$ lbm/hr
Steam Line Isolation Hi	$\leq 1.50 \times 10^6$ lbm/hr @ $\geq$ 530°F	$\leq 0.66 \times 10^6$ lbm/hr @ $\geq$ 543°F
Pressurizer Safety Lift Setting	$\leq$ 2542 psig	$\leq$ 2544 psig
Safety Injection	$\geq$ 1700 psig	$\geq$ 1715 psig
Containment Spray	$\leq$ 33.5 psig	$\leq$ 32.5 psig
P-8 Permissive (Single loop low flow)	$\leq$ 35%	$\leq$ 50%

Safety Analysis-Control Settings

Setting	EPU	Current
Pressurizer Level - Full Power - Zero Power	56% 20%	50% 35%
T_{Avg} - Full Power - Zero Power	574°F 547°F	561°F 547°F
Rod Control - Low Power Mismatch Gain - High	0.3 °F/% - 0.6 °F/% 1.5 °F/% - 3 °F/%	1.5 °F/% - 3 °F/% 5 °F/% - 10 °F/%
Steam Dump Modulation - Turbine Operating - Turbine Tripped	 4°F - 11°F 0°F - 11°F	 5°F - 20°F 0°F - 15°F
T_{Hot} Filter	4.5 sec	0 sec

Safety Analysis-Methods

Method	EPU	Current
Non-LOCA	RETRAN	LOFTRAN
Large Break LOCA	BE LOCA/ASTRUM	BE LOCA/SECY-83-472
Small Break LOCA	NOTRUMP	NOTRUMP
Control System Transients	LOFTRAN	LOFTRAN
Containment: LOCA MSLB	GOTHIC GOTHIC	GOTHIC COCO
Dose Assessment	AST	AST

Safety Analysis-Non-LOCA Approach

- Very conservative inputs for pre-EPU analyses used in EPU analyses where possible
- Certain limiting EPU analyses were not successful with pre-EPU inputs
- Inputs were adjusted until acceptable results demonstrated
- No attempt made to demonstrate additional margin
- Understand the conservative nature of methods, inputs and approved limits

Safety Analysis-Non-LOCA

	Event	Criteria	Result
Overheating (Reduced Primary Cooling)	Loss of Flow (Cond III) Locked Rotor (Cond IV)	DNBR $\geq$1.38 Pres $\leq$2997 psia	1.385 2782 psia
Overheating (Reduced Secondary Cooling)	Loss of Load (Cond II) (Bounds Loss of Feed) Feed Line Break (Cond IV) ATWS	Pres $\leq$2748.5 psia No T_{SAT} in HL Pres $\leq$3200 psig	2747 psia (No pzr fill) 2°F subcool 3193 psig
Overcooling	MSLB @ Power (Cond IV) (Bounds Increased FW/ARV)	DNBR $\geq$1.38 LHR $\leq$22.7 kw/ft	1.39 22.67 kw/ft
Reactivity Addition	Rod W/D @ Power (Cond II) Rod Ejection (Cond IV)	DNBR $\geq$1.38 Pres $\leq$2748.5 psia $\leq$200 cal/gm	1.381 2748.1 psia 178 cal/gm

Safety Analysis-Non-LOCA Loss of Flow DNB

CHF	1.0
Bounding Test Data- (95% probability/95% confidence)	1.17
Design Limit- accounts for parameter uncertainties (95/95)	1.24
Safety Analysis Limit- accounts for generic penalties with margin	1.38
Safety Analysis Result	1.385
Credit for Less Trip Delay	1.42
Credit for Overpressure	1.50

Safety Analysis-Non-LOCA Loss of Load Pressure

Potential Deformation- (ASME Service Level C Limit - Hot)	>3200 psig
Hydrostatic Test Pressure (Cold)	3107 psig
Design Limit- 110% of Design Pressure	2748.5 psia
Safety Analysis Result	2747 psia
Credit for Steam Dump and Pzr Spray	2605 psia
Credit for Steam Dump, Pzr Spray and PORVs	2565 psia
Credit for Reactor Trip on Turbine Trip	2348 psia

Safety Analysis-Non-LOCA

- All Non-LOCA results meet acceptance criteria
- Margin exists in the methods and the inputs
- Margin exists between the acceptance criteria and the failure point

Safety Analysis-LOCA

Results

- Large Break PCT 1870° F
- Small Break PCT 1167° F

Safety Analysis-Long Term Cooling

- The Ginna Design
  - High head safety injection (SI) pumps aligned to the RCS cold legs
  - Low head safety injection using the residual heat removal (RHR) pumps aligned to the upper plenum to provide upper plenum injection (UPI)
  - Simultaneous injection – both SI and RHR – will flush the core for all break locations, prevent boric acid concentration and assure Long Term Cooling

Safety Analysis-LTC-Large Break Analysis

- Mixing volume and void fraction calculated with Large Break LOCA code WCOBRA/TRAC
- No credit for mixing with UPI flow, no credit for beneficial effect of sump additives, no credit for containment pressure above atmospheric
- Credit for mixing with one-half lower plenum volume
- Time to reach boric acid solubility limit for atmospheric pressure is 6 hr 13 minutes
- Operators will restart SI beginning at 4.5 hours

Safety Analysis-LTC-Small Break Analysis

- Mixing volume and void fraction calculated with Small Break LOCA code NOTRUMP
- 4" break conservatively used to bound all small breaks
- Boric acid concentration is calculated as a function of time
- No credit for beneficial effect of sump additives
- Credit for mixing with one-half lower plenum volume
- Time to reach boric acid solubility limit for atmospheric pressure is 6 hr 48 minutes
- Operators will depressurize to initiate UPI, or refill to initiate natural circulation, in less than 5.5 hours

Safety Analysis-Conclusion

- All safety analyses meet acceptance criteria
- NSSS and Emergency Safety Features are robust
- Results are consistent with Kewaunee

Ginna Extended Power Uprate

Jim Dunne
Project Lead Engineer
Mechanical Impacts

Mechanical Impacts-Agenda

- Steam Generator Vibration
- BOP Heat Exchanger Vibration
- Vibration Monitoring Program
- Flow Accelerated Corrosion

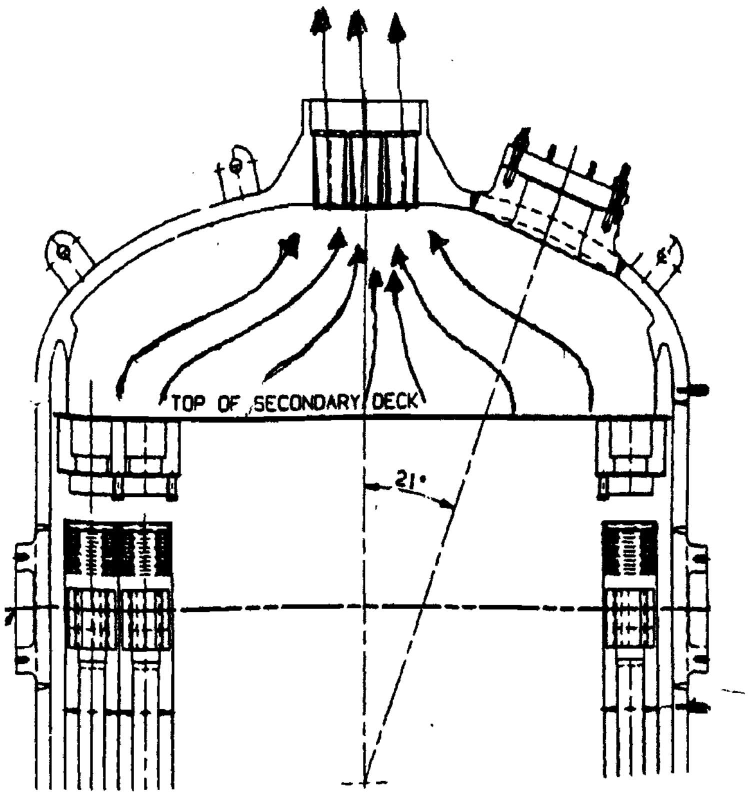
Mechanical Impacts-Steam Generator Vibration

- Steam Generator - Vibration
  - Vibration Potential in U-Bend & Tube Bundle Entrance
  - Fluidelastic Instability
  - Vortex Shedding (Tube Bundle Entrance)
  - Random Turbulence Excitation
  - Tube Wear (U-Bend Region)

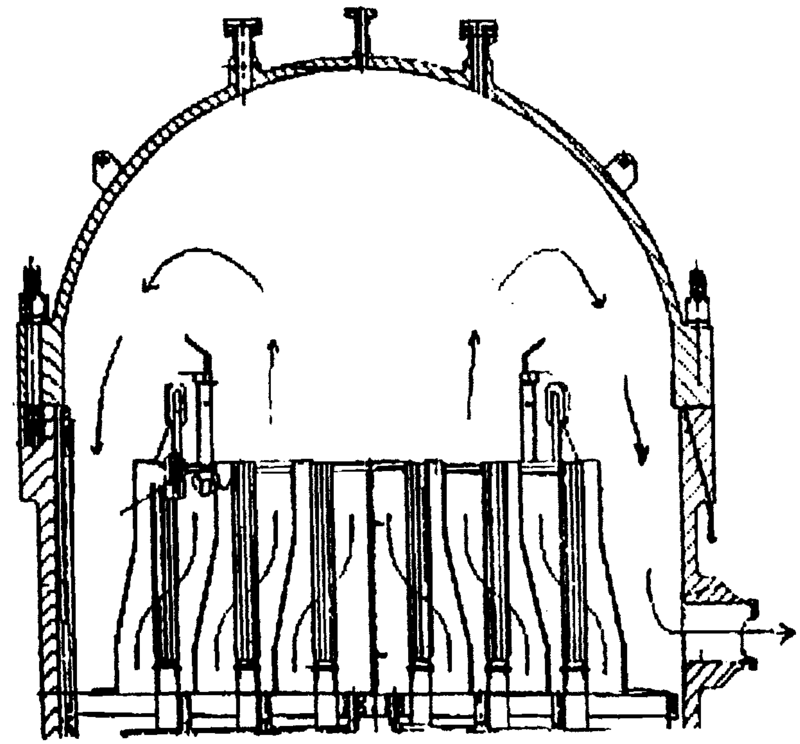
Mechanical Impacts-Steam Generator Separators

- Steam Generator Steam Separators
  - 85 Primary/Secondary Separator Modules
  - Primary & Secondary Centrifugal Type Separators
  - Minimal Cross-Flow Velocities
  - Rigid Separator Bundle
  - Full Scale Testing of Separator Modules
  - Up-rate Flow Bounded by Tested Flow Conditions

Ginna Separator / BWR Dryer Comparison



Ginna Steam Separators



BWR Steam Dryers

Mechanical Impacts-Vibration

- BOP Heat Exchangers - Vibration
  - Feedwater Heaters
  - Moisture Separator Reheaters
  - Condenser Tubing
- Vibration Monitoring Program
  - Pre-EPU Walkdown @ Full Power
  - Post EPU Walkdown (Pre- and Post-Full Power Levels)

Mechanical Impacts-Flow Accelerated Corrosion

- Flow Accelerated Corrosion (FAC)
  - Power Uprate effects evaluated using CHECWORKS
  - No component replacements required
  - Post Uprate Outage inspection sampling increased based on EPU conditions
  - Piping systems impacted will continue to be monitored to detect any deviation from predicted wear rates

Ginna Extended Power Uprate

Rob Cavedo
Risk Consultant
PRA

PRA-Agenda

- Scope
- Method
- Results
- Conclusion

PRA-Scope

- Address Impact On:
  - Initiating Event frequency
  - Success criteria
  - Equipment failure rates
  - Operator response times and Human Reliability Analysis (HRA)
- Identify Risk Beneficial Plant Changes
- Calculate the CDF and LERF Changes On:
  - Internal events
  - External events
  - Shutdown

PRA-Method

- Initiating Event Frequency
  - No new PSA initiators
  - Frequencies adjusted based on Engineering Evaluations
- Success Criteria
  - PCTRAN analyses to adjust success criteria as needed
  - Bleed-and-Feed Timing Adjusted

PRA-Method

- Equipment Failure Rates
  - Comprehensive reviews of equipment performed
  - Systems operate within allowable limits
  - No significant impact is expected to the likelihood of post-trip Equipment Failure Rates
- Operator Response Times / HRA
  - PCTRAN analyses to determine available action times
  - Higher decay heat reduced operator action times

PRA-Method

- Plant Beneficial Changes Identified and Incorporated
  - Use of high pressure SI pumps
  - Adjustment of RHR AOV
  - Addition of Back-up Air Supply for Charging Control

PRA-Results

Case	Pre or Post Uprate	CDF	LERF	Optimize SI Pump in Fire	Limit RHR AOVs	Back-Up Air to Charging
Base	Pre	6.36E-05	4.88E-06	No	No	No
Base	Post	7.12E-05	5.35E-06	No	No	No
SI	Post	6.40E-05	4.73E-06	Yes	No	No
SDAOV	Post	6.59E-05	5.32E-06	No	Yes	No
BK-IA-CHG	Post	7.10E-05	5.20E-06	No	No	Yes
SI-AOV-IC	Post	5.85E-05	4.56E-06	Yes	Yes	Yes

From EPU Submittal: Table 2.13-21

PRA-Conclusion

The Plant Risk Level Pre-EPU without the modifications is higher than the Risk Level Post-EPU with modifications

Ginna Extended Power Uprate

Dave Holm
Ginna Plant Manager
Conclusion

Conclusion

- Detailed and comprehensive reviews have been completed
- No safety issues were uncovered
- Comprehensive testing will be performed
- Ginna safety and reliability will be maintained through plant modifications, procedure changes and training

532nd Meeting of the Advisory Committee on Reactor Safeguards

NRC Staff Review of Extended Power Uprate Application
For
R.E. Ginna Nuclear Power Plant

May 4, 2006

Introduction

Patrick D. Milano
Senior Project Manager
Divsion of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Agenda -Topics

- Licensee Introduction
- Plant Modifications to Support Uprate
- Safety Analyses
- Mechanical Impact
- Probablistic Risk Assessment
- Other Evaluation Items
- Summary

Reactor Systems Analyses

- Fuel and Nuclear System System Design
- ECCS and Other Associated Systems
- Non-LOCA Transients
- LOCAs
- ATWS

Reactor Systems Review

Matrix 8 of NRC Review Standard RS-001

- **NRC Review Confirms:**
  - Use of NRC-Approved Codes and Methods for Plant-Specific Application
  - Compliance with Limitations or Conditions on Code Use
  - SG Plugging and Asymmetry Accounted in Analyses
  - Licensee's Evaluation of any Vendor Service Advisories
  - Appropriate Analytical Assumptions
  - Results Meet Applicable Requirements
  - Processes to Ensure Analyses Bound As-Operated Conditions
  - Boron Precipitation
  - Long-Term Cooling

Fuel and Nuclear Design

- **Continuity: WCAP- 9272-P-A, "Westinghouse Reload Safety Evaluation Methodology"**
- **Changing Fuel Design from OFA to 14X14 422V+**
- **Notable Differences between OFA and 422V+**
  - 14X14 422V+ Assembly Loss Coefficient is 20% less
  - VIPRE-01 replaces THINC IV Codes
  - Transition Core DNBR Penalty
- **Notable Similarities**
  - RTDP and WRB-1 DNB Correlation
  - STDP and W-3 DNB Correlation
  - DNBR Limits

Non-LOCA Transients

- **Followed the Guidelines of RS-001**
- **Most Events Analyzed with RETRAN and VIPRE**
  - Both NRC-approved
  - Not LOFTRAN and THINC
- **Important to Analyses and Evaluations:**
  - 1817 MWt (19% uprate) assumed in analyses
  - Steam generators replaced in 1996
  - License renewal in 2004 (term extended to 2029)
  - Fuel transition concurrent with EPU
  - Full-power Tavg operating window (564.6 °F to 576.0 °F)
  - Assumed up to 10% tube plugging in steam generators
- **Results Satisfied the Applicable Requirements and Design Limits of TS 2.1 (Safety Limits) for Peak CL Temperature, DNBR, and RCS Pressure**

Large-Break LOCA

- Analysis results for a double-ended guillotine break at the pump discharge
- Implemented Westinghouse Best-Estimate Large-Break LOCA Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)
- Conducted for a mixed core consisting of OFA and 422V+ fuel
- Met the acceptance criteria for ECCS performance, as specified in 10 CFR 50.46:
  - calculated peak cladding temperatures (PCTs)
  - maximum cladding oxidation (local)
  - maximum core-wide cladding oxidation

Small-Break LOCA

- Short-Term Behavior
- Within Limits of 10 CFR 50.46
- Confirmed Non-Limiting with Staff's RELAP5/MOD3 Analysis
- Post-LOCA Long-Term Cooling

Mechanical Impacts

- **Flow-Accelerated Corrosion**
  - Corrosion rates for FAC-susceptible components are determined by parameters such as temperature, flow velocity, moisture content, and component material
  - Components have been added to the program based on the potential for increased FAC rate at EPU conditions (higher temperature and velocity)
  - CHECWORKS computer models are being updated prior to implementing the EPU.
  - At EPU conditions the FAC program remains consistent with industry guidelines.

Mechanical Impacts

- **Flow-Induced Vibration**
  - Main Steam and Feedwater piping instrumented at critical locations to monitor vibration levels at current rated power and during EPU power ascension, up to the full authorized power level.
  - Vibration monitoring and collected data will be evaluated according to ASME OM3 Code
  - FIV effect on steam separator expected to increase at EPU. However, judged to be acceptable based on the design basis steam flow rate of the replacement steam generator that is bounding for EPU
  - Slight increase in FIV on the U-bend tubing, but remains within allowable limits (i.e., maximum stability ratio less than the limit of 1.0)

Mechanical Impacts

Steam Generator Dryer/Separator

- Flow rate and pressure used in testing bound EPU conditions
- Past inspections performed in operating plants not found FIV fatigue
- Integrity of rugged steam separators improved in new SG design
- Low flow velocity makes potential for loose parts to enter main steam line unlikely
- Low velocity and high stiffness reduces potential for FIV
- Capability to identify degradation of SGs through plant monitoring and outage inspections
- Filtering screen ensures collection of small parts in steam flow in unlikely event of degradation of SG internal components

Ginna EPU Risk Evaluation

- **Ginna PSA Level I covers:**
  - Internal Events, including Internal Floods
  - External Events
  - Shutdown Operations

- **Ginna PSA uses a simplified containment event tree to evaluate LERF**
  - Follows NUREG/CR-6595 for PWRs with a large dry containment

PRA Insights

- **Licensee used the Ginna EPU risk evaluation to gain insights and proposed plant modifications and operational improvements that could reduce risk**
- **5 risk and cost beneficial changes identified that would likely completely offset EPU risk increase**
  - Optimize use of safety injection pumps during fires
  - Mechanically limit RHR HCVs from failing completely open
  - Provide backup air supply to charging pumps
  - Relocate charging pump control power disconnect
  - Install local controls for the turbine-driven auxiliary feedwater pump discharge motor-operated valve

PRA Conclusion

- Licensee adequately modeled and addressed potential risk impacts of the proposed EPU
- Risks are acceptable (i.e., within RG 1.174 risk acceptance guidelines)
- Proposed EPU does not create “special circumstances”
- Licensee used its risk evaluation to identify potential changes that would offset any risk increase due to the proposed EPU

Other Key Items

- Balance-of Plant
- Operator Actions and Procedures
- Testing
- Inspection

BOP Scope of Review

- **Review per RS-001, Matrix 5**
  - Internal Hazards
  - Fission Product Control
  - Component Cooling and Decay Heat Removal
  - Balance-of-Plant Systems
  - Waste Management Systems
  - Emergency Diesel Fuel Oil Storage & Light Loads

BOP ReviewAreas of Emphasis

- **Areas Affected by Increased Decay Heat Load**
  - Spent Fuel Pool Cooling
  - Service Water System
  - Auxiliary Feedwater
- **Operational Considerations**
  - Feedwater and Condensate Systems

BOP REVIEW RESULTS

- Decay Heat Load Will Not Exceed Cooling Capability of Systems that are Relied Upon
- BOP Systems will not Pose Increased Challenges to Reactor Safety Systems
- Power Ascension and Transient Test Program Provides Adequate Assurance of BOP Performance Capability

Operator Actions and Procedural Improvements

- **Revisions to Emergency and Abnormal Operating Procedures**
  - automatic action verification steps in E-0 procedure to expedite diagnosis and plant stabilization
  - R-H.1, "Response to Loss of Secondary Heat Sink," to provide earlier initiation of SAFW System to mitigate high energy line break
  - Appendix R mitigation procedures enhanced for effectiveness of operator actions and to incorporate the physical plant changes
  - ES-1.2, "Post-LOCA Cooldown and Depressurization" to direct operators to initiate cooldown of RCS using condenser dump valves (or ADVs if condensers are unavailable) within 1 hr of SBLOCA
  - ES-1.3, "Transfer to Cold Leg Recirculation," to instruct operators to reestablish cold leg SI no later than 5.5 hours after the termination of SI in the cold leg to prevent boric acid precipitation

Operator Actions and Procedural Improvements

- **For LB LOCA and SBLOCA**
  - Operators to realign HHSI for cold leg injection within 10 minutes
  - Times were unaffected for overall operator actions, but procedure and plant modifications being made to maintain operator capability to perform actions in the established time
  - Operator training related to EOP changes to be conducted prior to EPU implementation
  - All times for operator actions affected by EPU modifications and procedure revisions to be validated using simulator and plant walk throughs prior to EPU implementation

Power Ascension and Test Program

- **SP 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," provides guidance based on Regulatory Guide 1.68 and plant specific initial test program.**
- **EPU test program**
  - includes testing sufficient to demonstrate structures, systems, and components will perform satisfactorily at the proposed power level
  - considers in part, original power ascension test program, and EPU related plant modifications
- **Manual turbine trip test at 30% EPU power to verify the plant's dynamic transient response and control system settings.**
  - pressurizer level and pressure control,
  - steam generator water level control,
  - steam dump control, and
  - rod control

Power Ascension and Test Program

Conclusion

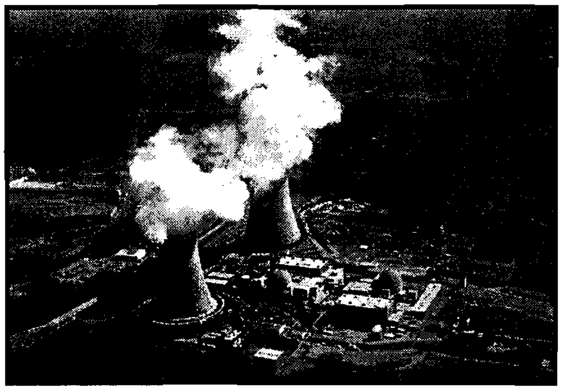
- The staff concludes that the proposed test program provides adequeate assurance that the plant will operate in accordance with its design criteria and that SSCs affected by the proposed EPU will perform satisfactorily in service.

NRC Inspection

- **Conducted by Resident Staff and Regional Specialists**
- **Inspection Procedure 71004, "Power Uprates"**
  - Describes inspections necessary for power uprate related activities
  - Provides guidance in conducting these inspections
- **Recommended Areas for Inspection**
  - Consider recommendations listed in final safety evaluation when selecting a sample for implementing IP 71004
  - These recommendations do not constitute inspection requirements
  - Provided to give the inspectors insight into important bases the NRC staff used for approving the EPU
  - Examples

(5)

BEAVER VALLEY POWER STATION
Extended Power Uprate



ACRS

Full Committee Meeting

May 4, 2006

FENOC

Introduction & Overview

Pete Sena
Director, Site Engineering

FENOC

Agenda

- Introduction
- Plant Changes
- Safety Analysis
- Mechanical Impacts
- PRA
- Conclusion

- Pete Sena
- Mark Manoleras
- Ken Frederick
- Mike Testa
- Colin Keller
- Pete Sena

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Introduction - Agenda

- Beaver Valley History
- Beaver Valley Peer Units
- Preparations for Uprate

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Beaver Valley History

- Beaver Valley Power Station Units 1 and 2
- Westinghouse NSSS 3 loop Pressurized Water Reactor (PWR)
- BV-1 Commercial Operation - 1976
- BV-2 Commercial Operation - 1987
- 2652 MWt original licensed Rated Thermal Power (RTP)
- 2689 MWt Appendix K Margin Recovery - 2001
- 2900 MWt Extended Power Uprate (EPU) - pending

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Plant	Uprated NSSS Power Level (MWt)
Beaver Valley Units 1 & 2	**2910**
North Anna Units 1 & 2	**2905**
V. C. Summer	**2912**
Shearon Harris	**2912**
Vandellos	**2954**
ASCO Units 1 & 2	**2952**

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Preparations for Uprate

To Position BVPS Units for EPU:

Supporting Submittals Completed:

- **New Fuel Storage Rack Enrichment Limit Increase**
- **Positive Moderator Temperature Coefficient**
- **Accumulator and RWST Increased Boron Concentration**
- **Selective implementation of AST**
- **Minimum Decay Time Before Fuel Movement**
- **Relaxed Axial Offset Control (RAOC)**

Replacement Steam Generators (RSG) BVPS-1

Containment Conversion

Large Break Best Estimate Loss-of-Coolant Accident (BELOCA) Methodology

Extended Power Uprate (EPU) - Pending

FENOC

WHAT'S NEW *inside BV Unit 1 Containment*

Take a look below at the improvements made inside Unit 1 Containment during Beaver Valley's 1R17 outage.

1) Three brand new 368-ton steam generators were installed. Shown here is the top portion of Steam Generator 'B.'
2) New catwalks on the steam generators will eliminate the need to build scaffolding, saving time and dose.
3) A new, simplified Reactor Vessel Head will save 12 polar crane lifts per outage in the future.
4) A new configuration for the Control Rod Drive Mechanism ventilation was installed.
5) The Cable Bridge will allow easier access to the head and will simplify the process of disconnecting the Control Rod Drive Mechanisms and Rod Position Indicators in future refueling outages.
6) New mirror insulation was installed on all three Steam Generators and the Reactor Vessel Head. The insulation will help keep heat inside the steam generators and debris out of the containment sump.
7) Steam Generator 'A.' Steam Generator 'C' is not shown.
8) Close to 1.4 miles of welds on the Reactor Coolant System, Main Steam System, Feedwater Lines and instrument tubing were completed.

Project Team and Oversight

- FENOC / BVPS
  - Overall project management
  - Review and approval of inputs
  - Proper interfacing of Information
  - Procedure / Training / Simulator updates
- Westinghouse, Stone & Webster, Siemens
- Oversight of the engineering and licensing process

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Plant Changes

Mark Manoleras
(Manager, Design Engineering)

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Major Modifications

- Replacement of charging/safety injection pump rotating assemblies
- Conversion from a sub-atmospheric to an atmospheric containment design
  - Installation of fast acting feedwater isolation valves (Unit 1)
  - Installation of auxiliary feedwater cavitating venturies (Unit 1)
  - Addition of reactor cavity drainage port
- Replacement of Steam Generators (Unit 1)

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Major Modifications

- Replace high pressure turbine with all-reaction design
- Install stakes in main condenser (Unit 2)
- Raise set-pressure of moisture separator reheater relief valves
- Increase Cv of main feedwater control valves
- Replace Turbine Generator (T/G) rotor and rewind stator (Unit 1)
- Instrument replacements for higher flow range

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Safety Analysis

Ken Frederick
(Nuclear Safety Analyst)

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Safety Analysis Objectives

- Demonstrate compliance with regulatory limits and acceptance criteria
- To show that BVPS will operate with adequate safety margins at EPU conditions

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Safety Analysis - Agenda

- EPU Operating Parameters
- Methods
- Non-LOCA Events
- LBLOCA
- SBLOCA
- Post LOCA Long Term Cooling
- Containment

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[illegible]

	EPU Condition	Pre-EPU Condition	Change
Core Power (MWt)	2900	2689	+7.9%
Taverage (F)	577.9	576.2	+1.7F
Tcold (F)	544.6	545.1	-0.5F
Delta T (F)	66.6	62.2	+4.4F
Thot (F)	611.2	607.3	+3.9F
Coolant Mass Flow (total lb/hr)	1.11E+08	1.11E+08	0%
Pressurizer Pressure (psia)	2250	2250	0 psi
SG Power (total MWt)	2910	2697	+7.9%
FW In (F)	440	434.3	+5.7F
Stm Out (psia)	805	825	-20 psi
Stm Mass Flow (total lb/hr)	1.27E+07	1.17E+07	+8.5%

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[redacted]

	EPU Condition	Pre-EPU Condition	Change
Core Power (MWt)	2900	2689	+7.9%
Taverage (F)	574.2	576.2	-2F
Tcold (F)	538.9	543.4	-4.5F
Delta T (F)	70.6	65.6	+5F
Thot (F)	609.5	609	+0.5F
Coolant Mass Flow (total lb/hr)	1.05E+08	1.05E+08	0%
Pressurizer Pressure (psia)	2250	2250	0 psi
SG Power (total MWt)	2910	2697	+7.9%
FW In (F)	437	434	+3F
Stm Out (psia)	774	821	-47 psi
Stm Mass Flow (total lb/hr)	1.27E+07	1.17E+07	+8.5%

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Safety Analysis Methods

Method	EPU	Current
Large Break LOCA	BELOCA/WCOBRA-TRAC	BASH (App K)
Small Break LOCA	NOTRUMP	NOTRUMP
Non-LOCA	LOFTRAN VIPRE	LOFTRAN THINC
Control System Transients	LOFTRAN	LOFTRAN
Containment	MAAP-DBA	MAAP-DBA (LOCTIC pre-CC)
Dose Assessment	AST/ARCON 96	TID/RAMSDELL

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Non-LOCA Acceptance Criteria

- Most Non-LOCA events are categorized as ANS Condition II for which the acceptance criteria are:
  - The critical heat flux is not exceeded (the calculated minimum DNBR does not go below the limit value at any time during the transient)
  - Peak heat generation rate remains within acceptable limits to prevent fuel centerline melt
  - Pressure in the RCS and main steam systems should be maintained below 110% of the design pressures
  - The event should not generate a more serious plant condition without other faults occurring independently

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Non-LOCA DNBR Margin

DNBR		Corresponding Limit
1.55		Safety Analysis Limit
	Retained Margin	
1.22		Design Limit
	RTDP Uncertainties	
1.14		Correlation Limit (TS)
	Correlation Uncertainty	
1.0		Critical Heat Flux

WRB-2M DNBR LIMITS

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Non-LOCA DNBR Results

DNBR Limited Events				
Event	DNBR Correlation	DNBR Limit	BVPS-1 DNBR	BVPS-2 DNBR
RCCA Bank Withdrawal from Subcritical	W-3,WRB-1	1.65, 1.45	1.83, 2.12	1.83, 2.12
RCCA Bank Withdrawal at Power	WRB-2M	1.55	1.57	1.58
RCCA Misalignment	WRB-2M	1.55	(1)	(1)
Loss of Load	WRB-2M	1.55	2.23	1.83
Feedwater System Malfunctions a. Feedwater Flow Increase b. Feedwater Enthalpy Decrease	WRB-2M WRB-2M	1.55 1.55	1.75 1.67	1.96 1.66
RCS Depressurization	WRB-2M	1.55	1.62	1.64
Main Steam Pipe Rupture (HFP)(2)	WRB-2M	1.55	2.56	2.56
Main Steam Pipe Rupture (HZP)(2)	W-3	1.61	2.41	1.83
Partial Loss of Flow	WRB-2M	1.55	2.25	2.25
Complete Loss of Flow	WRB-2M	1.55	1.64	1.64

(1) No DNBR Results-Analysis uses peaking factor limits for evaluation
(2) Condition IV event evaluated with Condition II limits

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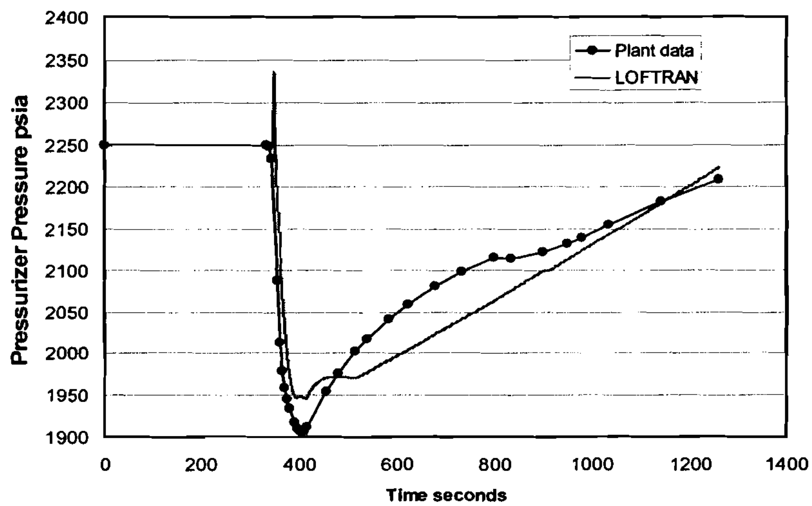
Non-LOCA Pressure Results

Limiting Overpressure Events						
Event	**Primary Pressure Limit (Psia)**	**BVPS-1 Peak Primary Pressure (Psia)**	**BVPS-2 Peak Primary Pressure (Psia)**	**Secondary Pressure Limit (Psia)**	**BVPS-1 Peak Secondary Pressure (Psia)**	**BVPS-2 Peak Secondary Pressure (Psia)**
Loss of Load	**2748.5**	**2747**	**2746**	**1208.5**	**1192**	**1191**
Feedwater System Malfunctions	**2748.5**	**2357**	**2353**	**1208.5**	**1124**	**1141**
Partial Loss of RCS Flow	**2748.5**	**2374**	**2361**	**1208.5**	**989**	**995**
Complete Loss of RCS Flow	**2748.5**	**2504**	**2503**	**1208.5**	**993**	**1003**
Locked Rotor	**2997**	**2797**	**2825**	-	-	-
ATWS	**3215**	**3060**	**2900**	-	-	-

FENOC

BVPS-2 Rx Trip on MUG Trip 4/2/2006



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Non-LOCA Other Results

Pressurizer Filling Events			
Event	**Pressurizer Water Volume Limit (ft^3)**	**BVPS-1 Peak Pressurizer Water Volume (ft^3)**	**BVPS-2 Peak Pressurizer Water Volume (ft^3)**
Loss of Normal Feedwater	**1458**	**1384**	**1193**
Loss of AC	**1458**	**1224**	**1194**
Spurious Safety Injection	**1458**	**Pressurizer Fills**	**Pressurizer Fills**
Margin to Hot Leg Saturation Event			
Event	**Margin to Hot Leg Boiling Limit (°F)**	**BVPS-1 Margin to Hot Leg Boiling (°F)**	**BVPS-2 Margin to Hot Leg Boiling (°F)**
Feedline Break	**0 (No boiling)**	**14.4**	**36**
Maximum Fuel Stored Energy Event			
Event	**Max Fuel Stored Energy Limit (Btu/Lbm)**	**BVPS-1 Max Fuel Stored Energy (Btu/ Lbm)**	**BVPS-1 Max Fuel Stored Energy (Btu/ Lbm)**
RCCA Ejection	**360**	**326.8**	**326.8**

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Non-LOCA Conclusions

- DNBR limits contain margin between safety analysis limits design limits to allow for core design flexibility
- Conservatism in peak pressure limits and analysis inputs allow for maintaining margins in operating limits
- All acceptance criteria for Condition II,III,IV Non-LOCA events are met at EPU conditions

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LOCA - Results

- PCT Results meet 10CFR50.46 acceptance criteria

Parameter	Current	EPU	Limit
Unit 1 Large Break PCT	**1996 °F**	**2021 °F**	**<2200 °F**
Unit 2 Large Break PCT	**1908 °F**	**1976 °F**	**<2200 °F**
Unit 1 Small Break PCT	**1902 °F**	**1895 °F**	**<2200 °F**
Unit 2 Small Break PCT	**1902 °F**	**1917 °F**	**<2200 °F**

- Oxidation results meet 10CFR50.46 acceptance criteria including consideration of pre-transient oxidation

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Long Term Cooling - Analysis

- Core voiding considered by reducing the mixing volume accordingly
- Time-based Mixing Volume / System Effects considered
- Effect of sump additives on Boric Acid solubility limit quantified but not credited
- Appendix K decay heat was used in all calculations

Long Term Cooling Summary

- Post LOCA long term core cooling has been adequately addressed
- Results show the following for switchover time to hot leg injection:
  - BVPS-1 - 6.5 hours (8 hours pre-EPU)
  - BVPS-2 - 6 hours (7 hours pre-EPU)
- For small breaks, cooldown and depressurization can be accomplished within required switchover time

Containment Analysis

- Containment will operate at slightly sub-atmospheric conditions
  - Prior to containment conversion 9 psia to 10.5 psia (air partial pressure)
  - Following containment conversion 12.8 psia to 14.2 psia
- Analysis credits plant modifications
  - Replacement Steam Generators (BVPS-1)
  - New feedwater isolation valves (BVPS-1)
  - AFW cavitating venturis (BVPS-1)
  - Reactor cavity drainage port
  - Lowered RWST level setpoint for transfer to SI recirculation
- Peak Containment pressures and temperatures within design for all accidents
- Containment Overpressure continues to be credited for BVPS-1

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Safety Analysis Conclusions

- All applicable acceptance criteria are met at EPU conditions
- Beneficial plant modifications have been made to maintain safety margins at EPU conditions

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Mechanical Impacts

Mike Testa
(EPU Project Manager)

FENOC

Mechanical Impacts – Agenda

- Steam Generator Vibration
- Piping and Component Vibration
- Flow Accelerated Corrosion

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Tube Bundle Region

- Unit 1 – Model 54F
  - Steam Generator installed in 1R17 (April 2006)
  - Designed for uprated conditions
- Unit 2 – Series 51M
  - Review for Flow Induced Vibration (FIV) affects showed acceptable results
  - Unsupported U-bends reviewed for increased fatigue
  - Increase in tube wear at Anti-Vibration Bar (AVB) interface evaluated

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Steam Dryer FIV Comparison

- Series 51/51M
  - Low Flow Rates Near Dryer vs BWR
    - Pre-Uprate – 3.5 ft/sec
    - Post Uprate – 4.1 ft/sec
    - *BWR ~ 100 ft/sec*
  - Low Turbulence Potential Vs. BWR
  - No Operational Issues Reported
    - 22 Domestic Plants
    - 74 Domestic SG
    - Operational from early 70's
- Series 54F
  - Low Flow Rates Near Dryer vs BWR
    - Pre-Uprate – 3.0 ft/sec
    - Post Uprate – 3.5 ft/sec
    - *BWR ~ 100 ft/sec*
  - Low Turbulence Potential Vs. BWR
  - No Operational Issues Reported
    - 6 Domestic Plants
    - 18 Domestic SG
    - Operational from mid 90's

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BOP Heat Exchanger Vibration

- Feedwater Heaters
- Moisture Separator Reheaters
- Condenser Tubing
  - BVPS-1 condenser tubes previously staked
  - BVPS-2 will be staked prior to power uprate

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Vibration Monitoring

- Monitor Secondary systems pre EPU
  - Baseline walk downs conducted on each plant
  - Areas of interest targeted for inspection under EPU
- Utilize guidance from ASME OM-S/G-2003, Part 3
- Collect and review data at each power escalation plateau
- Inspections will be augmented as required with vibration monitoring equipment
- Large equipment (e.g. Reactor Coolant Pump, Turbine) consistently monitored with existing plant instrumentation

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Flow Accelerated Corrosion

- EPU effects evaluated using CHECWORKS
- Turbine extraction steam tee proactively replaced
- Post Uprate Outage inspection sampling increased based on EPU conditions
- Piping systems impacted will continue to be monitored to detect any deviation from predicted wear rates

PRA

Colin Keller
Supervisor, PRA

Probabilistic Risk Assessment

- Scope of Assessment
  - PRA Model Elements
    - Initiating Event Frequency
    - Success Criteria
    - Equipment Failure Rates
    - Operator Response Times
  - Changes in CDF & LERF for each model

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PRA – Model Elements

- Initiating Events
  - No new initiators
  - No significant increase in Initiating Event frequencies due to the Power Uprate
- Success Criteria
  - MAAP analyses establishes EPU success criteria
  - No new accident sequences identified

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PRA – Model Elements

- Component and System Reliability
  - Comprehensive reviews of equipment performed
  - Systems operate within allowable limits
  - No impact on PRA failure rates or results
- Operator Response Times / HRA
  - MAAP analyses to determine operator action time available
  - Higher decay heat reduced times for some operator actions

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Summary of Changes (Unit 1)

BVPS-1 Risk Measures	Pre-EPU Model	Post-EPU Model	Change in Risk
Total CDF (/year)	2.25 E-05	2.29E-05	3.36E-07
Internal CDF (/year)	6.25 E-06	6.55 E-06	2.97 E-07
External CDF (/year)	1.63 E-05	1.63 E-05	3.95 E-08
Fire CDF (/year)	4.62 E-06	4.66 E-06	3.89 E-08
Total LERF (/year)	4.37 E-07	4.95 E-07	5.83 E-08

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Summary of Changes (Unit-2)

BVPS-2 Risk Measures	Pre-EPU Model	Post-EPU Model	Change in Risk
Total CDF (/year)	3.30 E-05	3.33 E-05	3.55 E-07
Internal CDF (/year)	1.86 E-05	1.89 E-05	2.92 E-07
External CDF (/year)	1.44 E-05	1.45 E-05	6.32 E-08
Fire CDF (/year)	4.89 E-06	4.95 E-06	6.38 E-08
Total LERF (/year)	1.03 E-06	1.07 E-06	4.61 E-08

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PRA Conclusion

- All PRA model elements reviewed for impact
- The increase in risk, due to the EPU for BVPS-1 and BVPS-2 is small compared to the current overall risk

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Concluding Remarks

Pete Sena
Director, Site Engineering

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Conclusion

- Detailed and comprehensive reviews have been performed
- No safety issues identified
- Beaver Valley Power Station safety and reliability will be maintained through plant modifications, procedure changes and training, and adherence to TS / Operating License

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End of Presentation

6

532nd Meeting of the Advisory Committee on Reactor Safeguards

NRC Staff Review of Extended Power Uprate Application
For
Beaver Valley Power Station, Unit Nos. 1 and 2

May 4, 2006

Introduction

Timothy G. Colburn
Senior Project Manager
Divsion of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Agenda - Topics

- Licensee Introduction
- Plant Modifications to Support the EPU
- Safety Analyses
- Mechanical Impacts - FIV, FAC
- Probabilistic Risk Assessment
- Implementation
- Summary

Introduction

- Pre-application Submittals Included
  - Containment conversions to atmospheric
    - Approval of MAAP-DBA for M/E release
    - BVPS-1 relies on COP, BVPS-2 does not
    - Staff performed independent M/E release calculations
  - SG Replacement (BVPS-1 only)
- October 4, 2004 application with numerous supplements -Included full AST implementation
- Staff Review Followed RS-001, Revision 0

Reactor Systems Analyses

- Fuel and Nuclear System Design (No Changes)
- Non-LOCA Analyses and Transients
- LOCA Analyses
- ATWS
- ECCS
- Boron Precipitation
- Long Term Cooling

Reactor Systems Review

- Staff Review Using Matrix 8 of RS-001
  - No changes from NRC-approved Codes and methodologies
  - No changes to fuel design - No DNBR transition penalty
  - Uncertainties applied to initial conditions in conservative manner and conservative analyses metheods and transient assumptions were used
  - All applicable acceptance criteria were met
  - There are acceptable margins in the safety analyses limits and the safety analyses results

Reactor Systems Review (cont.)

- Staff review looked at ECCS
  - Approach to control boron precipitation
- Large-break LOCA
  - Post-LOCA long term cooling (boron precipitation)
- Small-break LOCA
  - Short term behavior
  - Post-LOCA long term cooling (boron precipitation)
- Staff conducted independent analyses and audits of Westinghouse calculations

Non-LOCA Transients

- Followed the guidelines of RS-001
- Events analyzed with LOFTRAN and VIPRE
- Analyses considerations
  - 2917.4 MWt assumed in the analyses
  - BVPS-1 steam generators replaced spring 2006
  - Licensee qualified PZR safet valves for water relief during inadvertent SI actuation
- Results satisfied applicable acceptance criteria for peak clad temperature, DNBR, and RCS pressure

Large Break LOCA Analyses

- BELOCA methodology w/COBRA-TRAC
- Cold leg break limiting for boron precipitation
- Initiate simultaneous injection before boron precipitation occurs
- Increased minimum accumulator pressure and containment operating pressure partially offset increase in power
- Met 10 CFR 50.46 acceptance criteria for ECCS performance (PCT and cladding oxidation)

Small Break LOCA

- Analyses modeled using NOTRUMP
- Initial even-integer break size analysis expanded to include broader spectrum
- Initial model assumed broken loop seal clears for all SBLOCAs -licensee reanalyzed to assume only for certain SBLOCAs do loop seals clear
- Licensee increased accumulator pressure and SI injection flow to gain margin
- Staff independent calculations agree with licensee results - short term SBLOCA analyses and SBLOCA and LBLOCA long term cooling analyses meet 10 CFR 50.46 criteria
  - Identified need for EOP changes
  - Confirmed timing for boron precipitation

Mechanical Impacts

Flow - Induced Vibration

- MS and FW piping instrumented at critical locations and collected data are evaluated to ASME OM3
- FIV on steam separator typically increases at EPU conditions. FIV on steam separators is minimized due to its high stiffness and low flow velocity
- FIV on the U-bend tubing is within allowable limits (i.e. fluid-elastic instability ratio less than 1.0 and peak stresses less than the material endurance limit)
- The potential for FIV is not increased for the steam separators and SG tubes at EPU conditions

Flow Accelerated Corrosion (FAC)

- EPU conditions change the temperature, flow velocity, and moisture content for some components.
- Updated CHECWORKS computer models will determine future inspection and repair/replacement plans.
- The FAC program scoping criteria are consistent with industry guidelines (temperature, moisture content, component alloy content, amount of usage) at EPU conditions.

Scope of Risk Evaluation

- Full-power PRA model
  - Internal events, including internal flooding
  - Seismic
  - Internal fires
  - CDF and LERF
- Qualitative approach for other risk
  - High winds, external floods, other external events-screening per NUREG-1407
  - Shutdown risk-questions in SRP Chapter 19

NRC Staff Review of EPU Risk

- NRC onsite audit (10/05) to check quality of PRA and EPU risk assessment
- Minor impact on success criteria
  - Time to recover offsite power
  - AFW flow for ATWS (cavitating venturis)
  - Containment accident pressure credit for NPSH
- Less time available for some operator actions
  - Post-EPU CDF and LERF-MAAP timing
  - Validated important, short time available actions
  - HRA sensitivity analysis
- Important operator actions with short time available
  - Depressurize RCS
  - Implement feed and bleed cooling

PRA Conclusion

- Licensee assessed potential risk impacts of the EPU
  - CDF/change in CDF-very small
  - LERF/change in LERF-very small
- The EPU does not create special circumstances that rebut the presumption of adequate protection afforded by the licensee meeting current regulations
- Risks of BVPS EPU implementation were adequately addressed by the licensee and are acceptable

EPU Implementation

- Licensee will perform 2-phase implementation of EPU for both units
  - BVPS-1 will increase power 3 percent for the remainder of this operating cycle and will implement the remainder of the EPU increase next cycle (all BOP mods are currently complete)
  - BVPS-2 will increase power by 3 percent during the next operating cycle (following the fall 2006 RFO) and will implement the remainder of the EPU increase following all-reaction HP turbine mod (spring 2008 RFO)

Summary

- The staff reviewed the licensee's proposed EPU against the criteria in NRC Review Standard RS-001
- The licensee supplemented the application numerous times in response to staff requests for additional information-including providing revised analyses, additional commitments, and changes to the application
- Staff audits helped expedite reviews
- The licensee met all applicable review criteria of RS-001 for the uprated conditions

7

PART 52 RULEMAKING

Jerry Wilson, Senior Policy Analyst
Nanette Gilles, Senior Project Manager
Division of New Reactor Licensing, NRR

ACRS Full Committee
May 4, 2006

Proposed Part 52 Rule

- Proposed rule published in *Federal Register* on March 13, 2006 (71 FR 12781)
- Supersedes proposed rule published on July 3, 2003 (68 FR 40026)
- Revised proposal result of comments on 2003 rule and lessons learned

General Overview

- Rewritten Part 52 contains five subparts:
  - Early site permits (ESPs)
  - Standard design certifications
  - Combined licenses (COLs)
  - Standard design approvals
  - Manufacturing licenses
- Appendices A-D are design certification rules
- Standardized organization and content of each subpart
- Made conforming changes throughout 10 CFR
- Generally kept technical requirements in Parts 50, 100, etc., and put procedural requirements in Part 52

Rule Objectives

- Revised rule will enhance the NRC's effectiveness and efficiency in implementing the Part 52 licensing processes
- Revised rule will provide clarity regarding the applicability of technical and procedural requirements to each of the Part 52 regulatory processes

Key Rule Proposals Affecting Safety Requirements

- Emergency Planning
  - Mitigation measures for significant impediments
  - ITAAC required with complete plans or major features at ESP stage
  - Updated emergency preparedness information at the COL stage
- Quality assurance requirements for ESP applicants
- Applicability of 10 CFR Part 21 to ESPs and design certifications
- PRA requirements for COLs

(8)

NRC Staff's Response to ACRS Comments on the Draft Final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants"

Advisory Committee on Reactor Safeguards Meeting
May 5, 2006

George Tartal, I&C Engineer
Instrumentation and Electrical Engineering Branch
Division of Fuel, Engineering and Radiological Research
Office of Nuclear Regulatory Research
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OVERVIEW

- MARCH 10, 2006 MEETING
- PREVIOUS REGULATORY POSITION 1
- ACRS LETTER TO EDO DATED MARCH 28, 2006
- STAFF RESOLUTION OF ACRS COMMENTS
- REVISED REGULATORY POSITION 1
- CONCLUSION

MARCH 10, 2006 MEETING

- On March 10, 2006 RES staff presented draft final Regulatory Guide 1.97 Revision 4 to the ACRS
- The ACRS focused their comments and discussion on Regulatory Positions 1 and 4

- The staff concluded that the ACRS agreed with Regulatory Position 4, but had residual concerns with Regulatory Position 1.

PREVIOUS REGULATORY POSITION 1

- The December 2005 version of Regulatory Position 1 stated:

- Regulatory Guide 1.97 Rev. 4 is primarily intended for licensees of new nuclear power plants, and licensees of current operating reactors may voluntarily convert to the criteria in Rev. 4
- “Conversion” refers to adapting the plant’s entire accident monitoring program from Rev. 3 (its current licensing basis) to Rev. 4.

PREVIOUS REGULATORY POSITION 1 (cont.)

- Conversion could involve physical modifications and licensing basis changes, which could result in significant cost implications
    - Rev. 4 assigns design and qualification criteria by variable type
    - Rev. 4 variables selected based on EOPs, AOPs, etc.
- Partial conversions were not recommended due to potential for loss of variables or interaction with other variables without a complete analysis

ACRS LETTER TO EDO DATED MARCH 28, 2006

- Three conclusions / recommendations:

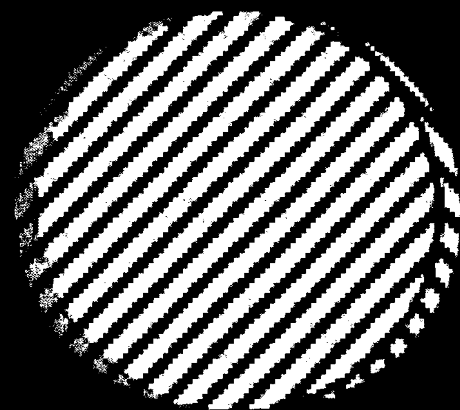
- "The staff has adopted a position that could frustrate the application of this Standard to modifying and upgrading portions of the AMI in existing plants."

STAFF RESOLUTION OF ACRS COMMENTS

- The staff agrees that a more flexible regulatory position should be provided to accommodate modifications.
- For modifications, an analysis should first be performed based on Rev. 4 selection criteria
- The analysis performed as a technical basis for modifications is a subset of the analysis performed for a conversion
- This analysis will produce the list of variables to be monitored and the assigned variable type

STAFF RESOLUTION OF ACRS COMMENTS (cont.)

- After analysis, Rev. 3 and Rev. 4 may compare:



STAFF RESOLUTION OF ACRS COMMENTS (cont.)

- Once the analysis has been performed, an evaluation can be done on a Rev. 4-based modification

- Examples of why the modification analysis is important...

STAFF RESOLUTION OF ACRS COMMENTS (cont.)

- Examples of why the modification analysis is important
  - Rev. 3 recommends SRV position as the key variable for monitoring main steam system status; classified as Type D Cat. 2
  - BWROG provided information to justify RPV pressure and suppression pool water temperature as the key variables
  - Proposed downgrade of SRV position from Cat. 2 to Cat. 3
  - Under Rev. 4, the key variables would be classified as type D and SRV position may be removed from the program

  - Rev. 3 recommends CST level as the key variable for monitoring aux feedwater system status; classified as Type D Cat. 1
  - WOG provided information to justify aux feedwater flow as the key variable and reclassify it as Type B Cat. 1; downgrade of CST level
  - Under Rev. 4, the key variable would be classified as type B and CST level may become type B or D or removed from the program

REVISED REGULATORY POSITION 1

- DELETE: “Partial conversions (i.e., conversions only performed on particular variables or systems) are not recommended because of the potential for loss of variables or interactions with other variables without a complete analysis in accordance with this guide.”
- ADD: “If a current operating reactor licensee voluntarily uses the criteria in Revision 4 of this guide for performing modifications that do not involve a conversion, the licensee should first perform an analysis to determine the complete list of accident monitoring variables and their associated types in accordance with the selection criteria in Revision 4.”

CONCLUSION

- Regulatory Guide 1.97, Rev. 4 endorses current IEEE Standard 497-2002 with exceptions and clarifications
- Intended for new nuclear plants
- Current operating plants can voluntarily convert to Rev. 4
- Current operating plants can voluntarily use Rev. 4 as a basis for modifications, and should first perform an analysis to determine the variable list and their associated variable types based on Rev. 4 selection criteria

10

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ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
532nd	10	10.1

Title: PLANNING & PROCEDURES/ FUTURE ACRS ACTIVITIES

Authors:
JOHN T. LARKINS

List of Documents Attached

PLANNING & PROCEDURES MINUTES

10

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

From Staff Person
JOHN T. LARKINS

May 4, 2006 (4:50pm)
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INTERNAL USE ONLY

SUMMARY/MINUTES OF THE
ACRS PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
May 5, 2006

The ACRS Subcommittee on Planning and Procedures held a meeting on May 3, 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 2:30 p.m. and adjourned at 3:30 p.m.

ATTENDEES
G. Wallis
W. Shack
J. Sieber

ACRS STAFF
J. T. Larkins
S. Duraiswamy
H. Nourbakhsh
M. Afshar-Tous
R. Caruso
J. Flack
E. Thornsbury
M. Junge
D. Fischer
M. Snodderly
A. Thadani
R. Savio
S. Meador

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

Member assignments and priorities for ACRS reports and letters for the May ACRS meeting are attached (pp. 8-9). 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 May ACRS meeting be as shown in the attachment (pp. 8-9).

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through July 2006 is attached (pp. 10). The objectives are to:

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

During this session, the Subcommittee also discussed and developed recommendations on items requiring Committee action (pp. 11-12).

RECOMMENDATION

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

3) Candidates to Fill the Vacancy on the Committee

The ACRS Member Candidate Screening Panel and the members interviewed several candidates for membership on the ACRS on March 8-9, 2006. Another candidate was interviewed by the Panel and several members on April 26, 2006. The ACRS Chairman provided the members' views to the Panel and the Panel will send a slate of candidates to the Commission in the near future, recommending that the Commission appoint three new members to the ACRS.

RECOMMENDATION

The Subcommittee recommends that the ACRS Executive Director keep the Committee informed of further developments.

4) Staff Requirements Memorandum (SRM Related to ACRS Request for Additional Resources to Handle Anticipated Increased Workload

In the December 20, 2005 SRM, resulting from the ACRS meeting with the NRC commissioners on December 8, 2005, the Commission stated that:

> "Following its retreat in January 2006, the ACRS should inform the Commission how the Committee plans to manage the increased workload resulting from anticipated receipt of new reactor designs and combined license (COL) applications"

The Committee responded to the Commission in a report dated March 15, 2006, recommending that the Commission:

- Authorize an increase in the number of Committee members to the maximum of 15 by 2008

- Approve a gradual increase in the ACRS staff, beginning in FY 2006, (2 senior staff engineers, 2 senior technical advisors, and 1 administrative assistant)

- Approve the necessary travel resources for holding additional Subcommittee meetings beginning in FY 2007.

In an SRM dated April 13, 2006 (pp. 13), the Commission responded to the Committee's request stating the following:

- The Commission has approved an increase in the number of ACRS members to the maximum of 15 by FY 2008

- The overall budget for the ACRS, including FTE for ACRS members and staff, travel funds, and other expenses should continue to be addressed through the budget process

- In determining if additional resources are needed, the ACRS should continue to look at its current budgeted and baseline activities to determine if the level of ACRS support for some of these activities can be reduced or eliminated.

- Some statements in the March 15, 2006 Committee's report could lead to misinterpretation of the breadth of required ACRS activity under Section 29 of the Atomic Energy Act of 1954, as amended. The Committee should carefully consider what is statutorily required of the Committee, including the activities requested by the Commission, as the Committee identifies, prioritizes, and describes its proposed activities.

- The ACRS and the staff should continue to work together to ensure that staff and ACRS reviews of important technical issues are coordinated in a manner to ensure timely resolution of these issues.

The ACRS/ACNW FY 2007-2008 budget request submitted to the Office of the Chief Financial Officer on March 23, 2006 already includes a request to increase the number of ACRS members to the maximum of 15; increase the staff support (5 FTE); and provide the necessary travel resources. This accommodates the Commission direction that the overall budget for the ACRS, including FTE for ACRS members and staff, travel funds, an other expenses should continue to be addressed through the budget process.

With regard to the Commission statement that the ACRS should continue to look at its current budgeted and baseline activities to determine if the level of ACRS support for some of the activities can be reduced or eliminated, it should be noted that the ACRS has a process in place to prioritize items proposed for review during each ACRS meeting. The Planning and Procedures Subcommittee plays a key role in implementing this process. This process was presented to, and discussed with the Commissioners, during the ACRS/Commission meeting on April 11, 2003. During CY 2005, the Committee decided either not to review, or defer its review after reconciliation of public comments, about 30 regulatory matters. This is twice as much as that for CY 2004.

With regard to Commission comment whether all baseline activities listed in the Enclosure to the March 15, 2006 ACRS report, it should be noted that even if some of these items are not explicitly called out in the Atomic Energy Act and may not fall within the statutory purview of the Committee, in accordance with 10 CFR 1.13 the Committee on its own initiative may conduct reviews of specific generic matters or nuclear facility safety-related items. Even with this flexibility, the Committee decided not to address proactive initiatives unless resources permit, and give high priority to the items of significant importance to the Agency.

If additional information is requested during the budget approval process, specifically on those issues raised by the Commission, the ACRS/ACNW Office will provide necessary information.

RECOMMENDATION

The Subcommittee recommends that the ACRS Chairman, Vice Chairman and Executive Director meet with individual Commissioners and discuss the NRC staff's work schedule for design certifications, COLs, and other matters that typically come before the Committee and what resources are needed to support this schedule.

5) Quadripartite Meeting Status

On March 31, 2006, all ACRS abstracts for the 2006 Quadripartite meeting were uploaded to the web site. During the April ACRS meeting, these abstracts were provided to the members for review. Some members provided minor comments. The members should provide final papers and power point presentation slides to Mugeh by Friday, July 28, 2006. We are anticipating receiving the abstract from the Japanese (NSC) prior to the end of May 2006. The Germans (RSK) and French (GPR) have provided most of their abstracts.

Draft letters have been prepared for the ACRS Chairman's review and comment, inviting Commissioners, EDO, and NRC Program Office Directors to participate/attend the Quadripartite Meeting.

RECOMMENDATION

The Subcommittee recommends that the members provide the papers and presentation slides for consideration during the July 12-14, 2006 ACRS meeting and finalize them by July 28, 2006.

6) Streamlining the NRR Rulemaking Process

In a memorandum (COMEXM-06-0006) dated April 7, 2006 (pp. 14-15) Chairman Diaz and Commissioner McGaffigan sent a proposal to Commissioners Merrifield, Jaczko, and Lyons for streamlining the NRR Rulemaking Process. In that memo, it is stated that ". . . not withstanding 10 CFR 2.809 and the Memorandum of Understanding between the ACRS and the EDO, the staff may waive review by the ACRS at the proposed rule stage." Also, it is stated "comments from the ACRS may be submitted to the Commission either during the comment period for the proposed rule, or following the close of the public comment period, but prior to issuance of the final rule."

If implemented, this proposal will limit the number of opportunities that the ACRS has now to review a proposed rule. Also, this will contradict Commission direction in previous SRMs. For example, in the April 5, 2000 SRM, the Commission stated that the ACRS should work with the NRC staff to enhance efforts to risk-inform 10 CFR Part 50, including Appendices A and B.

Also, in the April 13, 2006 SRM, the Commission stated that the ACRS and the staff should continue to work together to ensure that staff and ACRS reviews of important technical issues are coordinated in a manner to ensure timely resolution of these issues.

Without involvement by the ACRS in the early stages of the development of a proposed rule, the Committee may not be able to contribute effectively to the development of a rule. During the survey of the NRC staff related to 2005 self-assessment of ACRS, some NRC staff members stated that "Early interaction by the ACRS with the EDO and the NRC staff on the regulatory significance of complex technical issues was very useful."

A draft SRM is being circulated for comment. The ACRS staff, in consultation with the ACRS Chairman, provided comments on the draft SRM for consideration by the Commission.

RECOMMENDATION

The Subcommittee recommends that the Committee discuss the impact of the above proposal on the effectiveness of the ACRS review of proposed rules and recommend a course of action.

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

Each year, the members visit a nuclear plant and meet with the Regional Administrator to discuss items of mutual interest.

During its April 2006 meeting, the Committee decided to visit the Limerick Nuclear Plant and meet with the Region I Administrator. The proposed dates for the plant visit and meeting with the Regional Administrator are Tuesday, July 25 thru Thursday, July 27, 2006. Mr. Sieber, Plant Operations Subcommittee Chairman, has agreed to develop a list of proposed topics.

RECOMMENDATION

The Subcommittee recommends that the members provide feedback on the proposed dates for visiting the Limerick Nuclear Plant and meeting with the Region I Administrator. Mr. Sieber should provide a list of discussion topics for consideration during the June meeting.

8) Re-design of ACRS Conference Room

The ACRS Conference room is in the process of being upgraded to improve the audio/visual capabilities, including improving the projector and the teleconferencing/video-teleconferencing capabilities. Additionally, as a result of the Commission's approval to allow the ACRS to expand to its statutory limit of 15 members, the conference room table will be re-designed to accommodate the increase in membership and expanded use of laptop computers. The ACRS/ACNW Office staff is in the process of contracting this job, in an attempt to have this work done prior to the end of FY 2006 (September 30, 2006). There will be some inconveniences and optimization issues as we work our way through this job.

9) Ethics Training

NRC employees are required to complete an ethics training based on the government-wide standards of conduct regulations. The annual ethics training for ACRS members will be held at 8:30 a.m. on June 2, 2006. The topics include Office of Government Ethics regulations, security issues, and official government travel guidelines.

RECOMMENDATION

The Subcommittee recommends the members be present at this session to satisfy the training requirement. If the members have any specific issues that they believe should be addressed during this session, they should be sent to Jenny Gallo well in advance of the June meeting.

9) Member Issue

Issues Related to Regulatory Guide 1.174

Dr. Kress states that there are some "incoherences" with the current Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." He has documented his concerns in the attachment (pp. 16-19). If and when the Committee reviews proposed Revision 2 to Regulatory Guide 1.174, he would like to raise these issues.

Based on recent conversation with the NRC staff, we understand that the staff is in the process of revising Revision 1 to Regulatory Guide 1.174 to address PRA quality. The staff has not yet decided whether to address the late containment failure issue in this revision. The staff plans to issue proposed Revision 2 to Regulatory Guide 1.174 in Fall 2006 for public comment. The staff may seek ACRS review after reconciliation of public comments.

Other comments on Regulatory Guide 1.174 provided by Mr. Thadani and Dr. Wallis are attached (pp. 20-21).

RECOMMENDATION

The Subcommittee recommends the following:

- Dr. Kress could raise his issues during the ACRS discussion of the proposed Revision 2 to Regulatory Guide 1.174,

or

- Dr. Kress could forward his comments, as an individual member of the ACRS, to the staff for consideration in the proposed Revision 2 to Regulatory Guide 1.174,

or

- Dr. Kress could have an informal meeting with the staff to express his personal views and get a feel whether the staff agrees with his concerns and is willing to consider in the proposed Revision 2.

ANTICIPATED WORKLOAD
MAY 4-6, 2006

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	—	Thornsbury	**SUBCOMMITTEE REPORT** - ESBWR PRA **[SUBC. Mtg - 4/20/06]**	—	—	—
Denning	—	Caruso	Ginna Power Uprate Application and the Associated Safety Evaluation	A	To support staff schedule	**Draft**
		Caruso	Beaver Valley Power Uprate Application and the Associated Safety Evaluation	A	To support staff schedule	**Draft**
Kress	Powers	Fischer	Proposed Revisions to 10 CFR Part 52	A	To support staff schedule	**Draft**
Sieber	—	Santos	Final Review of the Brunswick License Renewal Application and the Associated Final SER	A	To support staff schedule	**Draft**
	—	Thornsbury	NRC Staff's Response to ACRS Comments on Draft Final Revision 4 to Reg. Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants"	A	To support staff schedule	**Draft**

ANTICIPATED WORKLOAD
May 31 - June 2, 2006

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	—	Fischer/Thornsbury	Risk Management Technical Specifications Initiative 4b-Flexible Completion Times **[TENTATIVE]**	—	—	—
Bonaca	—	Santos	Draft Final Generic Letter 2005-xx, "Inaccessible Underground Cable Failures that Disable Accident Mitigation Systems"	A	To support staff schedule	—
			Interim Staff Guidance on Aging Management Program for Inaccessible Areas of BWR Mark I Containment Drywell Shell **[INFORMATION BRIEFING]**	—	—	—
Denning	—	Junge/Nourbakhsh	Draft Final Generic Letter, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations"	A	To support staff schedule	—
Kress	—	Fisher/Snodderly	Overview of Advanced Reactor Activities **[INFORMATION BRIEFING]**	—	—	—
Wallis	—	Nourbakhsh/ Duraiswamy	Status Report on Quality Assessment of Selected NRC Research Projects	—	—	—

ANTICIPATED WORKLOAD
July 12-14, 2006

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Bonaca	—	Thornsbury	Safeguards and Security Matters **[CLOSED]**	—	—	—
Powers	—	Santos	Results of the Staff Study to Determine the Need for Establishing Limits for Phosphate Ion Concentrations in Groundwater at the Sites of Plants Applying for License Renewal	B	To provide Committee's views	—
		Fischer	Lessons learned from the Review of ESP Applications **[TENTATIVE]**	B	—	—
Shack	—	Nourbakhsh	Integrating Risk and Safety Margins	A	To provide Committee's views	—
Sieber	—	Junge/Santos	Final Review of the Nine Mile Point License Renewal Application and the Associated Final SER	A	To support staff schedule	—
Wallis	—	Caruso	Integrated Chemical Effects tests Related to the Resolution of GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance"	A	To provide Committee's views	—

ACRS Items Requiring Committee Action

1 **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"**

Member: Joseph Armijo **Engineer:** Cayetano Santos

Estimated Time1 hr

Purpose: Determine a Course of Action

Priority:

Requested by: RES W. Cullen, H. Gonzales

At the request of NRR, RES is developing a Regulatory Guide to address the effects of the reactor water environment on the fatigue life of carbon steel, low-alloy steel, and austentic steel components. Section III of the ASME Code specifies fatigue design curves that are based on experiments conducted in air at room temperature. These design curves were established by lowering the best-fit curves of the experimental data by a factor of 2 on stress or 20 on cycles (whichever was more conservative) to account for data scatter and the differences between laboratory specimens and actual components. Environmental effects were not considered. More recent fatigue tests show that light water environments can have a significant impact on fatigue life.

The resolution of GSI-166 (Adequacy of the Fatigue Life of Metal Components) and GSI-190 (Fatigue Evaluation of Metal Components for 60-year Plant Life) relied on conservatism in component fatigue analyses. The fatigue analyses of components in new reactors may not contain the same level of conservatism. This regulatory guide will establish the NRC's position in reviews of new reactor construction applications.

The draft regulatory guide and NUREG/CR-XXXX, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," were provided on April 27, 2006. The staff is requesting that the ACRS defer its review of this draft guide until after the public comment period.

The Planning and Procedures Subcommittee recommends that Dr. Armijo determine a course of action on this matter.

11.

2 Revision 4 of Clinton ESP FSAR: Probable Maximum Flood (PMF)

Member: Dana Powers **Engineer:** David Fischer

Estimated Time:

Purpose: Determine a Course of Action

Priority:

Requested by: John Segala

Revision 4 of Exelon's ESP application for the Clinton site included changes to the maximum rainfall rate, the maximum hydrostatic PMF water surface elevation, the coincident wind wave activity, and the maximum storm surge. Exelon presented PMF calculations using two different synthetic unit hydrograph methods (the Synder method and the Soil Conservation Service method) with two different conceptual watershed layouts (a two-basin plus lake model and a seven-basin plus lake model). The staff has evaluated Exelon's revised PMF analysis and the information in Revision 4 to the EGC ESP application and concluded that the revised analysis conservatively estimated the hydrostatic PMF elevation. The staff performed several independent analyses that confirmed EGC's hydrostatic PMF elevation. The staff has modified the FSER to document the basis for this conclusion.

The Planning and Procedures Subcommittee recommends that Dr. Powers recommend a course of action.

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April 13, 2006

MEMORANDUM TO: Graham B. Wallis, Chairman
Advisory Committee on Reactor Safeguards

FROM: J. Samuel Walker, Acting Secretary **/RA/**

SUBJECT: STAFF REQUIREMENTS - COMSECY-06-0018 - RESPONSE TO STAFF REQUIREMENTS MEMORANDUM - MEETING WITH ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, DECEMBER 8, 2005

The Commission has approved an increase in the number of Advisory Committee on Reactor Safeguards (ACRS) members to the maximum of 15 by FY 2008. The overall budget for the ACRS, including FTE for ACRS members and staff, travel funds, and other expenses, should continue to be addressed through the budget process. In determining if additional resources are needed, the ACRS should continue to look at its current budgeted and baseline activities to determine if the level of ACRS support for some of these activities can be reduced or eliminated.

Some statements in the COMSECY could lead to misinterpretation of the breadth of required ACRS activity under Section 29 of the Atomic Energy Act of 1954, as amended (AEA). Specifically, on Page 2 of the COMSECY, it is stated that "Baseline activities are all high-priority (statutory) activities, and are shown in Enclosure 1." The Committee should carefully consider what is statutorily required of the Committee, including the activities requested by the Commission, as the Committee identifies, prioritizes, and describes its proposed activities.

The ACRS and the staff should continue to work together to ensure that staff and ACRS reviews of important technical issues are coordinated in a manner to ensure timely resolution of these issues.

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
EDO
ACRS
OGC
CFO
OCA

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COMEXM-06-0006

REQUEST REPLY BY: 4/21

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 7, 2006

MEMORANDUM TO: Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons

FROM: Nils J. Diaz
Edward McGaffigan, Jr.

SUBJECT: STREAMLINING THE NRR RULEMAKING PROCESS

In light of increased rulemaking activities, which are only expected to grow in the near future, we believe it is of paramount importance to further enhance NRR rulemaking activities to improve efficiency and timeliness, while eliminating unnecessary burdens. Thus, we propose streamlining the rulemaking process by removing unnecessary constraints, while simultaneously enhancing transparency and public participation. There are several tools by which the agency can achieve these goals, including the following:

- At the discretion of the Director of NRR, and in consultation with the General Counsel, the staff may waive the development and submission of rulemaking plans;

- The staff may waive review by the Committee to Review Generic Requirements ("CRGR") at the proposed rule stage, and, notwithstanding 10 C.F.R. § 2.809 and the Memorandum of Understanding between the ACRS and the EDO, waive review by the Advisory Committee on Reactor Safeguards ("ACRS") at the proposed rule stage (as was done, for example, in the ongoing Part 52 rulemaking). Comments from CRGR should be limited to addressing, at the final rule stage, any public comments received relevant to backfit matters. Comments from the ACRS may be submitted to the Commission either during the comment period for the proposed rule, or following the close of the public comment period, but prior to issuance of the final rule.

- In addition, the staff may release proposed rule text for public review, and hold workshops, if necessary, prior to submission of the rule to the Commission. This has been successfully done in past rulemakings (*i.e.*, rulemakings associated with 10 CFR Parts 26, 35 and 70), and is done for most rulemakings by NMSS, at least with Agreement States. The early release of proposed rule text in concert with workshops should reduce or eliminate the need for extended public comment periods (i.e., those in excess of 75 days).

- An additional tool would be the widespread use of working groups and steering committees, designed to reduce the cumbersome concurrence process and eliminate duplicative management review.

We welcome additional mechanisms that the EDO, the General Counsel, or Director of NRR may develop for streamlining and increasing the transparency of the rulemaking process, thus allocating the appropriate level of resources for the most important rulemaking actions, and

(14)

ensuring that the staff's hands are not tied by perceived or real procedural prerequisites that are unnecessary for a given rulemaking.

These mechanisms should be employed for any rulemaking actions where the Director of NRR sees a net benefit. For example, some of these mechanisms clearly would be appropriate for the pending 10 CFR § 50.68 direct final rule. These techniques will likely save resources, which, with the vastly expanded rulemaking agenda, are a significant concern for the agency. These actions are not intended to reduce any public involvement or eliminate processes mandated by the Administrative Procedure Act. Rather, we believe they will further empower all stakeholders.

The Director of NRR should examine all current and planned rulemakings to assess whether these techniques would be appropriate for current and anticipated rulemaking activities. Any additional mechanisms that would streamline the process further should be raised to the Commission for consideration.

Moreover, we are concerned with contractor dependence in completing our rulemaking activities. Contractors are heavily utilized in NRR rulemakings, including resolution of public comments and development of statements of consideration. With significant elements of the rulemaking process fundamentally outside of the agency's day-to-day control, both resources and schedules could be negatively impacted. The NRR staff, in consultation with OGC, should provide the Commission with a paper addressing the feasibility, as well as the advantages and disadvantages, of reducing contractor dependence in the rulemaking arena. In a related vein, the staff should address the option of OGC assisting in the allocation of resources prior to the proposed rule stage to help determine the most efficient use of resources. Furthermore, the staff should take necessary steps to ensure that, when contracting is needed, it is accomplished in a manner that best serves the needs of the agency; *i.e.*, in the most efficient and effective manner possible.

Finally, the staff should consider whether streamlining mechanisms can be usefully employed by other program offices that undertake rulemaking.

SECY, please track.

cc: A. Vietti-Cook, SECY
L. Reyes, EDO
G. Wallis, ACRS
K. Cyr, OGC
J. Dyer, NRR

15

mbr Issue

From: <TSKress@aol.com>
To: <apostola@mit.edu>, <dapower@sandia.gov>, <graham.b.wallis@dartmouth.edu>, <mvbonaca@snet.net>, <denning@battelle.org>, <jsarmijo@msn.com>, <omaynard@charter.net>, <wjshack@anl.gov>, <JDSIEBER@aol.com>, <jtl@nrc.gov>, <ACT@NRC.GOV>, <sxd1@nrc.gov>
Date: 4/13/06 9:49AM
Subject: A Member Issue

Gentlemen and others:

As you know, I have some "issues" with Regulatory Guide 1.174. If and when we get another shot at it, I would like to submit the attached for your consideration as possible improvements. I suppose this can be considered as a member issue.

Cheers,
The Troublemaker

16

04/12/06

Regulatory Guide 1.174 and Other Member Issues

T. S. Kress

There are some "incoherences" with the current R.G. 1.174 that may need ACRS's attention. I would label these as:

1. Bundling
2. Risk Metrics
3. Acceptance Criteria

1. Bundling.

A "bundled set of changes to the licensing basis can exceed the ΔCDF/ΔLERF criteria and, therefore, would not be an acceptable set of changes. If taken one at a time, however, they could be acceptable [this is an entirely real situation]. The staff requires bundling of "related" changes but "unrelated" changes can come in individually and there is no limit on these other than they meet the criteria on deltas. This appears to me to be an incoherence in the process.

2. Risk Metrics.

As R.G. 1.174 generally deals only with currently operating plants, CDF and LERF are probably still appropriate for design acceptance metrics. However, these are not complete regulatory objectives by any means. We need to include some metric for late releases. In principle, there ought to be a frequency metric that covers all release magnitudes.

3. Acceptance Criteria.

The basic concept in R.G. 1.174 is that there should be only small increases in risk and that the allowed change magnitude will depend on the absolute values of CDF and LERF. While any particular change (or bundled set) is limited to a small increase in the deltas, there is no limit to the number of such changes so long as each still meets the criteria. Therefore there is no reason to believe that, over time, the cumulative delta risk will remain "small". Should it?

I think (therefore I am!) ACRS should address these issues the next time we get to review any update of R.G. 1.174. How would I recommend these "incoherences" be fixed?

I think the various criteria "charts" are unnecessarily complex. I would get rid of the limitations on the deltas and just put limits on the allowed CDF and LERF. Any delta would be acceptable so long as the CDF and LERF remained below the acceptance limits. The acceptance limits for existing plants would be: CDF = $1x10^{-4}$/yr and LERF = $1x10^{-5}$/yr. I would add a late release limit (LRF?) That would be $1x10^{-3}$/yr.

Alternatively, I would strongly consider abandoning the CDF and LERF concepts and use the frequency of exceedance of release of radioactivity of given magnitudes (TEDE?). That is I would use F-C curves where the F is frequency of exceedance and the C is the magnitude of radioactivity release.

Finally, I would add a site related acceptance criterion that would also be an F-C curve where, again, F is the frequency of exceedance but, here, C is the cumulative cost associated with the given frequency. The integral under the F-C curve would be the overall F-C acceptance value. As there is not a unique F-C curve to give a particular integral, I would "anchor" it as follows. Since we are dealing with current operating plants, the normal F-C outputs of PRAs asymptotically approach the CDF as the C approaches zero. Therefore I would make a constant F line until the C approaches a value such that a non-risk averse slope on it gives the desired integral.

From: <TSKress@aol.com>
To: <apostola@mit.edu>, <dapower@sandia.gov>, <graham.b.wallis@dartmouth.edu>, <mvbonaca@snet.net>, <denning@battelle.org>, <jsarmijo@msn.com>, <omaynard@charter.net>, <wjshack@anl.gov>, <JDSIEBER@aol.com>, <ACT@NRC.GOV>, <jtl@nrc.gov>, <sxd1@nrc.gov>
Date: 4/14/06 5:16PM
Subject: Correction on member issue

LRF should be about 5E-5/yr....Thanks to Rich for catching this error.

(19.)

From: Ashok Thadani
To: apostola@mit.edu; dapower@sandia.gov; denning@battelle.org; graham.b.wallis@dartmouth.edu; JDSIEBER@aol.com; John Larkins; jsarmijo@msn.com; mvbonaca@snet.net; omaynard@charter.net; Sam Duraiswamy; TSKress@aol.com; wjshack@anl.gov
Date: 4/13/06 2:02PM
Subject: Re: A Member Issue

During the last PSA meeting in San francisco in September, the industry representatives raised some concerns with RG 1.174 (e.g proposing changes to facilities that would balance increases and decreases in risk to achieve a net reduction in risk as being not acceptable under certain interpretations of the guide) and perhaps the committee wants to take a holistic look at the whole issue of risk informed applications/ lessons learned.

>>> <TSKress@aol.com> 04/13/06 9:48 AM >>>
Gentlemen and others:

As you know, I have some "issues" with Regulatory Guide 1.174. If and when we get another shot at it, I would like to submit the attached for your consideration as possible improvements. I suppose this can be considered as a member issue.

Cheers,
The Troublemaker

CC: Eric Thornsbury; Hossein Nourbakhsh; John Flack; John Larkins; Michael Snodderly

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--- You wrote:

During the last PSA meeting in San francisco in September, the industry representatives raised some concerns with RG 1.174 (e.g proposing changes to facilities that would balance increases and decreases in risk

--- end of quote ---

Ashok,

I have always felt that there should be more incentive to decrease risk, rather than ways to allow increases up to the boundaries of the Regions in 1.174.

G.

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(9)

Design and Qualification Requirements in Regulatory Guide 1.97, Draft Rev. 4

Wesley W. Bowers
Exelon Corporation
Chairman, BWROG RG 1.97 Committee

- **IEEE Std 497-2002 provides an improvement in the selection process for post accident monitoring variables.**
  - **Based on plant safety analysis and emergency operating procedures (EOPs).**
- **NRC endorsement of IEEE 497-2002 should not restrict adoption by existing plants.**
  - **Support modification of Regulatory Position C(1) in April 2006, draft rev. 4 of Reg. Guide 1.97.**

Design and Qualification Requirements

- Initial draft of revision 4 required "full conversion" if a current licensee wants to use the new guide.
- Current BWRs do not full comply with referenced standards.
- Current commitments to design and qualification requirements should be considered acceptable alternatives to standards referenced in draft revision 4 of Reg. Guide 1.97.

Design and Qualification Requirements with Acceptable Alternatives

- **Independence and separation**
  - Section 6.3 of IEEE 497 references IEEE 384-1992.
- **Isolation**
  - Section 6.4 of IEEE 497 references IEEE 384-1992.
- **Power supply**
  - Section 6.6 of IEEE 497 references IEEE 308-1991.

Design and Qualification Requirements with Acceptable Alternatives

- **Environmental and Seismic Qualification**
  - Sections 7.1 through 7.4 of IEEE 497 reference IEEE 344-1987 and IEEE 323-1983.
- **Human Factors**
  - Section 8.1.2 of IEEE 497 references IEEE 1023-1988, IEEE 1289-1998, and ISO 9241-3-1992.
- **Quality Assurance**
  - Section 9 of IEEE 497 references ASME NQA-1-2001.

Summary

Revised regulatory position C(1) in April 2006 draft of revision 4 of Regulatory Guide 1.97 allows operating plants to use current licensing basis for design and qualification requirements in lieu of "full conversion."

(11)

G:\ACRS-SECRETARY\reconciliation.wpd

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
532nd	11	1

Title **RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS**

List of Documents Attached	
See attached list	**11**

Instructions to Preparer	Lead Staff Person
1. Paginate Attachments 2. Punch holes 3. Place Copy in file box	SAM DURAISWAMY

SUBJECT	ANALYSIS	EDO LTR.	ACRS LTR.
raft Final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants" (JDS/EAT)	5/01/06 (p. 1)	4/20/06 (pp. 2-3)	3/28/06 (pp. 4-6)

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

May 1, 2006

MEMORANDUM TO: John D. Sieber, Chair
ACRS Plant Operations Subcommittee

FROM: E. Thornsbury, Senior Staff Engineer

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER ON DRAFT FINAL REVISION 4 TO REGULATORY GUIDE 1.97, "CRITERIA FOR ACCIDENT MONITORING INSTRUMENTATION FOR NUCLEAR POWER PLANTS"

Attached is a copy of the EDO's April 20, 2006 letter of response to the ACRS's March 28, 2006 report on the Committee's review of the Draft Final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants." A copy of the Committee's letter is also attached.

Committee Letter

In its letter, the Committee recommended that the staff not issue the draft final Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4. The Committee recommended the staff revise Regulatory Position 1 to allow licensees to adopt the proposed standard to modify individual accident monitoring instruments without a complete analysis of all accident monitoring instrumentation. The Committee agreed that licensees should not be allowed to partially use the new standard to eliminate or reclassify accident monitoring instrumentation required by earlier standards unless Revision 4 of the Regulatory Guide is adopted in its entirety.

EDO Response

On April 5, 2006, the staff discussed the ACRS recommendations with Drs. Sieber, Bonaca and Maynard to obtain further clarification of the Committee's comments. On the basis of that clarification, the staff proposes to modify Regulatory Position 1 to provide additional guidance to current operating reactor licensees with regard to performing modifications to accident monitoring instrumentation.

Analysis

The EDO's response is satisfactory. The staff plans to present the details of the changes made to address the Committee's recommendations at the 532nd meeting.

cc: ACRS Members
SDuraiswamy
MSnodderly

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 20, 2006

Dr. Graham B. Wallis, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: DRAFT FINAL REVISION 4 TO REGULATORY GUIDE 1.97, "CRITERIA FOR ACCIDENT MONITORING INSTRUMENTATION FOR NUCLEAR POWER PLANTS"

Dear Dr. Wallis:

I am responding to your letter, dated March 28, 2006 (ADAMS Accession #ML060870349), concerning the draft final Revision 4 of Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants" (hereinafter "Rev. 4"). In that letter, you presented feedback and recommendations from the Advisory Committee on Reactor Safeguards (ACRS or the Committee), based on the Committee's in-depth review of the guide and consideration of the related formal presentation by staff from the Office of Nuclear Regulatory Research (RES) on March 10, 2006. We appreciate the time and effort ACRS devoted to reviewing this guide.

As you know, the revised guide describes a new method for selecting and applying criteria to accident monitoring instrumentation, and is intended for new nuclear power plants. Nonetheless, your letter conveyed the Committee's recommendation that the U.S. Nuclear Regulatory Commission (NRC) should not issue Rev. 4 in its present form. In particular, the Committee recommended that "The staff should revise Regulatory Position 1 to allow licensees to adopt the IEEE 497-2002 Standard to modify individual accident monitoring instruments without a complete analysis of all accident monitoring instrumentation." Further, the Committee agreed with the staff's position "that licensees should not be allowed to use the IEEE 497-2002 Standard to eliminate or reclassify accident monitoring instrumentation required by previous editions of this Standard unless Revision 4 to Regulatory Guide 1.97 is adopted in its entirety."

On April 5, 2006, the staff discussed the ACRS recommendations with Drs. Sieber, Bonaca and Maynard to obtain further clarification of the Committee's comments. On the basis of that clarification, the staff proposes to modify Regulatory Position 1 to provide additional guidance to current operating reactor licensees with regard to performing modifications to accident monitoring instrumentation. Specifically, the staff's proposed changes to the draft final guide include removing the previous guidance regarding partial conversions from Regulatory Position 1 and adding the following new guidance regarding modifications:

> If a current operating reactor licensee voluntarily uses the criteria in Revision 4 of this guide to perform modifications that do not involve a conversion, the licensee should first perform an analysis to determine the complete list of accident monitoring variables and their associated types in accordance with the selection criteria in Revision 4.

Such an analysis will provide the licensee with the information needed to review the basis for proposed modifications, its impact on any other post accident monitoring instrumentation and to ensure that the accident monitoring instrumentation requirements for the subject nuclear power plant remain satisfied.

The staff's resolution to the ACRS recommendations provides additional flexibility for licensees of current operating reactors to perform modifications to accident monitoring instrumentation based on Rev. 4 criteria. This change will enable licensees who desire to upgrade a portion of their accident monitoring variables to adopt Rev. 4 for the applicable instrumentation without the unnecessary regulatory burden of converting all variables to all Rev. 4 criteria. The enclosed draft final Revision 4 of Regulatory Guide 1.97 incorporates this proposed resolution.

Also, as stated during the staff's presentation to ACRS on March 10, 2006, this guide is intended for licensees of new nuclear power plants, and conversion to Rev. 4 by licensees of current operating reactors is strictly voluntary. These licensees have made licensing commitments to Rev. 2 or Rev. 3, both of which provide sound technical guidance for the current fleet of light-water reactors. There is no regulatory requirement, incentive, or motivation for licensees to convert to Rev. 4 or to perform modifications in accordance with Rev. 4.

If you have any technical questions on this proposed resolution, please feel free to contact Mr. William E. Kemper at (301) 415-7585 or WEK@nrc.gov.

Sincerely,

for Luis A. Reyes
Executive Director
for Operations

Enclosures:
(1) Final Revision 4 of Regulatory Guide 1.97
(2) IEEE Std. 497-2002

cc w/enclosures: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
SECY

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

ACRSR-2183

March 28, 2006

Luis A. Reyes
Executive Director of Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: DRAFT FINAL REVISION 4 TO REGULATORY GUIDE 1.97, "CRITERIA FOR ACCIDENT MONITORING INSTRUMENTATION FOR NUCLEAR POWER PLANTS"

Dear Mr. Reyes:

During the 530th meeting of the Advisory Committee on Reactor Safeguards, March 9-11, 2006, we reviewed draft final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants." During our review, we had the benefit of discussions with representatives of the NRC staff and the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. Revision 4 to Regulatory Guide 1.97 should not be issued in its present form.

2. The staff should revise Regulatory Position 1 to allow licensees to adopt the IEEE 497-2002 Standard to modify individual accident monitoring instruments without a complete analysis of all accident monitoring instrumentation.

3. We agree that licensees should not be allowed to use the IEEE 497-2002 Standard to eliminate or reclassify accident monitoring instrumentation required by previous editions of this Standard unless Revision 4 to Regulatory Guide 1.97 is adopted in its entirety.

DISCUSSION

Draft final Revision 4 to Regulatory Guide 1.97 endorses, with certain exceptions, IEEE 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations." IEEE Standard 497-2002 supersedes IEEE 497-1981 and IEEE 497-1983, both of which are now inactive standards. This Standard provides a consolidated source of post-accident monitoring requirements and the associated bases for a new generation of advanced nuclear plant designs. This Standard also contains appropriate guidance and a flexible basis for making changes to such systems in operating plants. In addition to incorporating requirements from previous editions of this Standard, Revision 4 to Regulatory Guide 1.97 is designed to consider the current state-of-the-art digital design technology for accident monitoring displays, and incorporates user experience and feedback. This Standard addresses some important aspects of the design, installation, and qualification of digital technology for accident monitoring instrumentation.

The staff has reviewed this Standard and, after consideration of public comments, endorsed it, subject to eight regulatory positions. The staff's positions are technically sound. However, the staff has adopted a position that could frustrate the application of this Standard to modifying and upgrading portions of the accident monitoring instrumentation in existing plants.

Regulatory Position (1) states: "If a current operating reactor licensee voluntarily converts to the criteria in Revision 4 of this guide, the licensee should perform the conversion on the plant's entire accident monitoring program to ensure a complete analysis."

In this position, the staff sets forth its intentions with regard to the applicability of IEEE Standard 497-2002 to current operating reactors. Clause 1.1 of IEEE Standard 497-2002 states that the Standard is intended for new plants, although current plants may find its guidance useful in performing design-basis evaluations or implementing design modifications.

In Revision 4 to Regulatory Guide 1.97, the staff states that conversion means adapting the plant's entire accident monitoring program from the current licensing basis (Revision 2 or 3 of Regulatory Guide 1.97), to the guidance in Revision 4. This adaptation could include physical changes (e.g., replacing an instrument), licensing changes (e.g., technical specification changes), or both for each variable. The staff also recognizes that Revisions 3 and 4 of this Regulatory Guide differ in several ways, including variable type definitions and associated criteria, removal of design and qualification categories, removal of prescriptive tables of monitored variables, analysis required to produce the necessary design-basis documentation, and related changes in licensing basis and/or commitments. These differences could involve modifications to existing instrumentation and could impose unnecessary regulatory burden on current operating reactor licensees, inhibiting the adoption of the IEEE 497-2002 Standard.

Regulatory Position 1 is too restrictive. In the case where a licensee desires to upgrade a portion of its accident monitoring instrumentation, the licensee should be allowed to apply the IEEE 497-2002 Standard to perform such upgrades without being required to perform a complete analysis of the entire set of accident monitoring instruments at the plant.

We agree that in some cases where a licensee may want to eliminate or reclassify an instrument (variable) from its list of accident monitoring variables, the licensee should then be required to adopt the IEEE 497-2002 Standard in its entirety. This will ensure that operators have the necessary information to mitigate any accident, consistent with the Emergency Operating Procedures, Abnormal Operating Procedures, and Emergency Response Guidelines.

We look forward to reviewing the staff's resolution of this matter.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References:

1. Memorandum from J. Wiggins, RES, to J. Larkins, ACRS, Subject: Request for ACRS Review of Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4, January 30,2006 (ADAMS Accession No. ML053640127).

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2. Regulatory Guide 1.97 (Draft was issued as DG-1128, dated June 2005), "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4, December 2005 (ADAMS Accession No. ML053640151).
3. Staff Responses to Public Comments on DG-1128, January 31, 2006 (ADAMS Accession No. ML053640161).
4. IEEE Standard 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Generating Stations," September 2002.
5. Memorandum from J. Larkins, ACRS, to L. Reyes, EDO, Subject: Proposed Regulatory Guide (DG) -1128, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants (Revision 4 Regulatory Guide 1.97)," July 8, 2005 (ADAMS Accession No. ML051950526).

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