UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

March 26, 2004

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

SUBJECT: SUMMARY REPORT - 510th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, MARCH 4-6, 2004 AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

Dear Chairman Diaz:

During its 510th meeting, March 4-6, 2004, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports, letters, and memorandum:

REPORTS:

The following reports to Nils J. Diaz, Chairman, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Report on the Safety Aspects of the License Renewal Application for the Virgil C. Summer Nuclear Station dated March 17, 2004.

- Report on the Safety Aspects of the License Renewal Application for the H. B. Robinson Steam Electric Plant, Unit 2 dated March 18, 2004.

LETTERS:

The following letters to William D. Travers, Executive Director for Operations, NRC, from Mario V. Bonaca, Chairman, ACRS:

- ACRS Review of the Westinghouse Electric Company Application for Certification of the AP1000 Plant Design-Interim Letter dated March 17, 2004.

- Draft Final Revision 3 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" dated March 18, 2004.

MEMORANDUM:

Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, Subject: Anonymous Message Concerning the TRACE Computer Code Development and Review Practices dated March 8, 2004.

HIGHLIGHTS OF KEY ISSUES

1. Safeguards and Security Matters

The Committee was briefed by representatives from the Office of Nuclear Regulatory Research (RES), the Office of Nuclear Security and Incident Response (NSIR), and the Nuclear Energy Institute (NEI) regarding safeguards and security matters. This session was closed to protect information classified as national security information as well as unclassified safeguards information pursuant to 5 U.S.C. 552b(c)(1) and (3).

2. License Renewal Application for the H.B. Robinson Steam Electric Plant, Unit 2

The Committee heard presentations by, and held discussions with, representatives of the NRC staff and Carolina Power and Light Company regarding the staff's final Safety Evaluation Report for the H.B. Robinson Nuclear Power Plant, Unit 2 license renewal application. The staff discussed the resolution of open and confirmatory items that were included in the draft SER. The Committee noted that there are 27 enhanced aging management programs (AMPs) and ten new AMPs. The applicant plans to have 18 of these AMPs in place by mid 2004.

Committee Action

The Committee issued a report to the NRC Chairman on this matter dated March 18, 2004. The Committee recommended that the application for renewing the operating license be approved.

3. Interim Review of the AP1000 Design

The Committee heard presentations by and held discussions with representatives of the NRC staff and Westinghouse concerning the resolution of open items identified in the staff's draft safety evaluation report (DSER) as well as issues previously raised by the ACRS regarding thermal-hydraulic design issues for the AP1000 design, and related design certification matters.

The NRC staff indicated that originally there were 174 open items. Resolution of 164 open items has been completed. The remaining 10 open items include two open items regarding security, three open items addressing the aerosol removal coefficients supplied by Westinghouse, one open item regarding the leak-before-break for main steam piping, and four open items regarding the final AP1000 design control document review and the review of the combined license action items.

During the reviews, the ACRS has identified a number of issues that have subsequently been addressed by Westinghouse and the staff. The ACRS reviews have not addressed security matters and their impact on the design. The ACRS commented on several areas related to the certification of the AP1000 reactor design. These areas are the DSER and design compliance, codes and validation testing, probabilistic risk assessment, materials, and severe accidents.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter dated March 17, 2004, commenting on several areas related to the certification of the AP1000 design. The Committee is looking forward to reviewing the final SER and the resolution of any open items before conducting its final review.

4. License Renewal Application for the Virgil C. Summer Nuclear Station

The Committee met with the NRC staff and representatives of the South Carolina Electric and Gas Company (SCE&G) to review and discuss the results of the staff evaluation of the license renewal application for the South Carolina Electric and Gas Company, V.C. Summer Nuclear Power Station and the associated final Safety Evaluation Report. The applicant has requested approval for continued operation of the plant for a period of 20 years beyond the current license expiration date.

Committee Action

The Committee issued a report to the NRC Chairman dated March 17, 2004, recommending that the SCE&G application for renewal of the operating license for V.C. Summer Nuclear Station be approved.

5. Proposed Criteria for ACRS Evaluation of the Effectiveness (Quality) of the NRC Safety Research Programs

The Office of Nuclear Regulatory Research (RES) has been charged by the EDO to establish a process to evaluate the effectiveness (Quality) and utility of its programs. This evaluation is mandated by the Government Performance and Results Act (GPRA) and needs to be in place during the next fiscal year. The Committee has agreed to assist RES in this assessment. Its review will focus on assessing the quality of research programs. Cost characteristics and timeliness of the results will not be addressed in the ACRS evaluation. Timeliness will be measured as part of a "relevance" review, which is to be performed as a separate but related part of the overall RES quality metric. During the March 2004 ACRS meeting, the Committee discussed a process for developing a quantitative metric (a numerical grade) to be used for evaluating the quality of selected NRC research projects.

Committee Action

The Committee issued a letter to RES providing a proposed strategy for assessing the quality of individual research projects. This strategy is still under consideration, and the Committee has invited RES to comment so that it can better meet the RES management needs. During the April 2004 ACRS meeting, the Committee plans to further discuss criteria in evaluating the quality of the NRC research programs.

6. Divergence in Regulatory Approaches Between U.S. and Several Other Countries

In an April 28, 2003 Staff Requirements Memorandum (SRM), on the April 11, 2003 meeting with Advisory Committee on Reactor Safeguards (ACRS), the Commission stated that "In the course of its routine activities of reviewing and advising the Commission on reactor issues, the Committee should explore and consider other international regulatory approaches. Where there are significant differences in regulatory approaches and requirements. The Commission should be informed." During its March 2004 meeting, the Committee discussed the differences in regulatory approaches between the U.S. and several other countries.

Committee Action

During its April 15-17, 2004 ACRS meeting, the Committee plans to discuss the proposed ACRS report on divergence in regulatory requirements between U.S. and several other countries.

7. Joint Meeting of the ACRS/ACNW with the EDO/Office Directors of NRR/RES/NMSS

The ACRS met with the Executive Director for Operations (EDO) and Directors of the Offices of Nuclear Reactor Regulation, Nuclear Regulatory Research, and Nuclear Material Safety and Safeguards regarding items of mutual interest, including risk-informing 10 CFR 50.46, PWR sump performance issues, PRA quality, spent fuel pool issues, risk-informing NMSS regulations, and transportation-related issues. Several ACNW Members attended this meeting and participated in the discussions.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered three classified responses from the EDO dated September 11, 2003, December 12, 2003, and February 27, 2004, which addressed the ACRS reports on security issues dated July 18, October 27, and November 24, 2003.

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

- The Committee considered the response from the EDO dated February 3, 2004, to the ACRS report dated December 12, 2003, concerning Draft Final Rule Revising 10 CFR 50.48, "Fire Protection," to Permit Licensees to Voluntarily Adopt National Fire Protection Association Standard 805.

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

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from February 5, 2004, through March 3, 2003, the following Subcommittee meetings were held:

- Thermal Hydraulic Phenomena - February 10-11, 2004

The Subcommittee discussed the resolution of open thermal-hydraulic issues related to the AP1000 design, including ADS-4 entrainment, long term cooling, boron concentration, and computer code modeling differences.

- Reliability and Probabilistic Risk Assessment - February 19, 2004

The Subcommittee reviewed the ongoing resolution of public comments on the proposed 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components," and the staff's draft Regulatory Guide endorsing Revision D of NEI 00-04, "10 CFR 50.69 Structures, Systems, and Components Categorization Guideline."

- Planning and Procedures - March 3, 2004

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 ACRS Future Plant Designs Subcommittee has scheduled a meeting on June 25, 2004, to review the Final safety evaluation report regarding the Westinghouse AP1000 design, and related matters. The Committee also scheduled a session during its meeting on July 7-9, 2004 to finalize its views on this matter.

PROPOSED SCHEDULE FOR THE 511th ACRS MEETING

The Committee agreed to consider the following topics during the 511th ACRS meeting, to be held on April 15-17, 2004:

- Action Plan for Implementing the Phased Approach for Improving PRA Quality

- SECY-04-0037, "Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break LOCA Break Size and Plans for Rulemaking on LOCA with Coincident Loss-of-Offsite Power"
- Options and Recommendations for Functional Performance Requirements and Criteria for the Containments of Non-LWRs
- Criteria for Evaluating the Effectiveness (Quality) of the NRC Research Programs
- License Renewal Application for the R. E. Ginna Nuclear Power Plant
- Proposed Generic Communication Regarding Pressurizer Dissimilar Metal Weld Cracking Issues
- Subcommittee Report on the Interim Review of the License Renewal Application for the Dresden and Quad Cities Nuclear Power Plants
- Subcommittee Report on Digital I&C System Matters
- Preparation for Meeting with the NRC Commissioners

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

CERTIFIED

Date Issued: 4/8/2004
Date Certified: 4/19/2004

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A. Reconciliation of ACRS Comments and Recommendations

B. Report on the Meeting of the Planning and Procedures Subcommittee Held on March 3, 2004 (Open)

C. Future Meeting Agenda

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MEMORANDUM:

Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, Subject: Anonymous Message Concerning the TRACE Computer Code Development and Review Practices dated March 8, 2004.

APPENDICES

I. *Federal Register Notice*
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MINUTES OF THE 510th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS MARCH 3-6, 2004 ROCKVILLE, MARYLAND

The 510TH meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on March 3-6, 2004. Notice of this meeting was published in the *Federal Register* on February 20, 2004 (65 FR 7985) (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. Mario V. Bonaca (Chairman), Dr. Graham B. Wallis (Vice Chairman), and Mr. Stephen L. Rosen, (Member-at-Large), Dr. George E. Apostolakis, Dr. F. Peter Ford, Dr. Thomas S. Kress, Mr. Graham M. Leitch, Dr. Dana A. Powers, Dr. Victor H. Ransom, Dr. William J. Shack, and Mr. John D. Sieber. 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. Mario V. Bonaca, 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. Safeguards and Security Matters (Closed)

[Note: Dr. Richard P. Savio was the Designated Federal Official for this portion of the meeting.]

The Committee was briefed by representatives from the Office of Nuclear Regulatory Research (RES), the Office of Nuclear Security and Incident Response (NSIR), and the Nuclear Energy Institute (NEI) regarding safeguards and security matters. This session was closed to protect information classified as national security information as well as unclassified safeguards information pursuant to 5 U.S.C. 552b(c)(1) and (3).

III. License Renewal Application for the H. B. Robinson Steam Electric Plant, Unit 2

[Note: Dr. Bhagwat P. Jain was the Designated Federal Official for this portion of the meeting.]

Mr. R. Stewart of Carolina Power and Light Company (CP&L) began the applicant's presentation and stated that H.B. Robinson nuclear plant (RNP), Unit 2 is adjacent to Unit 1, a coal-fired steam power plant. He provided an overview of Unit 2 history and status, shared resources between the units, operating experience, major equipment replaced and upgraded, boric acid corrosion control program, operating experience, and commitment tracking system. Mr. Stewart stated that all NRC performance indicators and inspection findings are green.

Mr. Stewart stated that there are no major exceptions to the generic aging lessons learned (GALL) report and identified several design features that are unique to RNP Unit 2, such as, grouted tendons, containment liner insulation, 480 volt emergency power, shared site and some systems with a fossil unit, and a safe shutdown diesel in addition to two emergency diesel generators.

Mr. Leitch, ACRS Member, asked about the containment debris sump clogging issue relative to ensuring long term recirculation cooling following a LOCA. In particular, whether the insulation between the containment liner and the stainless steel sheathing will be affected and could potentially contribute to additional debris in the containment sump. Mr. Stewart responded that the function of the insulation is to limit the heat-up rate on the concrete during LOCA. There is a large missile shield around the primary components that would prevent anything from potentially impacting and knocking off this insulation.

Mr. Stewart informed the Committee that within the past 20 years, major equipment such as steam generators, service water piping, and low pressure and high pressure turbine rotors have been replaced or upgraded. CP&L plans to replace the reactor pressure vessel head during its refueling outage in 2005, expand dry fuel storage capacity, and refurbish the generator and the exciter in near future.

Mr. Stewart noted that CP&L has implemented a boric acid corrosion control program at all of its PWRs. He further elaborated that the boric acid corrosion control program requires all plant personnel to recognize borated system leakage, understand its significance, and initiate corrective action when boric acid residue is detected. The program also requires that if carbon

and low-alloy steel components are exposed to boric acid, the components shall be carefully cleaned and visually inspected.

Mr. Rosen asked whether it would be possible to better define a schedule for the implementation of many of the commitments that the licensee had made, rather than just state that they would be completed "before the start of the license renewal period." This is information that would be useful to the ACRS, the staff, and the regional inspectors, for planning purposes. The applicant reported that it would be replacing the reactor vessel head in fall 2005, and was strongly inclined to complete the other actions as soon as possible, consistent with the availability of data and components.

Mr. Stewart stated that there are 47 aging management programs credited for license renewal. This includes 10 existing programs, 27 enhanced programs, and 10 new programs. CP&L committed to have 18 of these programs in place by mid 2004. Mr. Stewart noted that all commitments are listed in the safety evaluation report (SER), and are tracked through the plant action tracking system. The supervisor of licensing/regulatory program has overall responsibility for management of commitment tracking. Mr. Stewart stated that all commitments will be identified in implementing documents and any change will be controlled by the 10 CAR 50.59 process. He also stated that the design configuration control process will incorporate guidance to ensure that requirements of 10 CAR 54.37(b) are met.

Mr. Stewart informed the Committee that all of the NRC performance indicators for RNP are green. Audits and inspections conducted by the Region II staff with regard to the LRA during March 31-April 4, 2003, June 9-27, 2003, and September 9-10, 2003, did not have any findings.

The staff presented an overview of the SER, the subsequent resolution of the open items, and the findings of the associated onsite audit and inspections. Mr. S.K. Mitra opened his presentation with the observation that RNP is the second plant to fully implement the GALL process. Mr. Mitra stated that the staff's review identified two open items and 30 confirmatory items. All open and confirmatory have been resolved.

Mr. Mitra described three inspections and two audits of the license renewal scoping and screening methodology, aging management program (AMP), and the commitment tracking system. He stated that RNP, Unit 2, is the first plant where staff has performed an audit of AMPs for consistency with GALL. Mr. Mitra noted that non-EQ insulated cable and connection program lacked details to conclude consistency with GALL. In response to the finding, the applicant revised the AMP and provided a response that was acceptable to the staff. The inspections and audits confirmed that material condition of the plant was being adequately maintained and the AMPs are consistent with GALL.

Mr. Mitra described the time limited aging analyses that were performed by the licensee to evaluate reactor vessel neutron embrittlement and upper shelf energy, metal fatigue for certain components, environmental qualification issues over a longer plant lifetime than had originally been licensed, grouted concrete containment tendon prestress, and aging of boraflex and foundation pile corrosion. All of these issues have been resolved satisfactorily. In the case of

reactor vessel neutron embrittlement and upper shelf energy, the staff performed its own independent calculations and found the applicant's analyses acceptable.

Committee Action

The Committee issued a report to the NRC Chairman on this matter dated March 16, 2004 in which it recommended that the application be approved.

IV. Interim Review of the AP1000 Design

[Note: Dr. Medhat El-Zeftawy was the Designated Federal Official for this portion of the meeting.]

Dr. Thomas S. Kress, Future Plant Designs Subcommittee Chairman, stated that the purpose of this meeting was to discussions with representatives from the NRC staff and Westinghouse concerning the resolution of open items identified in the staff's draft safety evaluation report (DSER) as well as issues previously raised by the ACRS regarding thermal-hydraulic design issues for the AP1000 design, and related design certification matters.

Mr. John Segala, Office of Nuclear Reactor Regulation (NRR), stated that the AP1000 design is a pressurized water reactor with a power rating of 3415 Mwt with an electrical output of at least 1000 Mwe. The AP1000 design contains features that are new in nature. The most significant improvement to the design is the use of safety systems that employ passive means such as gravity, natural circulation, condensation and evaporation, and stored energy for accident mitigation. These passive systems perform safety injection, residual heat removal, and containment cooling functions.

On July 16-18, 2003, the ACRS Thermal-Hydraulic Subcommittee and the Future Plant Designs Subcommittee held a meeting in Monroeville, PA, and discussed several issues. The Subcommittee(s) members raised the following significant points during their discussion with Westinghouse representatives and the NRC staff:

- In the thermal-hydraulic area, more discussion is in progress between the NRC staff and Westinghouse regarding entrainment, level swell, and boron precipitation.
- In the severe accident environment generated with the MAAP 4 code, more clarification is needed for the methodology used for the AP1000 environment for Lambda calculation.
- Clarification of the dominant core damage sequence from PRA.
- The vessel retention issue, where and how it breaks through the vessel and how is that related to the fuel-clad interaction (FCI).
- The squib valve reliability and the lack of actual valve testing.

- The performance of the sump screens and the acceptance of the AP1000 design to tolerate resident debris on screens.

Westinghouse claims that the AP1000 plant has a design objective of 60 years without a planned replacement of the reactor vessel. However, the design does provide for replaceability of other major components, including the steam generator.

Some features of the AP1000 design include a longer reactor core design, larger pressurizer, an in-containment refueling water storage tank, automatic depressurization system, revised main control room design with a digital microprocessor-based instrumentation and control system, hermetically sealed canned reactor coolant pump motors mounted to the steam generator, and increased battery capacity.

On March 28, 2002, Westinghouse submitted its application to the NRC for final design approval of the AP1000 design in accordance with Appendix O to Part 52 of Title 10 CAR, and for standard design certification in accordance with Subpart B of 10 CAR Part 52. Accordingly, the NRC staff has reviewed the design certification application and has issued its draft safety evaluation report (DSER) on June 16, 2003.

The DSER originally contained 174 open items. Resolution of 164 open items has been completed. The remaining 10 open items include two open items regarding security, three open items addressing the aerosol removal coefficients supplied by Westinghouse, one open item regarding the leak-before-break for main steam piping, and four open items regarding the final AP1000 design control document review and the review of the combined license action items. The security open items will be reviewed separately.

Westinghouse representatives stated that there are two types of failures of the ADS 4 squib valves that were modeled. These are failure to open after receiving a signal to open (5.8E-04/demand used), and opens spuriously (5.4E-05/year is used leading to large LOCA). Three different sources of failure probability were used to establish the AP1000 squib valve reliability in the fail to open mode. These are the ALWR Utility Requirements Document (URD) indicating a failure to operate probability of 3E-03 per demand; Sandia Laboratories failure probabilities of 2.0E-04 and 3.2E-04 per demand; and a geometric mean of the URD and Sandia for a failure probability of 5.8E-04 per demand. This is the value used in the AP1000. Westinghouse claims that the valve reliability information provided by the squib valve vendor CONAX indicated that the squib valves have high inherent reliability and the reliability for smaller valves is applicable to larger valves. In addition, no failures associated with shear section cracking under constant high pressure and temperature are expected.

For spurious opening failure, the dominate cause on ADS 4 valve is considered a spurious signal. Structural failure of the valve is deemed to be much less likely and is not estimated in the AP1000 PRA. The failure frequency of one or more ADS stage 4 squib valves opening due to spurious signal generation is estimated to be 5.4E-05/year. The fact that hardware failure of the valve leading to gross leakage is considered small compared to this value implies that the

contribution of failures from this failure mode is deemed to be less than 5E-06/year, or 5.7E-10/hour for four valves. This translates to 1.4E-10/hour failure rate for a single valve.

Westinghouse used a similar estimate of structural valve failure as a pipe segment. In AP1000 significant leakage from a pipe segment is assigned a failure rate of 8.5E-09/hour. In general, Westinghouse indicated that if the squib valve failure probability to open is doubled, then the plant CDF for internal events at power goes from 2.41E-07/year to 2.77E-07/year-- a 15% increase; and if the spurious opening of the ADS failure probability is doubled, the plant CDF will increase by 12.3%.

Westinghouse described the post-LOCA design basis aerosol deposition in an AP1000 containment. As part of the AP600 design certification a calculation was performed of radiological design basis fission product aerosol removal rates (lambda) by natural processes. Similar to the AP600, the AP1000 containment has a large steel shell cooled on the outside, leading to higher heat transfer rate and higher natural aerosol removal rate for fission product aerosols than would exist from sedimentation alone. Since AP1000 and AP600 have a similar design, the calculation is a repetition of the AP600 calculation with AP1000 parameters and thermal hydraulics. The AP600 sensitivity study was also referenced to assess possible variation of AP1000 lambdas. The AP1000 has, compared to AP600, 75% higher thermal power, 20% larger containment by volume, and 75% more aerosol mass.

The calculation procedure included a selection of the AP1000 sequence that has relatively high probability and has timing that is similar to the specified Regulatory Guide 1.183 timing for PWR fission product release; calculated containment thermal hydraulic conditions for selected sequence using MAAP4 code; and calculated containment aerosol removal rates for MAAP4 thermal hydraulic conditions and aerosol assumptions using STARNAUA code. The STARNAUA code has been documented and benchmarked against experiment. The expected results, compared to AP600, higher diffusion and thermophoresis due to higher heat transfer to containment shell, similar sedimentation due to similar concentration and well-mixed assumption (conservative), and higher containment lambda. Average lambda is 1.1 per hour.

Ms. Jennifer L. Uhle, NRR, outlined some of the AP1000 thermal-hydraulic design review performed by the NRC staff. She stated that the review relied on work performed for the AP600 design, such as RELAP5 code adequacy assessment. The review focused on phenomena that were more important in the AP1000 design such as level swell, entrainment in the upper plenum and hot leg. The NRC did not perform a code acceptance review of NOTRUMP and WCOBRA-TRAC. The identified code deficiencies were handled by performance of bounding calculations to demonstrate margins in the design. All components of the safety demonstration comprise the evaluation model and must be repeated by future licensees. Some of the open items included scaling of APEX, identification of limiting transient, backpressure assumption, early phase collapsed liquid level, long term cooling, and boron precipitation. Currently, the NRC has confirmed that the AP1000 thermal-hydraulic design meets the regulatory requirements and can be licensed.

The NRC staff plans to issue the final safety evaluation report (FSER) on September 13, 2004, issue the final design approval on October 25, 2004, and complete the design certification rulemaking in December 2005.

A portion of this meeting was closed to discuss Westinghouse proprietary information pursuant to 5 U.S.C. 552b(c)(4).

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter dated March 17, 2004, commenting on several areas related to the certification of the AP1000 design. The Committee plans to review the final SER and the resolution of any open items before concluding its review.

V. License Renewal Application (Open)

[Note: Mr. Marvin D. Sykes was the Designated Federal Official for this portion of the meeting.]

The Committee met with the NRC staff and representatives of the South Carolina Electric and Gas Company to review and discuss the results of the staff evaluation of the license renewal application for the South Carolina Electric and Gas Company (SCE&G), V.C. Summer Nuclear Power Station (VCSNS) and the associated FSER. The applicant has requested approval for continued operation of VCSNS for a period of 20 years beyond the current license expiration date.

The license renewal application for VCSNS was submitted by letters dated July 30 and August 6, 2002, and supplemented on September 12, 2002. The letters outlined specific actions that have been or will be taken to manage the effects of aging on the structures and components subject to the aging management review (AMR). The intended functions will be maintained consistent with the current licensing basis during the renewed term of the operating license.

There were **no open or confirmatory items** identified in the initial SER provided to the Committee on October 20, 2003. The final SER was issued ahead of schedule and forwarded to the ACRS on January 30, 2004. The staff concluded in the final SER that the applicant had satisfied the requirements of 10 CAR 54 and imposed two general license conditions requiring the applicant to include the USAR Supplement in the next USAR update required by 10 CAR 50.71(e) following issuance of the renewed license and complete future inspections identified in the USAR Supplement prior to the period of extended operation. No other plant-specific license conditions were included.

Committee's Action

The Committee issued a letter report to the NRC Chairman dated March 17, 2004, which concluded that the programs instituted and committed to by SCE&G, to manage age-related degradation, are appropriate and provide reasonable assurance that the V. C. Summer Nuclear

Station can be operated in accordance with its current licensing bases for the period of extended operation without undue risk to the health and safety of the public. The Committee recommended that the SCE&G application for renewal of the operating license for V.C. Summer Nuclear Station be approved

VI. Proposed Criteria for ACRS Evaluation of the Effectiveness (Quality) of the NRC Safety Research Programs (Open)

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

The Office of Nuclear Regulatory Research (RES) has been charged by the EDO to establish a process to evaluate the effectiveness (Quality) and utility of its programs. This evaluation is mandated by the Government Performance and Results Act (GPRA) and needs to be in place during the next fiscal year. The Committee has agreed to assist RES in this assessment. Its review will focus on assessing the quality of research programs. Cost characteristics and timeliness of the results will not be addressed in the ACRS evaluation. Timeliness will be measured as part of a "relevance" review, which is to be performed as a separate but related part of the overall RES quality metric. During the March 2004 ACRS meeting, the Committee discussed a process for developing a quantitative metric (a numerical grade) to be used for evaluating the quality of selected NRC research projects.

Committee Action

The Committee issued a letter to RES providing a proposed strategy for assessing the quality of individual research projects. This strategy is still under consideration, and the Committee has invited RES to comment so that it can better meet the RES management needs. During the April 2004 ACRS meeting, the Committee plans to further discuss criteria in evaluating the quality of the NRC research programs.

VII. Divergence in Regulatory Approaches Between U.S. and Several Other Countries (Open)

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

In an April 28, 2003 Staff Requirements Memorandum (SRM), on the April 11, 2003 meeting with Advisory Committee on Reactor Safeguards (ACRS), the Commission stated that "In the course of its routine activities of reviewing and advising the Commission on reactor issues, the Committee should explore and consider other international regulatory approaches. Where there are significant differences in regulatory approaches and requirements. The Commission should be informed." During its March 2004 meeting, the Committee discussed the differences in regulatory approaches between the U.S. and several other countries.

Committee Action

During its April 15-17, 2004 ACRS meeting, the Committee plans to discuss the proposed ACRS report on divergence in regulatory requirements between U.S. and several other countries.

VIII. Executive Session (Open)

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

A. Reconciliation of ACRS Comments and Recommendations/EDO Commitments

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

- The Committee considered three classified responses from the EDO dated September 11, 2003, December 12, 2003, and February 27, 2004, which addressed the ACRS reports on security issues dated July 18, October 27, and November 24, 2003.

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

- The Committee considered the response from the EDO dated February 3, 2004, to the ACRS report dated December 12, 2003, concerning Draft Final Rule Revising 10 CAR 50.48, "Fire Protection," to Permit Licensees to Voluntarily Adopt National Fire Protection Association Standard 805.

  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 March 3, 2004. The following items were discussed:

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

Member assignments and priorities for ACRS reports and letters for the March 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 May 2004 were addressed. 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

ACRS Meeting with the NRC Commissioners

The ACRS is scheduled to meet with the NRC Commissioners on Thursday, May 6, 2004, to discuss items of mutual interest. The following topics were proposed by the Committee during the February 2004 ACRS meeting:

A.) Overview (MVB)
B.) PRA Quality (GEA)
C.) NRC Safety Research Program Report (DAP)
D.) Interim Review of the AP1000 Design (TSK)
E.) ESBWR Pre-Application Review (TSK)
F.) Risk-Informing 10 CAR 50.46 (WJS)
G.) PWR Sump Performance (JDS)

ACRS Report on the NRC Safety Research Program

An advanced copy of the 2004 ACRS report on the NRC Safety Research Program was sent to the Commission on February 26, 2004. Any feedback from the Commissioners will be provided to Dr. Powers for incorporation, as appropriate. Subsequently, the final report will be published as NUREG-1635, Vol. 6.

Revision to ACRS Action Plan

As agreed to by the Committee during its January 29-30, 2004 retreat, the ACRS Action Plan that was issued in 2001 is being revised. A significant discussion of the proposed revision to the Action Plan will be the planned ACRS pro-active initiatives. Members should provide comments on items to be included or dropped from the current Action Plan to Maggalean Weston by March 12, 2004. A proposed revision to the Action Plan will be provided to the Planning and Procedures Subcommittee for consideration during its April 14, 2004 meeting. Subsequent to incorporating the Subcommittee's comments, this Plan will be provided to the ACRS members for comment.

Proposed Generic Letter 2004-XX, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors"

The staff plans to issue the subject Generic Letter (GL) for public comment in March 2004. This GL is to request the licensees to submit information to the NRC concerning the status of their compliance with 10 CAR 50.46 (b)(5), which requires long-term reactor core cooling, and with the additional plant-specific licensing basis requirements listed in the GL. This request is based on the identified potential susceptibility of PWR recirculation sump screens to debris blockage during design basis accidents requiring recirculation operation of the emergency core cooling system (ECCS) or containment spray system (CSS) and the potential for additional adverse effects due to debris blockage of flow paths necessary for ECCS and CSS recirculation and containment drainage.

We understand that the staff plans to provide the ACRS with the draft final version of this GL after reconciliation of public comments. In view of the significance of this issue and the expected discussion of the issue of PWR sump performance at the ACRS Commission meeting on May 6, 2004, the Committee needs to decide whether to review this GL prior to issuance for public comment or after reconciliation of public comments.

ACRS Member Notebooks

Prior to the March 2004 meeting, ACRS members received a CD containing the normal material included in the meeting notebooks. If this is an acceptable procedure, we will issue a CD each month prior to the meeting, in lieu of providing a three ring binder.

Visit to a Nuclear Plant and Regional Office

Each year the members visit a nuclear plant and the NRC Regional Office and meet with the licensee and the regional staff to discuss items of mutual interest. The Committee should decide on the plant to be visited and the date. We have been rotating between regions and this year's visit would normally be to Region III. Plants for consideration are Dresden, LaSalle, or Quad Cities.

Ethics Refresher Training

Mr. John Szabo, OGC, has agreed to provide ethics refresher training to the ACRS members on April 16, 2004, and answer any questions the members may have on this subject. If the members would like to send questions to Mr. Szabo in advance of the April meeting, they can do so through the ACRS Executive Director. Part of this hour will also be used to cover security and travel issues.

Review of NRC Codes

A member of the public sent an e-mail to Dr. Wallis, expressing concerns about the process used for reviewing NRC codes. The specific concern is about the process for reviewing the TRACE Code. Key points include:

- Several issues are overlooked when NRC codes are under review. However, they are not overlooked when NRC reviews codes submitted by other organizations.

- Extreme adversarial environment has been present at the NRC for the past 30 years. Free and open discussion of important technical issues has not been possible for all these years.

- The ACRS Thermal-Hydraulic Phenomena Subcommittee and its consultants have individual agendas and do not listen to the very people who know the most about the subject matter presented.

- Those organizations who submit codes for NRC review have very specialized people who know exactly what is important for each and every application of their codes and experimental data. The ACRS and its consultants, on the other hand, generally do not have the time, or more importantly the inclination, to digest the material to the depth necessary to understand the important issues. The Thermal-Hydraulic Phenomena Subcommittee and its consultants and the material on which they focus and the manner on which they discuss the material are the subject of many jokes and not-so kind comments all over the industry.

- It is accepted procedure that computer codes must be verified before the models and method are validated. Generally, the codes developed under NRC funding have never undergone verification. Additionally, almost all of the validation or assessment calculation done with the NRC codes have not been done under an approved and qualified procedure.

- It does not appear that verification of the TRACE code will be performed. Also, it is not clear that the NRC has a qualified and approved QA plan in place for the TRACE code. Such plans are required by the NRC for commercial organizations.

- If the documentation for the numerical solution methods used in TRACE are studied in detail, the results will show the basic SETS solution method is based on less-than-exact methodologies.

- There is a basic problem that has never been addressed, i.e., the numerical method does not solve for the void fraction in a way that can be theoretically justified.

This appears to be an allegation and has been forwarded to the EDO for action.

License Amendment Request by Duke Power to Insert Mixed Oxide Lead Test Assemblies

On February 27, 2003, Duke Energy Corporation filed a license amendment to revise the McGuire and Catawba Technical Specifications to allow insertion of four mixed oxide (MOX) lead test assemblies at either the McGuire or the Catawba Nuclear Station. Subsequently, the Blue Ridge Environmental Defense League (BREDL) and the Nuclear Information and Resources Service filed petition to intervene, and requests for hearings.

The Atomic Safety and Licensing Board (ASLB) decided in favor of the interveners on certain issues which are noted below. In an Order dated February 18, 2004 the Commission:

- Reversed the ASLB decision to allow access to BREDL to safeguards information.
- Exercised its supervisory authority and overturned another order of the ASLB that gave BREDL representatives a right to attend a safeguards-related meeting between the staff and Duke Power.
- Established a filing schedule for the filing of security-related contingencies by BREDL.

In an e-mail dated February 25, 2004, Dr. Powers stated that during the ACRS review of the Duke's license amendment, the interveners may raise issues that have been decided by the Commission. The Committee should discuss how the DFO and the Subcommittee Chairman should react to the issues that may be raised by interveners during the meeting, especially on those issues that the Commission has already taken a position. He suggested that the ACRS Chairman inform the Commissioners of the potential for the public to raise these issues at future ACRS meetings; and, to assure the Commissioners that the ACRS does not want to get involved in safeguards and security issues related to this matter in which the Commission has already taken a position.

LINK Technologies, Inc., Report

At the request of Mr. Rosen, LINK Technologies, Inc. has prepared a report that includes recommendations for enhancing the NRC training material for inspecting a licensee's corrective action program and explores the possibility of implementing performance indicators in the reactor oversight process for addressing the corrective action programs. Copies of the LINK report has been distributed to the members this morning.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 511th ACRS Meeting, April 15-17, 2004.

The 510th ACRS meeting was adjourned at 12:30 p.m. on March 6, 2004.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 8, 2004

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 510th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - MARCH 3-6, 2004

Enclosed are the proposed minutes of the 510th meeting of the ACRS. This draft is being provided to give you an opportunity to review the record of this meeting and provide comments. Your comments will be incorporated into the final certified set of minutes as appropriate, which will be distributed within six (6) working days from the date of this memorandum.

Attachment:
As stated

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 19, 2004

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

FROM: Mario V. Bonaca
Chairman

SUBJECT: CERTIFIED MINUTES OF THE 510th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), MARCH 3-6, 2004

I certify that based on my review of the minutes from the 510th 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.

1. *The title of the information collection:* 10 CFR Part 25—Access Authorization for Licensee Personnel.

2. *Current OMB approval number:* 3150–0046

3. *How often the collection is required:* On occasion.

4. *Who is required or asked to report:* NRC-regulated facilities and other organizations requiring access to NRC-classified information.

5. *The number of annual respondents:* 50

6. *The number of hours needed annually to complete the requirement or request:* 267 hours (242 hours reporting and 25 hours recordkeeping)

7. *Abstract:* NRC-regulated facilities and other organizations are required to provide information and maintain records to ensure that an adequate level of protection is provided NRC-classified information and material.

Submit, by April 20, 2004, comments that address the following questions:

1. Is the proposed collection of information necessary for the NRC to properly perform its functions? Does the information have practical utility?

2. Is the burden estimate accurate?

3. Is there a way to enhance the quality, utility, and clarity of the information to be collected?

4. How can the burden of the information collection be minimized, including the use of automated collection techniques or other forms of information technology?

A copy of the draft supporting statement may be viewed free of charge at the NRC Public Document Room, One White Flint North, 11555 Rockville Pike, Room O–1 F21, Rockville, MD 20852. OMB clearance requests are available at the NRC worldwide web site: *http://www.nrc.gov/public-involve/doc-comment/omb/index.html.* The document will be available on the NRC home page site for 60 days after the signature date of this notice.

Comments and questions about the information collection requirements may be directed to the NRC Clearance Officer, Brenda Jo. Shelton, U.S. Nuclear Regulatory Commission, T–5 F52, Washington, DC 20555–0001, by telephone at 301–415–7233, or by Internet electronic mail to *INFOCOLLECTS@NRC.GOV.*

Dated at Rockville, Maryland, this 12th day of February 2004.

For the Nuclear Regulatory Commission.

Brenda Jo. Shelton,

NRC Clearance Officer, Office of the Chief Information Officer.

[FR Doc. 04–3675 Filed 2–19–04; 8:45 am]

BILLING CODE 7590–01–U

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 March 3–6, 2004, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Monday, November 21, 2003 (68 FR 65743).

Wednesday, March 3, 2004 (Closed)

11 a.m.–6:30 p.m.: Safeguards and Security (Closed)—The Committee will hear presentations by and hold discussions with representatives of the Office of Nuclear Regulatory Research, the Office of Nuclear Security and Incident Response, and the Nuclear Energy Institute regarding safeguards and security matters.

Thursday, March 4, 2004, 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.: License Renewal Application for the H. B. Robinson Steam Electric Plant, Unit 2 (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Carolina Power and Light regarding the License Renewal Application for the H. B. Robinson Steam Electric Plant, Unit 2 and the associated final Safety Evaluation Report prepared by the NRC staff.

10:15 a.m.–12:15 p.m.: Interim Review of the AP1000 Design (Open/Closed)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Westinghouse regarding the resolution of open items identified in the NRC staff's Draft Safety Evaluation Report as well as the issues previously raised by the ACRS Subcommittee on Thermal-Hydraulic Phenomena, and related matters.

1:15 p.m.–2:45 p.m.: License Renewal Application for the Virgil C. Summer Nuclear Station (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and South Carolina Electric and Gas regarding the License Renewal Application for the Virgil C. Summer Nuclear Station and the associated final Safety Evaluation Report prepared by the NRC staff.

3 p.m.–4 p.m.: Proposed Criteria for ACRS Evaluation of the Effectiveness (Quality) of the NRC Safety Research Programs (Open)—The Committee will discuss the proposed criteria for use by the ACRS in evaluating the effectiveness (Quality) of the NRC safety research programs.

4:15 p.m.–6:15 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting, as well as proposed ACRS reports on Resolution of Certain Items Identified by the ACRS in NUREG–1740 Related to Differing Professional Opinion on Steam Generator Tube Integrity, and Response to the December 22, 2003 EDO Response to the September 30, 2003 ACRS Report on the Draft Final Revision 3 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident."

Friday, March 5, 2004, 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.–9:15 a.m.: Divergence in Regulatory Approaches Between U.S. and Several Other Countries (Open)—The Committee will discuss the differences in regulatory approaches between U.S. and several other countries.

9:30 a.m.–11:30 a.m.: Joint Meeting of ACRS/ACNW with the EDO/Office Directors of NRR/RES/NMSS (Open)—The Committee will meet with the NRC Executive Director for Operations (EDO) and Directors of the Offices of Nuclear Reactor Regulation (NRR), Nuclear Regulatory Research (RES), and Nuclear Material Safety and Safeguards (NMSS) to discuss items of mutual interest, including: Risk-informing 10 CFR 50.46, PWR sump performance issues, PRA quality, spent fuel pool issues, risk-informing NMSS regulations, and transportation-related issues.

12:30 p.m.–1:30 p.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of

ACRS business, including anticipated workload and member assignments.

1:30 p.m.–1:45 p.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the EDO to comments and recommendations included in recent ACRS reports and letters. The EDO responses are expected to be made available to the Committee prior to the meeting.

2:00 p.m.–6:30 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Saturday, March 6, 2004, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

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

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 16, 2003 (68 FR 59644). In accordance with those procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting. Persons desiring to make oral statements should notify the Cognizant ACRS staff named below five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Cognizant ACRS staff prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Cognizant ACRS staff if such rescheduling would result in major inconvenience.

In accordance with Subsection 10(d) Public Law 92–463, I have determined that it is necessary to close a portion of this meeting noted above to discuss and protect information classified as national security information as well as unclassified safeguards information pursuant to 5 U.S.C. 552b(c)(1) and (3), and Westinghouse proprietary information per 5 U.S.C. 552(b)(c)(4).

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, as well as the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Sam Duraiswamy, Cognizant ACRS staff (301–415–7364), between 7:30 a.m. and 4:15 p.m., et.

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

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

Dated: February 13, 2004.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 04–3674 Filed 2–19–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Notice of Availability of Model Application Concerning Technical Specification Improvement To Extend the Completion Times for Inoperable Containment Isolation Valves at Combustion Engineering Plants Using the Consolidated Line Item Improvement Process

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of Availability.

SUMMARY: Notice is hereby given that the staff of the Nuclear Regulatory Commission (NRC) has prepared a model application relating to changes to the completion time in Standard Technical Specifications (STS) 3.6.3, "Containment Isolation Valves (Atmospheric and Dual)," for Combustion Engineering (CE) plants. The change to the Technical Specifications (TSs) would extend to 7 days the completion time to isolate the affected penetration flow path when selected containment isolation valves (CIVs) are inoperable in either a penetration flow path with two CIVs or in a penetration flow path with one CIV in a closed system. These changes are based on Revision 2 of Technical Specification Task Force (TSTF) change traveler TSTF–373, "Increase CIV Completion Time in Accordance with CE–NPSD–1168," which has been approved for incorporation into the STS for CE plants (NUREG–1432). The purpose of this model is to permit the NRC to efficiently process amendments that propose to modify TSs to extend the completion time for CIVs. Licensees of nuclear power reactors to which the model applies may request amendments using the model application.

DATES: The NRC staff issued a **Federal Register** Notice (68 FR 64375, November 13, 2003) which provided a model safety evaluation (SE) and a model no significant hazards consideration (NSHC) determination relating to the extension of the completion time for TS actions related to inoperable CIVs at CE plants. The NRC staff hereby announces that the model SE and NSHC determination may be referenced in plant-specific applications to extend the CIV completion times as described in Revision 2 to TSTF–373. The staff has posted a model application on the NRC web site to assist licensees in using the consolidated line item improvement process (CLIIP) to request the subject TS change. The NRC staff can most efficiently consider applications based upon the model application if the application is submitted within a year of this **Federal Register** Notice.

FOR FURTHER INFORMATION CONTACT: William Reckley, Mail Stop: O–7D1, Division of Licensing Project Management, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone (301) 415–1323.

SUPPLEMENTARY INFORMATION:

Background

Regulatory Issue Summary 2000–06, "Consolidated Line Item Improvement Process for Adopting Standard Technical Specifications Changes for Power Reactors," was issued on March

APPENDIX II

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

February 12, 2004

SCHEDULE AND OUTLINE FOR DISCUSSION
510th ACRS MEETING
MARCH 3-6, 2004

WEDNESDAY, MARCH 3, 2004 (CLOSED)

1) 11:00 - 11:10 A.M. Opening Remarks by the ACRS Chairman **(Closed)** (MVB/JTL)

2) 11:10 - ~~6:30~~ 5:15 P.M. Safeguards and Security Matters **(Closed)** (MVB/RPS/RKM)
(12:30-1:30 P.M.- LUNCH)
2.1) Remarks by the Subcommittee Chairman
2.2) Briefing by and discussions with representatives of the Office of Nuclear Regulatory Research, the Office of Nuclear Security and Incident Response, and the Nuclear Energy Institute regarding safeguards and security matters.

[NOTE: This session will be closed to protect information classified as national security information as well as unclassified safeguards information pursuant to 5 U.S.C. 552b(c)(1) and (3).]

THURSDAY, MARCH 4, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

3) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open)
3.1) Opening Statement (MVB/JTL/SD)
3.2) Items of current interest (MVB/SD)

4) 8:35 - ~~10:00~~ 9:45 A.M. License Renewal Application for the H. B. Robinson Steam Electric Plant, Unit 2 (Open) (GML/MVB/BPJ)
4.1) Remarks by the Subcommittee Chairman
4.2) Briefing by and discussions with representatives of the NRC staff and Carolina Power and Light regarding the License Renewal Application for the H. B. Robinson Steam Electric Plant, Unit 2 and the associated final Safety Evaluation Report prepared by the NRC staff.

~~10:00~~ 9:45 - 10:15 A.M. ***BREAK***

5) 10:15 - ~~12:15~~ 12:45 P.M. Interim Review of the AP1000 Design (Open/Closed) (TSK/GBW/MME)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff and Westinghouse regarding the resolution of the open items identified in the NRC staff's Draft Safety Evaluation Report as well as the issues previously raised by the ACRS Subcommittee on Thermal-Hydraulic Phenomena, and related matters.

[CLOSED 11:00 am - 12:45 pm]

[NOTE: A portion of this session may be closed to discuss Westinghouse proprietary information pursuant to 5 U.S.C. 552b(c)(4).]

~~12:15~~ 12:45 - ~~1:15~~ 1:30 P.M. ***LUNCH***

6) ~~1:15~~ 1:30 - ~~2:45~~ 2:37 P.M. License Renewal Application for the Virgil C. Summer Nuclear Station (Open) (MVB/GML/MDS)

6.1) Remarks by the Subcommittee Chairman

6.2) Briefing by and discussions with representatives of the NRC staff and South Carolina Electric and Gas regarding the License Renewal Application for the Virgil C. Summer Nuclear Station and the associated final Safety Evaluation Report prepared by the NRC staff.

~~2:45~~ 2:37 - 3:00 P.M. ***BREAK***

7) 3:00 - ~~4:00~~ 4:19 P.M. Proposed Criteria for ACRS Evaluation of the Effectiveness (Quality) of the NRC Safety Research Programs (Open) (DAP/GEA/GBW/WJS/HPN/SD)

7.1) Remarks by the Subcommittee Chairman

7.2) Discussion of the proposed criteria for use by the ACRS in evaluating the effectiveness (Quality) of the NRC safety research programs.

Representatives of the NRC staff may provide their views, as appropriate.

4:00 - 4:15 P.M. ***BREAK***

8) 4:15 - ~~6:15~~ 6:30 P.M. Preparation of ACRS Reports (Open)

Discussion of the proposed ACRS reports on:

8.1) License Renewal Application for the H. B. Robinson Steam Electric Plant, Unit 2 (GML/MVB/BPJ)

4:40-5:35 8.2) Interim Review of the AP1000 Design (TSK/GBW/MME)

8.3) License Renewal Application for the Virgil C. Summer Nuclear Station (MVB/GML/MDS)

5:50-6:30 8.4) Resolution of Certain Items Identified by the ACRS in NUREG-1740 Related to Differing Professional Opinion on Steam Generator Tube Integrity (FPF/GBW/BPJ)

8.5) Proposed Criteria for ACRS Evaluation of the Effectiveness (Quality) of the NRC Safety Research Programs (DAP/GEA/GBW/WJS/HPN/SD)

5:35-5:49 8.6) Response to the December 22, 2003 EDO Response to the September 30, 2003 ACRS Report on the Draft Final Revision 3 to Regulatory Guide 1.82 (GBW/RC) FINAL

FRIDAY, MARCH 5, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

9) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (MVB/JTL/SD)

10) 8:35 - 9:15 A.M. Divergence in Regulatory Approaches Between U.S. and Several Other Countries (Open) (DAP/HPN/SD)

10.1) Remarks by the Subcommittee Chairman

10.2) Discussion of the differences in regulatory approaches between U.S. and several other countries.

9:15 - 9:30 A.M. *****BREAK*****

11) 9:30 - ~~11:30~~ 11:10 A.M. Joint Meeting of the ACRS/ACNW with the EDO/Office Directors of NRR/RES/NMSS (Open) (MVB/BJG/JTL)

11.1) Remarks by the ACRS/ACNW Chairmen

11.2) Briefing by and discussions with the Executive Director for Operations (EDO) and Directors of the Offices of Nuclear Reactor Regulation (NRR), Nuclear Regulatory Research (RES), and Nuclear Material Safety and Safeguards (NMSS) regarding items of mutual interest, including: risk-informing 10 CFR 50.46, PWR sump performance issues, PRA quality, spent fuel pool issues, risk-informing NMSS regulations, and transportation-related issues.

11:30 - 12:30 P.M. *****LUNCH******

12) 12:30 - 1:30 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (MVB/JTL/SD)

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

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

13) 1:30 - 1:45 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (MVB, 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.

1:45 - 2:00 P.M. ***BREAK***

14) 2:00 - ~~6:30~~ 6:40 P.M. Preparation of ACRS Reports (Open)
Discussion of the proposed ACRS reports on:

2:30-3:35 14.1) License Renewal Application for the H. B. Robinson Steam Electric Plant, Unit 2 (GML/MVB/BPJ) FINAL

5:17-6:37 14.2) Interim Review of the AP1000 Design (TSK/GBW/MME)

3:35-5:00 14.3) License Renewal Application for the Virgil C. Summer Nuclear Station (MVB/GML/MDS) FINAL

14.4) Resolution of Certain Items Identified by the ACRS in NUREG-1740 Related to Differing Professional Opinion on Steam Generator Tube Integrity (FPF/GBW/BPJ)

14.5) ~~Response to the December 22, 2003 EDO Response to the September 30, 2003 ACRS Report on the Draft Final Revision 3 to Regulatory Guide 1.82 (GBW/RC)~~

14.6) Proposed Criteria for ACRS Evaluation of the Effectiveness (Quality) of the NRC Safety Research Programs (DAP/GEA/GBW/WJS/HPN/SD)

14.7) Response to SRM on Divergence in Regulatory Approaches Between U.S. and Several Other Countries (DAP/HPN/SD)

SATURDAY, MARCH 6, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

15) 8:30 - 12:00 Noon Preparation of ACRS Reports (Open)
The Committee will continue discussion of the proposed ACRS reports listed under Item 14.

8:30-9:30 AP-1000 FINAL

16) 12:00 - 12:30 P.M. Miscellaneous (Open) (MVB/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.

9:30-9:47 RES Quality FINAL

NOTE: 10:30-12:00 DPO letter discussion

- **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) copies of the presentation materials should be provided to the ACRS.**

APPENDIX III: MEETING ATTENDEES

510th ACRS MEETING
MARCH 3-6, 2004

NRC STAFF (3/4/2004)

Y. Gene Hsii, NRR
L. Ward, NRR
L. Lois, NRR
W. Johnson, NRR
L. Q. Navarro, NRR
E. Throm, NRR
S. Sun, NRR
J. Colaccino, NRR
M. Hart, NRR
J. Wilson, NRR
B. Palla, NRR
S. Bloom, NRR
J. Starefos, RES
R. architzel, NRR
A. Druzd, NRR
N. Saltos, NRR
S. Bajorek, RES
J. Lyons, NRR
J. Guo, NRR
D. Nguyen, NRR
R. Subbaratnam, NRR
P. Shemanski, NRR
R. Auluck, NRR
J. Raval, NRR
R. Anand, NRR
J. Segato, NRR
R. Arrighi, NRR
N. Iqbal, NRR
D. Jeng, NRR
G. Georgiev, NRR
B. Elliot, NRR
J. Struisha, NRR
S. Lee, NRR
Y. C. Li, NRR
S. West, NRR
K. Chang, NRR
C. Lauron, NRR
J. Honcharik, NRR
J. Tsao, NRR
A. Lee, NRR
J. Mitra, NRR
P.T. Kuo, NRR
J. Strnisha, NRR
A. N. Pal, NRR
B. Fu, NRR
S. Lee, NRR
F. Talbot, NRR
R. Auluck, NRR
M. Murgan, NRR
Y. Diaz, NRR
R. Taylor, NRR
A. Lee, NRR
L. Lois, NRR
S. Milton, NRR
J. Rivera-Ortiz, NRR
Y. C. Li, NRR
R. Arighi, NRR
J. Golla, NRR
S. Marshall, RES
K. Chang, NRR
D. Frumkin, NRR
S. Coffin, NRR
K. Corp, NRR
P. Kang, NRR
C. Patel, NRR
R. Architzel, NRR
H. Walker, NRR
J. Tsao, NRR
J. Medoff, NRR
S. Miranda, NRR
P. Y. Chen, NRR
H. Asher, NRR
D. Jeng, NRR
M. Hartzman, NRR
S. Bentes, NRR
S. Jones, NRR
C. Li, NRR
J. Ma, NRR
C. Lauren, NRR
M. Jenkins, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

R. Stewart, Progress Energy
M. Heath, Progress Energy
G. Wrobel, RGTE,
S. Sterret, Duke University
E. Cummins, Westinghouse
C. Frepoli, Westinghouse
S. Traiforos, LINK
T. Estes, SCE&G
S. Crumbo, SCE&G
R. Vijuk, Westinghouse
C. Frepoli, Westinghouse
G. Wrobel, RG&E
T. Clements, Progress Energy
J. Kozyra, Progress Energy
J. Donohue, Progress Energy
K. Ohkawa, Westinghouse
J. Scobel, Westinghouse
T. Schulz, Westinghouse
J. LaBorde, SCE&G
M. Dantzler, SCE&G
R. Gold, Westinghouse
K. Ohkawa, Westinghouse
R. Clary, SCE&G

NRC STAFF (3/5/2004)
T. Liu, NRR
S. Jones, NRR
S. Miranda, NRR
C. Wu, NRR
J. Ma, NRR
M. Hartzman, NRR
R. McNally, NRR
A. Levin, RES
L. Rossback, NRR
T. Scarbrough, NRR
I. Schoenfeld, OEDO
E. Benner, NRR
M. Waters, NMSS
M. Drouin, RES
J. Strosnider, RES
C. Grimes, NRR
C. Paperiello, OEDO
D. Widmayer, NMSS
S. Black, NRR
D. Terao, NRR
A. McMurtray, NRR

APPENDIX IV

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

March 16, 2004

SCHEDULE AND OUTLINE FOR DISCUSSION
511th ACRS MEETING
APRIL 15-17, 2004

THURSDAY, APRIL 15, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - 10:00 A.M. Action Plan for Implementing the Phased Approach for Improving PRA Quality (Open) (GEA/MRS)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff regarding the staff's proposed action plan for implementing the phased approach for improving PRA quality.

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

10:00 - 10:15 A.M. ***BREAK***

3) 10:15 - 12:30 P.M. SECY-04-0037, "Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break LOCA Break Size and Plans for Rulemaking on LOCA with Coincident Loss-of-Offsite Power" (Open) (WJS/GBW/MRS/MDS)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding SECY-04-0037.

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

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

4) 1:30 - 3:30 P.M. Options and Recommendations for Functional Performance Requirements and Criteria for the Containments of Non-LWRs (Open) (TSK/MME)
   - 4.1) Remarks by the Subcommittee Chairman
   - 4.2) Briefing by and discussions with representatives of the NRC staff regarding proposed options and recommendations for functional performance requirements and criteria for the containments of non-light water reactors (LWRs).

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

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

5) 3:45 - 4:45 P.M. Criteria for Evaluating the Effectiveness (Quality) of the NRC Research Programs (Open) (DAP/GEA/GBW/WJS/HPN/SD)

5.1) Remarks by the Subcommittee Chairman

5.2) Discussion of the final criteria for use by the ACRS in evaluating the effectiveness (quality) of the NRC research programs.

4:45 - 5:00 P.M. ***BREAK***

6) 5:00 - 6:30 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

6.1) Action Plan for Implementing the Phased Approach for Improving PRA Quality (GEA/MRS)

6.2) SECY-04-0037, "Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break LOCA Break Size and Plans for Rulemaking on LOCA with Coincident Loss-of-Offsite Power" (WJS/GBW/MRS/MDS)

6.3) Options and Recommendations for Functional Performance Requirements and Criteria for the Containments of Non-LWRs (TSK/MME)

6.4) Criteria for Evaluating the Effectiveness (Quality) of the NRC Research Programs (DAP/GEA/GBW/WJS/HPN/SD)

6.5) Resolution of Certain Items Identified by the ACRS in NUREG-1740 Related to Differing Professional Opinion on Steam Generator Tube Integrity (FPF/GBW/BPJ)

6.6) Divergence in Regulatory Requirements Between U.S. and Several Other Countries (DAP/HPN/SD)

FRIDAY, APRIL 16, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

8) 8:35 - 10:00 A.M. License Renewal Application for the R. E. Ginna Nuclear Power Plant (Open) (MVB/MDS)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff and Rochester Gas and Electric Company regarding the license renewal application for the R. E. Ginna Nuclear Power Plant and the associated final Safety Evaluation Report prepared by the NRC staff.

10:00 - 10:15 A.M. ***BREAK***

9) 10:15 - 12:00 Noon Proposed Generic Communication Regarding Pressurizer Dissimilar Metal Weld Cracking Issues (Open) (FPF/MWW)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed NRC generic communication related to pressurizer dissimilar metal weld cracking issues.

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

12:00 - 1:15 P.M. ***LUNCH***

10) 1:15 - 1:30 P.M. Subcommittee Report on the Interim Review of the License Renewal Application for the Dresden and Quad Cities Nuclear Power Plants (Open) (MVB/BPJ)

Report by and discussions with the Chairman of the Plant License Renewal Subcommittee regarding the Subcommittee's review of the license renewal application for the Dresden and Quad Cities Nuclear Power Plants and the initial Safety Evaluation Report prepared by the NRC staff.

11) 1:30 - 1:45 P.M. Subcommittee Report on Digital I&C System Matters (Open) (JDS/GEA/MDS)

Report by and discussions with the Chairman of the Plant Operations Subcommittee regarding the Subcommittee's review of the digital instrumentation and control (I&C) system matters.

12) 1:45 - 2:30 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (MVB/JTL/SD)

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

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

13) 2:30 - 2:45 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (MVB, 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.

2:45 - 3:00 P.M. ***BREAK***

14) 3:00 - 4:30 P.M. Preparation for Meeting with the NRC Commissioners (Open) (MVB, et.al/JTL, et.al)

Discussion of proposed topics for meeting with the NRC Commissioners, which is scheduled to be held between 1:30 and 3:30 p.m. on Thursday, May 6, 2004.

4:30 - 4:45 P.M. *****BREAK*****

15) 4:45 - 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of the proposed ACRS reports on:

15.1) Action Plan for Implementing the Phased Approach for Improving PRA Quality (GEA/MRS)

15.2) SECY-04-0037, "Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break LOCA Break Size and Plans for Rulemaking on LOCA with Coincident Loss-of-Offsite Power" (WJS/GBW/MRS/MDS)

15.3) Options and Recommendations for Functional Performance Requirements and Criteria for the Containments of Non-LWRs (TSK/MME)

15.4) Criteria for Evaluating the Effectiveness (Quality) of the NRC Research Programs (DAP/GEA/GBW/WJS/HPN/SD)

15.5) Resolution of Certain Items Identified by the ACRS in NUREG-1740 Related to Differing Professional Opinion on Steam Generator Tube Integrity (FPF/GBW/BPJ)

15.6) Divergence in Regulatory Requirements Between U.S. and Several Other Countries (DAP/HPN/SD)

15.7) License Renewal Application for the R.E. Ginna Nuclear Power Plant (MVB/MDS)

15.8) Proposed Generic Communication Regarding Pressurizer Dissimilar Metal Weld Cracking Issues (FPF/MWW)

SATURDAY, APRIL 17, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

16) 8:30 - 12:30 P.M. Preparation of ACRS Reports (Open/Closed)
The Committee will continue discussion of the proposed ACRS reports listed under Item 15

17) 12:30 - 1:00 P.M. Miscellaneous (Open) (MVB/JTL)
Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

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

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

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
510th ACRS MEETING
MARCH 3-6, 2004

[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
3	Opening Remarks by the ACRS Chairman
	1. Items of Interest, dated March 3-6, 2004
4	License Renewal Application for the H.B. Robinson Steam electric Plant, Unit 2
	2. H. B. Robinson Steam Electric Plant, Unit 2 presentation by Progress Energy [Viewgraphs]
	3. H. B. Robinson Steam Electric Plant, Unit 2 License Renewal Safety Evaluation Report presentation by NRR [Viewgraphs]
5	Interim Review of the AP1000 Design
	4. AP1000 Status presentation by J. Segala, NRR [Viewgraphs]
6	License Renewal Application for the Virgil C. Summer Nuclear Station
	5. VC Summer Nuclear Station presentation by South Carolina Electric and Gas [Viewgraphs]
	6. V.C. Summer Nuclear Station License Renewal Application presentation by NRR [Viewgraphs]
7	Proposed Criteria for ACRS Evaluation of the Effectiveness (Quality) of the NRC Safety Research Program
	7. ACRS Review of Research Quality presentation by Dr. Apostolakis, ACRS Member [Viewgraphs]
11	Joint Meeting of the ACRS/ACNW with the EDO/Office Directors of NRR/RES/NMSS
	8. ACRS/ACNW Meeting with the EDO and Program Office Directors presentation [Viewgraphs]
12	Future ACRS Activities/Report of the Planning and Procedures Subcommittee
	9. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - March 13, 2004 [Handout #12.1]
13	Reconciliation of ACRS Comments and Recommendations
	10. Reconciliation of ACRS Comments and Recommendations [Handout #13.1]

Appendix V
5**XX**th ACRS Meeting

Appendix V
5**XX**th ACRS Meeting

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
Model 2	

1. Table of Contents
2. Proposed Agenda/Schedule
3. Project Status Report, dated [Internal Committee Use Only: Predecisional Material Attached]

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
510TH FULL COMMITTEE MEETING

MARCH 4-6, 2004

March ~~3~~ 4, 2004
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING

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NAME	NRC ORGANIZATION
Y. Gene Hsii	NRR/SRXB
LEN WARD	NRR/SRXB
Lambros Lois	" "
Walton Jensen	NRR / DSSA
Lauren Quinones-Navarro	NRR/DE
Edward D Throm	NRR/DSSA
Summer B. Sun	NRR/SRXB
Joseph Colaccino	NRR
Michelle Hart	NRR/DSSA
Jerry Wilson	NRR/DRIP
BOB PALLA	NRR/DSSA
Steve Bloom	NRR/DRIP
Falk Strasser	RES/DSARE
Ralph Architzel	NRR/DSSA
Andre Drozd	NRR/DSSA
Nick Saltos	NRR/DSSA/SPSB
Steve Bajorek	RES / DSARE
Jim Lyons	NRR/RNRP
Jin-Sien Guo	SPLB/DSSA/NRR

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
510TH FULL COMMITTEE MEETING

MARCH 4-6, 2004

March ~~5~~ 4, 2004
Today's Date

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NAME	NRC ORGANIZATION
DUC NGUYEN	NRR/DE/EEIB
RAM SUBBARATHAM	NRR/RLEP-B.
~~BOB WHORTON~~	
PAUL SHEMANSKI	NRR/DE/EEIB
Raj Auluck	NRR/DRIP/RLEP
J.H. Raval	NRR/DSSA/SPSB
NAEEM IQBAL	NRR/DSSA/SPLB
David Jeng	NRR/DE/EMEB
George Georgiev	NRR/DE/EMCB
BARRY J. ELLIOT	NRR/DE/EMCB
Jim Strunisha	NRR/DE/EMEB
SAM LEE	NRR/DRIP/RLEP
Y. C. (Renee) Li	NRR/DE/EMEB
Steve West	NRR/DRIP/RLEP
Ken Chang	NRR/DRIP/RLEP
Carolyn Lauron	NRR/DE/EMCB
John Honcharik	NRR/DE/EMCB
JOHN TSAO	NRR/DE/EMCB
Arnold Lee	NRR/DE/EMEB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
510TH FULL COMMITTEE MEETING

MARCH 4-6, 2004

March 4, 2004
Today's Date

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NAME	NRC ORGANIZATION
J. K. MITRA	NRR/DRIP/RLEP
P T KUO	NRR/DRIP/RLEP
Jim Strnisha	NRR/DE/EMEB
A. N. Pal	NRR/DE/EEIB
Barry Fu	NRR/DE/EMCB
SAM LEE	NRR/DRIP/RLEP
FRANCIS X. TALBOT	NRR/DIPM/IEPB
Raj Auluck	NRR/DRIP/RLEP
MICHAEL MORGAN	NRR/DRIP/RLEP
Yoira Diaz	NRR/DRIP/RLEP
Robert M Taylor	NRR/DSSA/SRXB
Joel Rivera-Ortiz	NRR/DSSA/SRXB
Y. C. (Renee) Li	NRR/DE/EMEB
Russ Arrighi	NRR/DRIP/RLEP
Joe Golla	NRR/DSSA/SPLB
SHAWN O. MARSHALL	RES/PSARE/SMSAB
Ken Chang	NRR/DRIP/RLEP
Daniel Frumkin	NRR/DSSA/SPLB
Stephanie Coffin	NRR/DE/EMCB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
510TH FULL COMMITTEE MEETING

MARCH 4-6, 2004

March 4, 2004
Today's Date

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NAME	NRC ORGANIZATION
Kimberley Cooper	NRR/DRIP/RGEP
Peter J. Kang	NRR/DRIP/RGEP
Chandu Patel	NRR/DLPM
Ralph Architzel	NRR/DSSA
HAROLD WALKER	NRR/DSSA
John Tsao	NRR/DE/EMCB
JAMES MEDOFF	NRR/DE/EMCB
Sam Miranda	NRR/DSSA/SRXB
Pei-Ying Chen	NRR/DE/EMEB
Hans Ashar	NRR/DE/EMEB
David C. Jeng	NRR/DE/EMEB
MARK HARTZMAN	NRR/DE/EMEB
Stewart Bailey	NRR/DE/EMEB
Steven Jones	NRR/DSSA/SPLB
Chang-Yang Li	NRR/DSSA/SPLB
John S. Ma	NRR/DE/EMEB
~~Ed Cummins~~	
Carolyn Lauron	NRR/DE/EMCB
Arnold Lee	NRR/DE/EMEB
John Segala	NRR/DRIP
RAJ ANAND	NRR/DRIP

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

511th FULL COMMITTEE MEETING

MARCH 4-6, 2004

March 4, 2004
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
Roger Stewart	Progress Energy
Talmage Clements	Progress Energy
Michael Heath	Progress Energy
Jan Kozyra	Progress Energy
Jan Fridrichsen	Southern Nuclear
George Wrobel	RGRS
Joseph W Donohue	Progress Energy
Susan G. Sterrett	Duke University
~~Katsu~~ Ohkawa	Westinghouse
Ed Cummins	Westinghouse
Jim Scobel	Westinghouse
Cesare Frepoli	Westinghouse
Melissa Jenkins	NRC
Terry Schulz	Westinghouse
Spyros Traiforos	LINK
James CLaBorde	SCE&G
Tyndall Estes	SCE&G
Mike Dantzler	SCE&G
Stan Crumbo	SCE&G

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

511th FULL COMMITTEE MEETING

MARCH 4-6, 2004

March 4, 2004
Today's Date

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NAME	AFFILIATION
Robert Gold	W-NSBU
Ronald Vijuk	Westinghouse
KATSU OHKAWA	WESTINGHOUSE
CESARE FREPOLI	WESTINGHOUSE
Ron Clary	SCE&G
Ross Arrighi	NRR/DRIP/RLEP
S. K. Mitra	NRR/DRIP/RLEP
George Wrobel	Rochester Gas + Electric

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
510TH FULL COMMITTEE MEETING

MARCH 4-6, 2004

March ~~6~~ 5, 2004
Today's Date

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NAME	NRC ORGANIZATION
TILA LIU	NRR/DRIP
Steven Jones	NRR/DSSA/SPLB
Sam Miranda	NRR/DSSA/SRXB
CHENG-IH (JOHN) WU	NRR/DE/EMEB
John S. Ma	NRC/DE/EMEB
M Hartzman	NRC/DE/EMEB
Richard McNally	NRC/NRR/DE/EMEB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
510TH FULL COMMITTEE MEETING

MARCH 4-6, 2004

March ~~4~~ 5, 2004
Today's Date

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NAME	NRC ORGANIZATION
ALAN LEVIN	RED/OD
Larry Rossbach	NRR/DLPM/PD3-2
Thomas Scarbrough	NRR/DE/EMEB
Rabelle Schonfeld	OEDO
Eric Benner	NRR/OD
Michael Waters	NMSS
Mary Drouin	RES
JACK STROSNIDER	RES
Chris Grimes	NRR/DE
Carl J Paperiello	OEDO
Derek Widmayer	NMSS | DWM
Suzanne Black	NRR/DSSA
David Terao	NRR/DE/EMEB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
510TH FULL COMMITTEE MEETING

MARCH 4-6, 2004

March ~~6~~ 5, 2004
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING

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NAME	NRC ORGANIZATION
Anthony McMurtray	NRR/DLPM

1

ITEMS OF INTEREST

510th ACRS MEETING

MARCH 3-6, 2004

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
510th MEETING
MARCH 3-6, 2004

Page

EDO UPDATES

- EDO Update Email to NRC Staff, dated March 1, 2004, 11:37 a.m. 1

CORRESPONDENCE

- NRC Response to a Letter to Chairman Nils J. Diaz, from Readers on Eye on Wackenhut, Regarding Security at Nuclear Power Plants, dated February 5, 2004 . 2-6

NRC PRESS RELEASES

- NRC to Conduct Special Inspection at Palo Verde Nuclear Generating Station, dated March 1, 2004 7
- NRC Issues Letter on Planned Davis-Besse Order, dated February 26, 2004 8-10

OPERATING PLANT ISSUES

- Letter from James L. Caldwell, Regional Administrator, /s/ by Geoffrey E. Grant to Mr. William R. Kanda, Vice President-Nuclear, Perry. Subject: Final Significance Determination for a White Finding for Perry Nuclear Plant, dated January 28, 2003 11-14
- Letter from Hubert J. Miller, Regional Administrator, /s/ by James T. Wiggins, to Mr. Christopher M. Crane, President and CNO Exelon Nuclear. Subject: Final Significance Determination for a White Finding and Notice of Violation for Peach Bottom Atomic Power Station, Units 2 and 3, dated February 3, 2004 15-18

INSIDE NRC ARTICLES

- Article titled, "BWROG tells NRC About Plans to Address Power Uprate Problems", Volume 26/Number 3 dated February 9, 2004 19-22
- Article titled, "NRC Calls for Safety Culture Probe by PSEG at Hope Creek, Salem," Volume 26/Number 3 dated February 9, 2004 23-27
- Article entitled, "NRC Critical of NEI's Draft Sump Evaluation Guidance," Volume 26/Number 4 dated February 23, 2004 28-30
- Article entitled, "'One size fits all' said not appropriate for SCWE best practices guidance," Volume 26/Number 4 dated February 23, 2004 31-33
- Article entitled, "IG says NRC overlooking key issue in response to Davis-Besse findings," Volume 26/Number 4 dated February 23, 2004 34-36

FACT SHEET

- Article on NRC Website entitled, "The Accident at Three Mile Island" (http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/3mile-isle.html) 37-44

From: William Travers
To: William Travers
Date: 3/1/04 11:38 AM
Subject: EDO Updates

EDO Update, 3/1, 11:37 a.m.

In addition to the public forum that NRC is hosting to commemorate the 25th anniversary of the Three Mile Island accident on March 3, the Smithsonian is sponsoring a series of programs. On March 26 at noon, at the Museum of American History, NRC historian Samuel Walker will discuss his book on the history of the event, how the crisis was handled, and its subsequent implications. On March 28 at 2:00 p.m., the actual anniversary of the accident, the museum will host a panel discussion with Richard Thornburgh, the governor of Pennsylvania at the time of the accident, Harold Denton, NRC's former Director of Nuclear Reactor Regulation, and Jessica Tuchman Mathews, former chief of the Office of Global Issues at the National Security Council. An exhibit on TMI will be on display at the museum through April 30. On March 11, the Regulatory Information Conference will also feature a panel presentation and discussion of the accident with Chairman Diaz, Harold Denton, Oliver Kingsley (President and Chief Operating Office of Exelon), former Pennsylvania Governor Thornburgh, and myself.

A focus of all of the NRC events related to the accident is to communicate how the agency has matured and improved over the years, and to demonstrate the **excellence** and **commitment** of the NRC staff in developing strong new initiatives focused on ensuring safety.

The closed Commission meeting with the industry on security issues last week went well. In classified sessions, we discussed the aircraft vulnerability studies and mitigative strategies for reactors. The hard work performed by RES and NSIR on this meeting was evident and is greatly appreciated.

As we progress toward a restart decision for Davis-Besse, I want to thank just about all the staff - particularly Region III - for an excellent job and for showing the American public that our **service** and **integrity** in adhering to our safety mission is strong. This event reminds us only too clearly of the need for continued vigilance in licensee and NRC actions for current operating reactors. There is very little room for complacency. The Lessons Learned Task Force found that enhancements are needed and reinforced our belief that evolutionary change is far preferable to revolutionary change.

I will be testifying at a hearing on March 4 before the Senate Energy and Natural Resources Committee's Subcommittee on Energy, focusing on NRC's role in any new nuclear power generation in the U.S. and the recovery of Browns Ferry Unit 1. Ellis Merschoff, CIO, may also be testifying at a March 10 hearing on our computer security achievements. Also possible is a House Committee on Energy and Commerce hearing on high-level radioactive waste (date uncertain).

I want everyone to be aware that travel money is tight and we are going to have to work hard to manage staff travel. This includes prioritizing trips that are directly mission-related, limiting trips abroad and reducing the number of staff accompanying the principal traveler. So please, let's all be conservative and selective when we assess the need for travel.

Finally, Pat Norry was interviewed by WTOP radio for their "Ask the Chief Human Capital Officer" program. The discussion focused on human capital challenges facing federal agencies in the years ahead. The entire interview will be available this morning beginning at 10:30 a.m. on www.Federalnewsradio.com

NRC RESPONSE TO A LETTER TO NRC CHAIRMAN, NILS J. DIAZ FROM READERS OF EYE ON WACKENHUT (A Web site hosted by the Service Employees International Union) REGARDING SECURITY AT NUCLEAR POWER PLANTS

INTRODUCTION

The Nuclear Regulatory Commission (NRC) has received a large number of electronic mail messages and facsimiles containing a letter addressed to NRC Chairman Nils J. Diaz. The letter raises concerns about security at nuclear power plants. These communications began in November 2003. In general, the issues raised in the letter are dated. Many significant actions have been taken to enhance the security at NRC-licensed facilities since September 11, 2001. Most of the concerns raised in these letters involve security at the Indian Point nuclear power plant, Unit 2, located in Buchanan, New York. The NRC addressed all of these concerns and has concluded that security at Indian Point Unit 2 is adequate to ensure protection of the public health and safety and the common defense and security. The following NRC response is presented in two parts. The first part describes the substantial enhancements in security since early 2002. The second part responds to the specific issues raised in these letters, including those related to Indian Point Unit 2.

GENERAL RESPONSE

For over 25 years, the NRC has required that major NRC licensees maintain security programs. As a result of the September 2001 terrorist attacks, the NRC launched a comprehensive evaluation of the security and safeguards programs at nuclear power plants, nuclear materials and waste facilities, and radioactive material transportation activities.

The NRC has issued orders to licensees requiring enhancements designed to raise the level of security at nuclear power reactors by upgrading security in the areas of physical protection, access authorization (including improved background checks), security force training and qualification, security force work hours (fitness for duty), and protection against a revised design basis threat (DBT). The DBT is characterized by the type, composition, and capabilities of an adversary. The DBT is used to design safeguards systems to protect against acts of radiological sabotage and to prevent the theft of special nuclear material. Many of these enhancements had already been put in place voluntarily by licensees; however, the orders provided the means to make them legally binding and to ensure consistent implementation.

Force-on-force exercises (simulated commando-style attacks on nuclear power plants) are conducted to assess and improve, as necessary, performance of defensive strategies at licensed facilities. These exercises were temporarily suspended immediately following the terrorist attacks of September 11, 2001, because such exercises would have been a significant distraction to licensee security forces which were at the highest level of security alert. In February 2003, the NRC decided to establish an expanded Force-on-Force exercise pilot program. The pilot force-on-force exercises are aimed at reducing artificialities, thereby increasing the realism of the exercises and improving NRC's processes for assessing the licensees' readiness to respond to the DBT. In resuming these exercises, the NRC also increased the exercise frequency at nuclear power reactor facilities from once every eight years to once every three years.

SPECIFIC RESPONSE

Issue 1) Security Procedures

NRC Response:

The NRC has verified, through its inspection program, that Indian Point security procedures are adequate to meet the NRC's regulatory requirements. The Indian Point security has been enhanced as required by the February 25, 2002 Orders to all licensees of nuclear power plants. During the last several years, NRC security specialists from Region I and the NRC Office of Nuclear Security and Incident Response have devoted hundreds of hours to inspecting the security arrangements at Indian Point, including confirmation of the effectiveness of Entergy's actions to comply with the Additional Security Measures that were ordered on February 25, 2002. To date, these inspection activities have not revealed significant deficiencies in the implementation and execution of the physical protection and security program at Indian Point. Specific details of the Orders are Safeguards Information and, therefore, cannot be disclosed to the public or other entities who are not authorized and who lack an official need to know the information.

Issue 2) Security Officers

NRC Response:

The Entergy report issued on January 25, 2002, regarding security at Indian Point concluded that only 19% of the security officers interviewed stated that they could adequately defend the plant against a terrorist attack in light of the uncertainty following the terrorist attacks of September 11, 2001. In the weeks and months directly following the World Trade Center and Pentagon attacks, the uncertainty of the nature of the potential threats against nuclear plants caused concern among many, including security officers across the country. NRC has conducted interviews of security officers and has initiated reviews of the specific concerns in accordance with its Reactor Oversight Process inspection program and allegation process. Some security officers have voiced concerns about their ability to defend against a perceived level of threat that exceeds the current security planning basis. However, in general, security personnel at the site interviewed by NRC since the Entergy report was issued were confident in their ability to implement the heightened security program currently in place. In July 2003, the NRC had more than 20 staff and expert contractors overseeing the force-on-force security exercise at Indian Point. The exercise was also observed by the FBI, New York State Office of Public Security, and other State and local officials. There has been, and continues to be, excellent support from local, State, and Federal authorities, including on-site National Guard, Coast Guard, and local law enforcement officers. NRC's observations at Indian Point indicate that the licensee has a strong defensive strategy and capability. The Indian Point security force personnel successfully protected the plant from repeated mock-adversary attacks during the exercise. These observations and ongoing NRC oversight support the conclusion that public health and safety continues to be adequately protected at Indian Point.

The NRC staff believes that security force members have provided, and continue to provide, valuable first-hand input to the assessment of the adequacy of nuclear facility security. Their input has been important to the NRC's development of additional requirements on training and fitness for duty that were imposed in April 2003. Their input is only a portion of the security assessment information that has been assimilated since the events of September 11, 2001.

The NRC will continue to evaluate all relevant input from stakeholders as NRC continues to review and enhance, as appropriate, the safeguards and security programs.

Issue 3) Physical agility training

NRC Response:

In December 2002 and January 2003, NRC conducted several reviews and inspections of security program performance at Indian Point, including a comprehensive verification of the completion and effectiveness of the additional security measures required by the Order of February 25, 2002. The reviews and inspections confirmed that Entergy was implementing the physical protection and security program in accordance with the specifications of the NRC-approved Indian Point Security Plan and Entergy's Training and Qualification Plan.

Requirements for physical and mental qualifications for security personnel at nuclear power plants are defined in the licensees' NRC-approved Training and Qualification Plans. These requirements comply with 10 CFR Part 73, Appendix B and include physical fitness tests representative of actions required by the site response strategy. NRC enhanced requirements for training and qualification of the security force in an Order in April 2003.

The NRC reviews of Indian Point security personnel records have indicated that all officers have passed the required physical agility tests and meet the qualification requirements. Many officers would like more training on tactics and weapons proficiency and some believe that qualification requirements should be more challenging. The NRC considered these views in developing and imposing the required enhancements to training and qualifications in April 2003. These requirements are designed to ensure security personnel are capable of responding to an event and are prepared to respond effectively to a terrorist attack.

Issue 4) Qualifying Examinations for Carrying Weapons

NRC Response:

The NRC has no information to indicate that qualifying exams for carrying weapons had been rigged so officers could pass. In order to assess concerns regarding weapons qualification, NRC inspectors observed evaluations of security officers on the firing range during a January 2003 team inspection. The Entergy weapons re-qualification process, which is typical of accepted industry practice, allows familiarization firing before starting the re-qualification testing. If the individual does not qualify in the first cycle, the individual is allowed to perform a second cycle. If still not qualified, the individual is removed from armed response duties, provided remedial training by a range instructor, and allowed to attempt one more qualifying cycle at the next re-qualification, usually within 30 days. In accordance with current Entergy policy, an individual who fails to re-qualify can no longer be employed as an armed responder.

The re-qualification process for the challenge test or watchperson test described in the January 2002 Entergy report is similar to weapons re-qualification. It allows two chances prior to remediation and a final chance following remediation. In all cases, officers would not be allowed to return to armed response duties until they successfully passed the re-qualification test.

Issue 5) Information Provided by Wackenhut to Plant Management and NRC Investigators

NRC Response:

The NRC reviewed the Entergy report of January 25, 2002, with respect to the statements and conclusions regarding the March 2001 Wackenhut report. NRC has also reviewed the licensee's responses to the NRC letter which asked Consolidated Edison (the licensee at the time), in the aftermath of the termination of the security officer, to explain its actions, taken or planned, to ensure there was not a reluctance (chilling effect) on the part of security officers to raise safety issues at Indian Point due to fear of retribution.

The NRC concluded that the Wackenhut report (as described in the January 2002 Entergy report) provided an accurate account regarding the issue of a chilled environment with respect to raising nuclear safety concerns. As indicated in the January 2002 Entergy report, the officers' responses to Wackenhut interviewers in March 2001 were consistent with their responses to the independent contractor in November and December 2001 regarding the willingness to raise safety concerns. NRC reviews of the Entergy report indicate that the assertions that officers at Indian Point are discouraged from raising concerns apply to issues not directly related to nuclear safety issues. The Entergy report indicates that some of the officers interviewed in November and December 2001 did not feel comfortable raising certain concerns, such as labor issues, to management. However, the officers indicated that this reservation did not apply to raising nuclear safety issues. According to the Entergy report, the officers provided similar responses regarding the chilled environment and their willingness to raise safety concerns to the Wackenhut interviewers in March 2001. The Entergy report also indicates that the Wackenhut report acknowledged the concerns with management raised by the officers. Therefore, the NRC concluded that the Wackenhut report did not contain false information regarding the issue of a chilled environment with respect to raising nuclear safety concerns.

Issue 6) Wackenhut's Ability to Provide Security

NRC Response:

Since March 29, 2003, security at the Indian Point facility is provided by Entergy employees. Based on NRC inspection findings to date and the ongoing review of Entergy's response to NRC threat advisories and Orders, the NRC staff considers security at Indian Point to be adequate to protect the public health and safety and the common defense and security. The number of available security responders at the Indian Point facility has been substantially increased since September 11, 2001. Additionally, Entergy has taken significant steps since the September 11, 2001, attacks to strengthen physical barriers, security equipment, and response strategies at the facility. Security personnel interviewed at the site in 2002 and early 2003 by NRC staff indicated that they were comfortable with their ability to implement the current heightened security program requirements.

The NRC inspection program is designed to verify compliance with the regulatory requirements, regardless of whether the security force is composed of licensee employees or contractor employees. The NRC conducts inspections that evaluate the effectiveness of security program performance and include observations of the security force members and their supervisors. Although the NRC interacts with contractor personnel, the NRC holds licensees, not contractors, accountable for security performance.

Issue 7) Security officers who have raised questions about security vulnerabilities

NRC Response:

The NRC has an established process for reviewing cases where discrimination is alleged. When NRC receives information indicating that an individual may have been harassed or intimidated due to raising safety concerns at any NRC licensed facility, or about any potential security vulnerabilities, the NRC evaluates the situation and investigates as warranted by the circumstances in accordance with established procedures. To date, the NRC has not substantiated any specific examples of security officers having been discriminated against for having raised safety concerns at Indian Point.

Issue 8) Security-related employee allegations

NRC Response:

There is no direct relationship between the number of concerns and alleged or substantiated cases of discrimination received from nuclear power plant security employees. In the normal course of business, the NRC receives allegations from nuclear power plant employees and others. The NRC evaluates the validity of those allegations through an established allegations review process and takes appropriate action. Sometimes that action results in an in-depth plant review or review of associated programs.

Issue 9) Wackenhut Corporation's Track Record

NRC Response:

Beginning March 29, 2003, security at the Indian Point facility is no longer provided by contractor employees but by Entergy itself. The NRC was in no way involved in that decision, but still requires that Entergy and other licensees meet the NRC safety and security requirements stipulated in the regulations. The NRC inspection program is designed to verify compliance with the regulatory requirements regardless of whether the security force is composed of licensee employees or contractor employees. The NRC conducts inspections that evaluate the effectiveness of security program performance and include observations of the security force members and their supervisors. Although the NRC interacts with contractor personnel, the NRC holds licensees, not contractors, accountable for security performance.

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs, Region IV
611 Ryan Plaza Drive, Suite 400, Arlington TX 76011
www.nrc.gov

No. IV-04-007
CONTACT: Victor Dricks
Phone: 817-860-8128

March 1, 2004
E-mail: opa4@nrc.gov

NRC TO CONDUCT SPECIAL INSPECTION AT PALO VERDE NUCLEAR GENERATING STATION

The Nuclear Regulatory Commission has begun a special inspection to evaluate problems related to a new steam generator and the operation of the shutdown cooling system at the Palo Verde Nuclear Generating Station, located 50 miles west of Phoenix, Arizona.

On February 19, operators shut down Palo Verde Unit 2 after monitors detected a minute leak in one of two steam generators that had been replaced last fall. Despite some problems subsequently encountered, the plant remained in a safe condition at all times and there was no danger to public health or safety. After the reactor cooled, operators reduced the level of water in the reactor coolant system to facilitate access to the leaking steam generator. However, problems with some equipment led them to prolong the time that the plant remained in this condition. Workers discovered that air had displaced some of the water in the reactor shutdown cooling system, forcing them to open valves to vent air into the auxiliary building every two hours.

The NRC staff has decided to conduct a special inspection to evaluate the adequacy of the licensee's response to the situation, the root cause, and corrective actions.

The NRC's Special Inspection Team, consisting of two reactor engineers from the NRC's Region IV Office in Arlington, Texas, an inspector from the Callaway nuclear plant in Missouri, and a Headquarters specialist, arrived on site last week and have begun their review.

The inspection report will be issued about four weeks after the inspection is completed, and will be available on the agency's web site and through its Electronic Reading Room at: http//www.nrc.gov as an Agencywide Document Access and Management System (ADAMS) document. Help in using ADAMS is available through the NRC Public Document Room at 301-417-4737 or 1-800-397-4209.

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs, Region III
801 Warrenville Road, Lisle IL 60532
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No. III-04-009
CONTACT: Jan Strasma (630) 829-9663
Viktoria Mitlyng (630) 829-9662

February 26, 2004
E-mail: opa3@nrc.gov

NRC ISSUES LETTER ON PLANNED DAVIS-BESSE ORDER

Printable Version

The Nuclear Regulatory Commission has issued the attached letter to FirstEnergy Nuclear Operating Company concerning an Order that the agency intends to issue requiring independent assessments and mid-cycle inspections at the Davis-Besse Nuclear Power Station to provide reasonable assurance that the long-term corrective actions remain effective.

FirstEnergy may inform the NRC whether it will consent to the conditions of the Order and agree to have those conditions incorporated into a Confirmatory Order that will be immediately effective upon issuance.

The letter does not imply that permission from the NRC to restart will be forthcoming. However, NRC restart approval is contingent upon the conditions in the Order being in effect. The NRC plans to issue the Order regardless of the utility's consent.

Attachment: Letter to FirstEnergy dated February 26, 2004.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
801 WARRENVILLE ROAD
LISLE, ILLINOIS 60532-4351

February 26, 2004

EA-03-214

Mr. Lew W. Myers
Chief Operating Officer
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5[illegible] North State Route 2
C[illegible]rbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1 - CONDITIONS TO BE CONFIRMED BY ORDER MODIFYING LICENSE (EFFECTIVE IMMEDIATELY) (TAC NO. MC1335)

Dear Mr. Myers:

As a result of FirstEnergy Nuclear Operating Company (FENOC) identification of extensive degradation of the reactor pressure vessel head, the Nuclear Regulatory Commission (NRC) issued Confirmatory Action Letter (CAL) No. 3-02-001 on March 13, 2002. The CAL documented commitments that FENOC was expected to fulfill prior to restart of the Davis-Besse Nuclear Power Station, Unit 1. These commitments included meeting with the NRC to obtain restart approval.

On November 23, 2003, FENOC submitted the "Integrated Report to Support Restart of the Davis-Besse Nuclear Power Station and Request for Restart Approval," and concluded that the plant, programs, and personnel were ready to support safe operation, subject to completion of a few well-defined work activities prior to restart. That letter outlined FENOC's commitments regarding post-restart continuing improvement initiatives and self and external assessments to assure lasting performance improvement and requested that the NRC schedule a meeting as stated in the CAL and then approve restart. A meeting was originally scheduled for December 18, 2003, to discuss restart. However, due to self-revealing equipment and operational problems and issues identified by NRC Restart Readiness Assessment and the Management and Human Performance inspection teams, the meeting was delayed.

During the course of the extended shutdown of Davis-Besse beginning in February 2002, FENOC conducted a number of thorough evaluations and self-assessments. Several examples include the evaluation of system design, the assessment of the completeness and accuracy of docketed information, the evaluation of operational performance deficiencies during the normal operating pressure test, and the evaluation of the failure to comply with technical specification requirements during testing of the steam and feedwater rupture control system. However, self-assessments of operational performance prior to both the normal operating pressure test in September 2003, and the NRC's Restart Readiness Assessment Team Inspection in December 2003 failed to identify a number of deficiencies. Additionally, NRC inspections earlier during the shutdown discovered issues not originally identified through Davis-Besse self-assessments, most notably in the corrective action program, in the quality of engineering evaluations, calculations and analyses, and in safety culture. The NRC recognizes that FENOC has implemented significant corrective actions resulting in improved performance to address the CAL and the NRC Davis-Besse Oversight Panel Restart Checklist. Notwithstanding the improved performance, consistently effective FENOC self-assessments are an important factor in assuring lasting corrective actions for the deficiencies that resulted in the reactor pressure vessel head degradation.

To ensure effective assessment and sustained safe performance in the areas of operations, engineering, and corrective actions at Davis-Besse, the NRC has determined that additional measures are needed to provide the requisite assurance should restart of Davis-Besse be authorized. Therefore, the NRC will issue an Order modifying License No. NPF-3, requiring independent assessments and mid-cycle inspections to provide reasonable assurance that the long-term corrective actions remain effective.

FENOC may inform the NRC whether it will consent to the enclosed conditions by providing written response to the Regional Administrator at NRC Region III, 801 Warrenville Road, Lisle, IL 60532-4351, within five working days of the date of this letter. If you consent, I request that you sign the enclosed Consent and Hearing Waiver form and return it to the above address. By signing the enclosed form, the management of FENOC will agree to have those conditions incorporated into a Confirmatory Order that will be immediately effective upon issuance and FENOC will waive any and all rights to a hearing concerning the Order.

This letter does not imply that permission from the NRC to restart will be forthcoming. However, NRC restart authorization is contingent upon the conditions in the Order being in effect. The NRC plans to issue the Order regardless of your consent.

If you have any questions, please call me at 630-829-9657.

Sincerely,

/signed/

James L. Caldwell
Regional
Administrator

Docket No. 50-346

Enclosures: 1. Consent and Hearing Waiver Form [Not part of attachment to news release]
2. Conditions to be confirmed by Order

Enclosure to letter to Mr. Lew Myers dated February 26, 2004:

CONDITIONS TO BE CONFIRMED BY ORDER

Accordingly, pursuant to Sections 103, 161b, 161i, 161o, 182 and 186 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR 2.202 and 10 CFR Part 50, IT IS HEREBY ORDERED, EFFECTIVE IMMEDIATELY, THAT LICENSE NO. NPF-3 IS MODIFIED AS FOLLOWS:

1. FENOC shall contract with independent outside organizations to conduct comprehensive assessments of the Davis-Besse operations performance, organizational safety culture, including safety conscious work environment, corrective action program implementation, and engineering program effectiveness. Ninety days prior to the assessments, FENOC shall inform the Regional Administrator, NRC Region III, in writing, of the identity of its outside assessment organizations, including the qualifications of the assessors, and the planned scope and depth of the assessments. These outside independent assessments at Davis-Besse shall be completed before the end of the 4th calendar quarter of 2004 and annually thereafter for 5 years. Within 45 days of completion of the assessments, the Licensee shall submit by letter to the Regional Administrator, NRC Region III, all assessment results and any action plans intended to address issues raised by the assessment results.

2. FENOC shall conduct a visual examination of the reactor pressure vessel upper head bare metal surface, including the head-to-penetration interfaces; the reactor pressure vessel lower head bare metal surface, including the head-to-penetration interfaces; and the control rod drive mechanism flanges, using VT-2 qualified personnel and procedures during the Cycle 14 midcycle outage. The results and evaluation of the inspections will be reported by letter to the Regional Administrator, NRC Region III, prior to restart from the mid-cycle outage, and any evidence of reactor coolant leakage found during the inspections will be reported by telephone within 24 hours of discovery to the Regional Administrator, NRC Region III, or designee.

If the Licensee determines that submittals made in accordance with these conditions contain proprietary information as defined by 10 CFR 2.390, the Licensee shall also provide a nonproprietary version in accordance with 10 CFR 2.390(b)(1)(ii). The Regional Administrator, NRC Region III, may, in writing, relax or rescind any of the above conditions upon demonstration by the Licensee of good cause.

EA-03-197 - Perry 1 - (FirstEnergy Nuclear Operating Company)

January 28, 2004

EA-03-197

Mr. William R. Kanda
Vice President - Nuclear, Perry
FirstEnergy Nuclear Operating Company
P. O. Box 97, A210
10 Center Road
Perry, OH 44081

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING (NRC INSPECTION REPORT 50-440/2004-005)

Dear Mr. Kanda:

The purpose of this letter is to provide you with the final results of our significance determination of the preliminary White finding associated with the A emergency service water (ESW) pump discussed in Inspection Report 50-440/2003-006, dated October 30, 2003. The inspection finding was assessed using the Significance Determination Process and was preliminarily characterized as White (i.e., a finding with low to moderate increased importance to safety, which may require additional NRC inspection). This preliminary White finding was identified on September 1, 2003, when the ESW A pump shaft failure occurred at the Perry Nuclear Plant.

At your request, a Regulatory Conference was held on December 17, 2003, to further discuss your views on this issue. At the Regulatory Conference, your staff presented an overview of the event, related corrective actions, and the methodology and results of your independent safety assessment of the preliminary White finding. A copy of your slide presentation is enclosed. Your staff's presentation also included a discussion, based upon the additional information presented, that it is your view that the finding would be more properly characterized as Green, that is, a finding of very low safety significance. The NRC and Perry staffs held extensive discussions regarding specific technical issues related to your analysis. You provided the details of your risk analysis to us in a letter dated December 23, 2003.

Following our review of your information, we determined that the results of your risk analysis were not compelling enough to change our preliminary position that the finding is White. Your initial determination, as stated in LER 2003-004-00, concluded the finding was White consistent with our preliminary assessment. Subsequently, in the regulatory conference and your submittal of December 23, 2003, you shared with us your analysis that the risk significance of the finding was reduced by partitioning the risk and by addressing conservatisms in the PSA model and in the specific condition assessment. In general, your analysis supported your conclusion that the overall risk was reduced below the Green/White threshold. We acknowledge the validity of the points you made in your analysis. However, your approach appeared to be limited to considering only those factors where the overall risk could be decreased. It did not consider other factors where the overall risk could be increased. As a check, we performed a similar assessment for Large Early Release Frequency (LERF), utilizing your CDF reduction factor and the NRC's risk model, and determined the finding would be above the Green/White threshold. The overall results of your calculations were approximately 9.7XE-7, or just below the Green/White threshold of 1XE-6. The NRC's analysis concluded the significance value was 2.7XE-6. Considering the two analytic approaches and their differences, we determined that the issue should be considered a White finding.

Therefore, after considering the information developed during the inspection; the additional information you provided in

your letter dated December 23, 2003; and the information you provided at the conference; the NRC has concluded that the inspection finding is appropriately characterized as White, that is, an issue with low to moderate increased importance to safety, which may require additional NRC inspection.

You have 30 calendar 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 0609, Attachment 2.

The NRC has also determined that the failure to develop a procedure that adequately incorporated instructions for proper reassembly of the ESW pump is a violation of Technical Specification 5.4, as cited in the enclosed Notice of Violation (Notice). The circumstances surrounding the violation are described in detail in Inspection Report 05000440/2003-006. In accordance with the NRC Enforcement Policy, NUREG-1600, the Notice of Violation 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.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if any, will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, 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**.

Sincerely,

RA by Geoffrey E. Grant for/

James L. Caldwell
Regional Administrator

Docket No. 50-440
License No. NPF-58

Enclosure:
1. Notice of Violation
2. Licensee Presentation
3. Regulatory Conference Attendance List

cc w/encl:
G. Leidich, President - FENOC
K. Cimorelli, Acting Director,
Maintenance Department
V. Higaki, Manager, Regulatory Affairs
J. Messina, Director, Nuclear
Services Department
T. Lentz, Director, Nuclear
Engineering Department
T. Rausch, Plant Manager,
Nuclear Power Plant Department
M. O'Reilly, Attorney, First Energy
Public Utilities Commission of Ohio
Ohio State Liaison Officer
R[illegible], Ohio Department of Health

NOTICE OF VIOLATION

ergy Nuclear Operating Company
Nuclear Power Plant, Unit 1

Docket No. 50-440
License No. NFP-58
EA-03-197

During an NRC inspection conducted from July 1 through September 30, 2003, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

> Technical Specification 5.4 requires, in part, that procedures shall be established, implemented and maintained as recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, Section 9, "Procedures for Performing Maintenance," recommends, in part, that maintenance activities that can affect the performance of safety-related equipment be performed in accordance with written procedures appropriate to the circumstances. The licensee developed Procedure GMI-0039, "Disassembly of the Emergency Service Water Pumps," to disassemble and reassemble the Division 1 and 2 emergency service water pumps. This procedure provides direction for the coupling of the pump shaft sections during reassembly.
>
> Contrary to the above, the licensee failed to establish written procedures appropriate to the circumstances. Specifically, the vendor procedure specified that the setscrew holes should align with the groove in the slip ring such that the setscrews should be flush with the coupling sleeve. The licensee failed to transfer this requirement to the procedure for pump reassembly resulting in improper alignment of the coupling components when the pump was reassembled on July 24, 1997, using GMI-0039, Revision 3. As a result, the pump subsequently failed on September 1, 2003.

This violation is associated with a White Significance Determination Process finding.

Pnt to the provisions of 10 CFR 2.201, FirstEnergy Nuclear Operating Company is hereby required to submit a written sent or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region III, and a copy to the NRC Resident Inspector at the Perry Nuclear Power Plant, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-03-197" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

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

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(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 rdance with 10 CFR 19.11, you may be required to post this Notice within two working days.

Dated this 28th day of January 2004

EA-03-224 - Peach Bottom 2 & 3 (Exelon Generation Co., LLC)

February 3, 2004

EA 03-224

Mr. Christopher M. Crane
President and CNO
Exelon Nuclear
200 Exelon Way
KSA3-E
Kennett Square, PA 19348

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION
(NRC Inspection Report No. 50-277/03-13 and 50-278/03-13)
Peach Bottom Atomic Power Station, Units 2 and 3

Dear Mr. Crane:

The purpose of this letter is to provide you with the final results of our significance determination for the preliminary White finding identified at Unit 2 during an inspection completed on November 18, 2003. The results of the inspection were discussed with you and other members of your staff at an exit meeting on November 18, 2003. The inspection finding was assessed using the significance determination process and was preliminarily characterized as White, a finding with low to moderate importance to safety, which may require additional NRC inspections.

This preliminary White finding involved issues related to the E2 emergency diesel generator (EDG), one of four EDGs that constitute the highly risk important standby emergency AC power system. Specifically, implementation of a deficient maintenance procedure during installation of EDG cylinder liner adapter gaskets in 1992 eventually allowed combustion gas to leak into the jacket water cooling system over time. This condition ultimately led to a low jacket water pressure condition and an automatic trip of the EDG on September 15, 2003. In addition, corrective actions taken when low jacket water pressure conditions were observed on the E2 EDG in March and April 2003, were inadequate because an action request was closed without resolving the low jacket water pressure condition, which was indicative of a continuing EDG performance problem.

In a letter dated December 18, 2003, the NRC transmitted the referenced inspection report and provided you an opportunity to either request a regulatory conference to discuss this finding, or explain your position in a written response. In a telephone conversation on January 6, 2003, Mr. M. Gallagher, Director, Licensing and Regulatory Affairs, informed Mr. R. Crlenjak, NRC, Region I, that Exelon did not contest the risk significance of this finding, declined an opportunity to discuss this finding in a Regulatory Conference, and would not be providing a written response prior to issuance of this Final Significance Determination.

After considering the information developed during the inspection, the NRC has concluded that the inspection finding at Unit 2 is appropriately characterized as White, an issue with low to moderate importance to safety, which may require additional NRC inspections. You have 30 calendar 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, Attachment 2.

The NRC has also determined that the White finding resulted in two violations, as described in the enclosed Notice of

Violation (Notice). The circumstances surrounding these violations were also described in detail in the subject inspection report. In accordance with the NRC Enforcement Policy, NUREG-1600, the Notice of Violation is considered escalated enforcement action because it is associated with a White finding.

The NRC has concluded that information regarding the reason for the violations, the corrective actions taken and planned to correct the violations and prevent recurrence, and the date when full compliance was achieved are already adequately addressed on the docket as summarized in NRC Inspection Report 50-277/03-13; 50-278/03-13 dated December 18, 2003. Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

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

In addition, although the deficiency associated with the E2 EDG affected both Peach Bottom units, the NRC has concluded that the inspection finding for Unit 3 is appropriately characterized as Green, an issue of very low safety significance. The finding for Unit 3 is less risk significant than Unit 2 because there are fewer safety-related electrical loads powered by the E2 EDG on Unit 3.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room). The NRC also includes significant enforcement actions in its Web site at www.nrc.gov; select **What We Do**, **Enforcement**, then **Significant Enforcement Actions**.

Sincerely,

/RA/ James T. Wiggins ***Acting For/***

Hubert J. Miller
Regional Administrator

Docket Nos: 50-277, 50-278
License Nos: DPR-44, DPR-56

Enclosure: Notice of Violation

cc w/encl:
President and CNO, Exelon Nuclear
Chief Operating Officer, Exelon Generation Company,
Site Vice President, Peach Bottom Atomic Power Station
Plant Manager, Peach Bottom Atomic Power Station
Regulatory Assurance Manager - Peach Bottom
Senior Vice President, Nuclear Services
Vice President, Mid-Atlantic Operations
Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
Director, Licensing, Exelon Generation Company, LLC
Manager, Licensing - Limerick and Peach Bottom
Vice President, General Counsel and Secretary
Correspondence Control Desk
Manager License Renewal
D. Quinlan, Manager, Financial Control, PSEG
R. McLean, Power Plant and Environmental Review Division
D[illegible]n, Acting Secretary of Harford County Council
M[illegible] Mrs. Dennis Hiebert, Peach Bottom Alliance
Mr. & Mrs. Kip Adams

D. Allard, Director, Pennsylvania Bureau of Radiation Protection
R. Janati, Chief, Division of Nuclear Safety, Pennsylvania Bureau of Radiation Protection
Director, Nuclear Training
TMI Alert (TMIA)
Board of Supervisors, Peach Bottom Township
R. Fletcher, Department of Environment, Radiological Health Program
J. Johnsrud, National Energy Committee, Sierra Club
Public Service Commission of Maryland, Engineering Division
J. Bradley Fewell, Assistant General Counsel, Exelon Nuclear
Commonwealth of Pennsylvania
State of Maryland

NOTICE OF VIOLATION

Exelon Generation Company, LLC
Peach Bottom Units 2 and 3

Docket Nos. 50-277; 50-278
License Nos. DPR-44; DPR-56
EA-03-224

During an NRC inspection conducted between September 24, 2003 - November 18, 2003, two violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violations are listed below:

A. Technical Specification 5.4.1 states, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Appendix A, November 1972. Regulatory Guide 1.33 lists procedures for performing maintenance that can affect the performance of safety related equipment.

Contrary to the above, between July 1992 and September 2003, Exelon did not maintain maintenance procedure M-052-011, "Standby Diesel Generator Cylinder Liner Replacement," Revision 1, dated March 24, 1992, because it did not incorporate adequate guidance to ensure proper sealing of adaptor gaskets. Specifically, the procedure did not include directions to inspect the adapter gaskets prior to installation to ensure that a required annealing process did not cause gasket damage or irregularities. Prior notice of this issue was specified in Fairbanks Morse Service Information Letter A-15 dated June 22, 1987.

B. 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," requires, in part, that measures be established to assure that conditions adverse to quality, such as malfunctions and deficiencies, are promptly identified and corrected.

Contrary to the above, when low, swinging jacket water pressure indications were identified in March and April 2003, actions taken to correct the condition were inadequate. Although Exelon initiated a maintenance action request for the observed conditions, the action request was closed without performing several troubleshooting steps that could have identified the problem, such as checks of air or combustion gas in-leakage and engine compression. These troubleshooting steps were specified in the Fairbanks Morse vendor manual.

These violations are associated with a WHITE significance determination process finding for Unit 2 and a GREEN significance determination process finding for Unit 3.

The NRC has concluded that information regarding the reason for the violations, the corrective actions taken and planned to correct the violations and prevent recurrence, and the date when full compliance was achieved are already adequately addressed on the docket in NRC Inspection Report 50-277/03-13; 50-278/03-13 dated December 18, 2003. 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-03-224," 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 I, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this 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), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. Therefore, to the extent possible, the response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

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

Dated this 3rd day of February 2004.

BWROG tells NRC about plans to address power uprate problems

At a standing-room-only meeting Feb. 3 with representatives of the BWR Owners Group (BWROG) and General Electric, NRC got some, but not all, of the answers it was looking for on the impact of extended power uprates (EPU) on plant operations.

At the end of the 2.5-hour meeting, NRC's Richard Barrett, director of the engineering division of the Office of Nuclear Reactor Regulation (NRR), said that NRC had heard "more than I expected" about what actions were planned by industry. He also said NRC had "heard a lot" about increased monitoring of components that might be affected by the increased steam and feedwater flow resulting from extended power uprates—those that increase thermal power by 8% to 20%.
But Barrett said the agency still had questions about what components might be susceptible to problems as a result of an EPU. In addition, he said, NRC still is unsure about which plants might be vulnerable. And Barrett said that NRC still wanted to hear more about what the potential remedies might be.

Concerns about the impacts of EPUs have received increasing levels of NRC attention as a result of cracks in nonsafety related steam dryer parts at four Exelon units—Quad Cities-1 and -2, and Dresden-2 and -3—that occurred

over the past two years. Other components, including feedwater sampling probes, also have failed.
One concern NRC has about these failures is the potential for loose parts from damaged steam dryers or probes to damage other components. Exelon restarted both Dresden-2 and Quad Cities-1 with loose parts still missing after justifying to NRC that the loose parts wouldn't affect the safe shutdown of the reactors.

But at the Feb. 2 meeting, NRC's Brian Sheron, NRR's associate director for project licensing and technical analysis, was anything but sanguine about loose parts left in a reactor, especially those left for an extended period. Some loose parts at Dreseden-2 damaged other non-safety-related equipment. Sheron said that a question he would put to the BWROG is whether damage to non-safety-related components should be considered acceptable. Sheron said he would have trouble defending that logic. And he indicated that if an unfound loose part remains in the reactor for years, then the part probably should become part of the plant's design basis.

NRC's Ledyard Marsh, director of NRR's division of licensing project management, added he was concerned about whether "we are causing more loose parts with power uprates." He added that it is a possible cause-andeffect relationship "that is bothering me."

Although problems have not surfaced at other EPU plants, a representative from Progress Energy's Brunswick said that some steam dryer repairs will be necessary to allow that plant to achieve the full benefit of its 15% uprate, which NRC approved in 2002.

And Barrett said that while some EPU plants may be okay under normal operating conditions, it is unclear if problems wouldn't surface at those plants under transient conditions.

At the meeting, Exelon, the BWROG, and GE told NRC several near-term initiatives are under way to address concerns about the impact of EPUs on plant equipment. There

is "no value to an uprate if the unit is in a forced outage," Ken Putnam from the Nuclear Management Co.'s Duane Arnold said at the meeting.

Exelon said that after testing main steam line components to accelerated aging from vibrations at Quad Cities-1, it found that all components, except for electromagnetic relief valve (ERV) actuators, were acceptable for cycle operation at the EPU level. Exelon said additional testing would be done to determine any further actions required to ensure acceptable operation of the ERVs.
Exelon also said it is working with GE to develop recommendations for analytical and operational improvements to preclude component failures from the EPU. The recommendations are due in May.

The BWROG presentation indicated that GE also is working on a series of separate steam and feedwater flow path component evaluations that could, among other things, affect future EPU submittals. That work is scheduled to be completed in June.

The BWROG told NRC that it sent a survey Jan. 12 to BWRs with EPU operating experience, asking about component and subcomponent failures, increased component wear and maintenance, plant unavailability changes due to the EPU, and any EPU operational impacts (such as water chemistry changes). Those survey responses are due Feb. 12.

The BWROG said it is also reviewing the uprate database of the Institute of Nuclear Power Operations (INPO), a task that is expected to be completed Feb. 24. BWOG said it hopes to send NRC a letter in May that details the results of its survey and review of the INPO information.
Barrett acknowledged the difficulty of getting a handle on some of these issues. But he said the "very complexity" of these issues "makes it more urgent for us to understand" them. NRC, he added, will continue to evaluate what role it should be playing in this area. To date the agency has been monitoring what has been going on at the Exelon plants and has issued several information notices.

Only two plants currently have EPU uprate applications pending before NRC. Entergy's Vermont Yankee has applied for a license for a 20% EPU; Energy's Waterford, a PWR, has asked for an 8% uprate. During the meeting, Marsh said the EPU issues "raise questions about what we ask licensees" for future uprates, but he said at the end of the meeting that so far the agency has not hear anything that would "derail the Vermont Yankee schedule."

Representatives from the plant said that Exelon's experiences with power uprates would affect how quickly Vermont Yankee ascends to its uprated power level. The Vermont Yankee application is controversial, with critics marshaling experts to try to review detials of the project. And recently, NRC briefed staffers of Vermont's U.S. senators, James Jeffords (I) and Patrick Leahy (D) about the uprate project. After that conference call Jan. 15, the staffers said that formal correspondence from the senators would follow.—*Michael Knapik, Washington*

NRC calls for safety culture probe by PSEG at Hope Creek, Salem

Public Service Enterprise Group (PSEG) last week said it was undertaking "an in-depth self-assessment of the work environment" at Salem and Hope Creek after receiving an NRC letter that warned of the dangers of "creat[ing] an unacceptable, chilled environment for raising issues and making appropriate operational decisions."

The Jan. 28 NRC letter, from Region I Administrator Hubert Miller to PSEG Chairman/President/CEO E. James Ferland, said NRC had "initiated a special review" at the two plants in late 2003. PSEG subsidiary PSEG Nuclear operates both Salem and Hope Creek. Miller said the agency decided to conduct the review "in light of information received in various allegations and inspections over the past few years."

Though NRC has not identified any serious safety violations, thus far, Miller said that "collectively, information gathered has led to concerns about the stations' work environment, particularly as it relates to the handling of emergent equipment issues and associated operational decision making." As Miller noted, NRC has found shortcomings in PSEG's problem identification and resolution (PI&R) efforts, an observation recorded in the agency's mid-cycle plant assessment last August (INRC, 8 Sept. '03, 1) and earlier.

NRC spokesman Scott Burnell called the PI&R concerns an example, or "subset," of issues that led to a more generalized concern about the safety conscious work environment (SCWE) at the two plants.

According to Miller's letter, the NRC review "has accumulated information about a number of events which, to varying degrees, call into question the openness of management to concerns and alternative views, strength of communications, and effectiveness of the stations' corrective action and feedback processes." Miller continued, "Several events involved disagreements or differing perspectives of operators and senior managers on plant operating decisions, particularly as they might impact on continuing plant operation and outage schedules."

PSEG spokesman Chic Cannon said that some issues raised by employees have been addressed, but not exactly in the way the workers suggested. In such cases, Cannon said, the employee might feel "disgruntled."

Interviews at Hope Creek and Salem to date, Miller said in his letter, "have raised questions about whether management has fully assessed and addressed the negative impact" such disagreements had on plant staff. "If left unresolved," Miller said, "negative outfall [sic] from events relayed to us can create an unacceptable, chilled environment for raising issues and making appropriate operational decisions."

The letter also highlighted the potential role of schedule pressures. While acknowledging that "virtually all plants, including those with strong safety performance, operate with aggressive schedules" and that "schedule pressure does not, by itself, lead to safety concerns," Miller said, "[W]e consider it important for you to take action to thoroughly understand what 'messages' the staffs at Salem and Hope Creek have taken from various events over the past few years and address any situations that significantly detract from maintenance of a strong safety conscious work environment."

Cannon said the need to adhere to a schedule could lead to "competing interests" and that an operator would "have to resolve that competition in a conservative way." While it may be true, he said, that "when you're in the fit and fury of [trying to complete a job on schedule], it may be you're not as conservative as you think you're being," PSEG's procedures and practices ensured that nuclear safety was never compromised.

The issue of schedule pressures is a sensitive one—perhaps even more so in the aftermath of the discovery in March 2002 of severe corrosion of the reactor vessel head at Davis-Besse. Both NRC and Davis-Besse operator FirstEnergy Nuclear Operating Co. (Fenoc) found that an emphasis on production at the expense of safety was an overarching cause of the problems at the Ohio plant.

PSEG claims progress

In a Jan. 30 statement, Ferland said PSEG had been "fully aware of" the SCWE issues "for a period of time." He said a management reorganization last year that included the hiring of Roy Anderson as chief nuclear officer had "addressed many of the issues." He continued, "We are pleased that the NRC has recognized the management realignment that has occurred and the positive response it has generated at the stations."

Miller's letter cited the management changes as "an attempt to better support the separate activities of Hope Creek and Salem and to improve implementation" of PSEG's corrective action program. "While some interviewees have indicated that these steps may be leading to some change under new management, it is vital to assess the climate at the station, address the current impact of previous unresolved conflict, and take steps to assure the staffs of Salem and Hope Creek are willing to participate," he added.

NRC Region I spokesman Neil Sheehan characterized his agency's letter as "put[ting] PSEG on notice" with regard to the SCWE issues. Ferland acknowledged in his statement the utility was aware it must "work diligently" to continue improving the workplace environment.

Miller asked PSEG for a "plan of action" within 30 days to address the issues he raised. He said NRC would like to meet with PSEG about two weeks after the submittal. The Miller letter is available on Adams, NRC's electronic document retrieval system, with accession number ML040280476.

Differing reactions from groups

Advocacy groups differed in their responses to the NRC and PSEG actions. In a Jan. 29 press release, Norm Cohen of the Unplug Salem Campaign criticized the notion of "allowing" PSEG to conduct an internal investigation as "yet another example of letting the fox guard the henhouse." Cohen said Miller's letter contained information that showed "PSEG has already conducted numerous in-house investigations, and safety culture problems have only worsened." The NRC, he argued, should have shut all three reactors, "pending a complete investigation."

In contrast, David Lochbaum of the Union of Concerned Scientists praised the agency in a Feb. 2 letter to the commission for its "prudent" step. Region I's response in this instance was much better than what the agency did in the cases of Millstone and Davis-Besse, Lochbaum said. There, he said in a Feb. 5 interview, NRC "waited for the wheels to fall off" before taking action.

However, Lochbaum drew a distinction between the situations that developed at the two plants, saying that in the Millstone case, workers were "intimidated" and therefore did not raise safety concerns, while at Davis-Besse the workers did bring up such concerns. But after seeing that Fenoc was not responsive, they stopped doing so, he said.

Lochbaum said there were "a lot of signs of trouble" at the two PSEG plants. In addition to the assessments Miller mentioned, Lochbaum cited a marked increase over the past two years in the number of allegations made to the NRC. He said he was familiar with the specific allegations but could not elaborate; they are still under investigation.
In his letter, Lochbaum acknowledged that some observers might criticize the NRC for acting either "too

hastily or heavy-handedly" or "too belatedly or too meekly." But he lauded Region I for having "blazed a new trail" in the absence of procedural guidance on handling safety-culture issues.

The problem, he said, is that the Reactor Oversight Process (ROP) "lamely hand-waves at the thorny subject by labeling safety culture a 'cross-cutting issue.'" It deals more with restoring a plant's safety culture once the damage caused by deficiencies has become apparent than with "preventing safety culture debacles," he said.

Lochbaum assigned himself part of the blame for this ROP shortcoming, citing his involvement in the process of designing the system. Quoting from NRC's Inspection Manual Chapter 0308, he noted that the agency decided that "no separate and distinct assessment of licensee safety culture is needed because it is subsumed by either the [performance indicators] or baseline inspection activities."

However, he said, the flaw in that logic was demonstrated by Davis-Besse. In the inspections prior to the discovery of the cavity in the reactor head, all of the performance indicators were green, indicating NRC's lowest level of safety concern.

The solution to the problem, Lochbaum said, was a "mini-ROP," as he put it in the interview. In the letter, he laid out his concept of a smaller version of the fall 1997 public workshop on the draft ROP. It would involve identification of the key elements of a strong safety culture, as well as the regulatory structure, including inspections and performance indicators, that would be needed to maintain quality and provide early warning of problems. Lochbaum said he anticipated that the effort "would culminate in mostly re-packaging of existing inspection processes with minor additions and revisions rather than wholesale retooling." He said he had been wrong to oppose safety-culture rulemaking —as he did in 1997 and 2002—"not because safety conscious work environment rulemaking was necessary, but because I failed to advocate what should be done to address the underlying problem."—*Daniel Horner, Washington*

Inside NRC
Volume 26 / Number 4 / February 23, 2004

NRC critical of NEI's draft sump evaluation guidance

NRC expressed concern that the nuclear industry would not have its sump evaluation guidance ready in time to meet the agency's schedule for issuing a generic letter to PWRs on the subject.

In a Feb. 9 letter, NRC told NEI that a Los Alamos National Laboratory (LANL) preliminary review of NEI's draft guidance "identifies a number of problems with the methodology [for evaluating sumps] which, together with the omission of significant elements—and the draft quality and status of the methodology—pose concern for meeting the schedule."

LANL's review was conducted under a technical support contract the lab has with NRC's Office of Nuclear Reactor Regulation.

The potential for blockage from debris generated from a loss-of-coolant accident (LOCA) is known within the NRC as generic safety issue (GSI)-191. NRC plans to issue a generic letter in August to PWR licensees, asking them to evaluate the susceptibility of their sumps to blockage.

"In its current condition, the NEI guidance is incomplete in many respects and will result in industry blockage evaluations at that are inconsistent and potentially nonconservative," LANL said in an eight-page evaluation of NEI's draft guidance that accompanied the Feb. 9 letter. "[T]he present guidance will not serve the needs of a typical plant engineer who will be faced with the complexities of an emergencycore-cooling-system (ECCS) assessment."

Among its criticisms, LANL said the NEI guidance "does not establish an expectation of quality and consistency among individual plant evaluations." LANL also said that

NEI guidance "contains a mix of assumptions, both over conservative and under conservative. However, the overconservative assumptions cannot be relied upon to counter the under-conservative assumptions in the overall assessment." The letter included 11 specific examples of where NEI's guidance is "not conservative." Among that list, LANL said that NEI guidance used "a spherical shaped zone of influence" for debris generation that was based on guidance developed for BWRs when the blockage issue was addressed for them even though the BWR guidance "is not directly applicable to a PWR pipe break"; that the NEI guidance "relies heavily upon the extrapolation of existing debris test data to recommend values for untested material" in terms of such data points as debris size and transportability but not justifying the rational for the extrapolation; that NEI's guidance for estimating size distributions of LOCA-generated debris "is limited and partially incorrect"; and that the guidance "does not adequately address the pathways by which debris enters the sump pool." LANL also said the NEI guidance "does not address potential mitigation strategies for identified vulnerabilities."

Generic letter on schedule

Although the letter was not specific on what areas of the GSI-191 schedule could be affected by the gaps in NEI's guidance, NRC staffer Michael Marshall said the next step—the issuance of a draft generic letter for comment—would happen on schedule, either by the month's end or first week in March.

The letter does note that the final generic letter, which will instruct PWR licensees to perform an evaluation of their containment sumps to determine the susceptibility to blockage during a LOCA, "will rely on licensees using a methodology approved by NRC staff" and that NRC staff wants to work with NEI to developing a methodology that "could be endorsed by NRC" by the time the agency issues the final generic letter in August.

The GSI-191 resolution schedule is "firm at this point," Marshall said. He said that right now NRC will be working with NEI to make sure the industry understands the deficiencies outlined in the Feb. 9 letter and will encourage NEI

to "fill in the holes" so the guidance can be endorsed by the agency. If those holes still exist in the guidance, Marshall said the NRC would fill them in. Marshall noted that there will be a two-day meeting with industry, tentatively scheduled for March 17-18 at NRC headquarters in Rockville, Md. John Butler, a senior project manager at NEI, told Inside NRC that the industry is "still evaluating the impact of the [LANL] comments" and is awaiting a full list of comments from NRC, which he said is expected to be delivered to NEI by the end of the month, as well as any issues discussed in next month's meeting before revising its guidance document.
—Gregory Twachtman, Washington

'One size fits all' said not appropriate for SCWE best practices guidance

During a recent workshop held by NRC to gather input on the development of a "best practices" guidance document to establish and maintain a safety conscious work environment (SCWE), industry and other interested stakeholders suggested that a single guidance document covering all NRC licensees may be difficult to develop because the safety culture varies widely from licensee to licensee.

Additionally, the attendees at the Feb. 19 workshop at NRC headquarters in Rockville, Md., also debated the form a best practices guidance document should take in order to get the maximum benefit.

Charles Thebaud, an attorney with Morgan Lewis, said the notion of calling the document a "best practices" document a misnomer. To say that the practices in a guidance document are the best practices "is not a good idea because what works at one place may not work at another." Ellen Ginsberg, deputy attorney general for the Nuclear Energy Institute (NEI), also said that there needs to be a level of flexibility for different licensees to implement different practices.

One workshop participant added that as the culture of the workplace changes, through events such as employee or management turnover or a change in company ownership, what was a best practice may no longer be applicable, making it difficult to capture the best practices in one guidance document. It was also noted that it is possible that some best practices could be in conflict with one another, but no specific examples were given.

Debate on guidance presentation

Chuck Dugger, NEI's vice president of operations, opened
t he workshop by stating that the best way to address SCWE

issues is "not going through this exercise" but rather through industry initiatives. He noted that NEI already has issued guidance on establishing and maintaining a SCWE (NEI 97-05) and that revision 2 of that document was issued in December 2003. He added that a second document, "Principles for a Strong Nuclear Safety Culture," has been issued for comment and is expected to be issued later this year. "The industry has not been silent" on the SCWE issue, Dugger said.

But Billie Garde, an attorney who represents workers in discrimination cases, argued that NEI's work is limited to the nuclear power industry and unless NEI is willing to broaden its reach to cover materials licensees, NEI guidance is not enough. "The NEI can't reach everyone," Garde said.
Garde noted that many materials licensees don't have guidance directing them in the development of a SCWE and reiterated her call for an SCWE rule. Ginsberg suggested that NRC might consider separate guidance for different classes of
1 icensees.

Frank Congel, director of NRC's Office of Enforcement, reminded the workshop attendees that the NRC commission in a March 26, 2003 staff requirements memorandum rejected the idea of pursuing rulemaking for oversight of an SCWE and in that SRM instructed NRC staff to develop SCWE guidance based on industry best practices (INRC, 7 April '03, 6).

However, the concept of a best practices document raised concerns among the attendees. In particular, there were questions about what a best practices documents should contain, what form it should take and whether there would be any enforcement authority related to the best practices document.

Michael Brothers, a safety culture consultant with Brothers Engineering & Consulting LLC and former vice president of operations at Millstone, said there needs to be "a clear understanding" of what it means when one licensee says that it is not going to follow any of the recommendations in a best practices document.

David Lochbaum, a nuclear safety engineer with the

Union of Concerned Scientists, suggested that the best practices guidance take the form of a regulatory guide or some other kind of enforceable document. He added that simply issuing a best practice guidance document that does not define minimum standards will not serve the industry, the agency, or the public.

Brothers also agreed with the idea of a reg guide, particularly because it would allow a degree of flexibility. He noted that if the reg guide outlined the best practices, licensees would then either implement those practices or submit justification to the NRC on why they would be deviating from the guidance.

Dugger, however, cautioned that a licensee could end up interpreting a reg guide as a checklist of what has to be done to maintain an SCWE. Morgan Lewis' Thebaud echoed Dugger's comments. He questioned whether a licensee would be viewed as having substandard performance if a best practice in a reg guide was not implemented

Congel was not supportive of the idea of a SCWE best practices reg guide. He said the goal of the best practice guidance is more about the sharing of information.
Gregory Twachtman, Washington

IG says NRC overlooking key issue in responses to Davis-Besse findings

NRC's Inspector General (IG) registered the latest in a series of differences with the commission over the agency's handling of the Davis-Besse incident, saying NRC's planned responses to the discovery of serious reactor vessel head corrosion at the FirstEnergy Nuclear Operating Co. (Fenoc) plant fall short because they "do not address the underlying, more generic communication failures" that the IG's Office identified in a report last fall.

Inspector General Hubert Bell's Feb. 2 memorandum to NRC Chairman Nils Diaz was a rebuttal to a Jan. 12 memo from Samuel Collins, the agency's deputy executive director for reactor programs. Collins was responding to the October 2003 IG report (OIG-03-02S), in which Bell and his staff found serious shortcomings in NRC's handling of boric acid issues at Davis-Besse before and during the plant's 2000 refueling outage (INRC, 3 Nov. '03, 10). The report included a discussion of what the IG staff said was NRC's failure to recognize the significance of a photograph and accompanying condition report indicating a large accumulation of rust-colored boric acid on the reactor head.

In a separate, earlier report (OIG-02-03S), the IG had raised questions about the decision-making of the staff in allowing Davis-Besse to operate past Dec. 31, 2001. That report drew a harsh response from the commission (INRC, 13 Jan. '03, 1). Collins, in his Jan. 12 memo replying to the October 2003 IG report, said that "it is important to recognize that the problem at Davis-Besse represented an institutional failure on the part of the NRC." But he said the agency was in the process of addressing the problems, citing the list of corrective actions it is taking on the basis of recommendations made by a Lessons Learned Task Force on Davis-Besse (INRC, 7 April '03, 9). Bell replied those actions focus too narrowly "on specific findings pertaining to NRC's handling of boric acid and corrosion"

at Davis-Besse." George Mulley, a senior level assistant in the IG's Office and drafter of the memo, told Inside NRC Feb. 18 that much of the work NRC is doing is "right on the money" but the broader communication problem "jumps out at you" from the IG report.

While NRC headquarters had been sending out generic communications on boric acid issues for 15 years, "there was never a directive to follow through," he said. Thus, the efforts "began at headquarters and ended at headquarters," he said. That contributed to the resident inspector's "seeing a photograph and not knowing what it means," Mulley said. If the inspector had recognized what the photograph was depicting, there is "no doubt in our mind" that he would have pursued the matter, Mulley added.

But the communication problem also worked in the other direction, failing to send important information up the chain of command, he said. He cited a finding of the report that a log book kept by the reactor projects branch 4 chief contained 38 entries from April 1999 to April 2000—when the refueling outage began—on issues pertaining to boric acid.

He said he was not aware of other cases in which a similar communications failure had caused such problems, but he noted the report constituted a probe of the specific case at Davis-Besse rather than a more general audit of NRC practices. Mulley said the IG was trying to send a message in its Feb. 2 memo that NRC was "missing a key point of the report." He said he wasn't sure what type of response would be forthcoming. The commission is required to respond to IG reports, but the Feb. 2 document was a response to the response to a report, he noted.

The commission is scheduled to receive an update from the staff Feb. 26 on the implementation of the post-Davis-Besse actions. A staffer said discussion of the Feb. 2 IG memo is not on the agenda.

The staffer said the program was "pretty much running according to schedule," although some plans "had to be rescheduled." There also have been some additions, he said.

He said the presentation would be led by James Dyer, director of the Office of Nuclear Reactor Regulation, with reports from program managers for the four areas into which the task force's recommendations have been divided: stress corrosion cracking; operating experience program effectiveness; inspection, assessment, and project management guidance; and assessment of barrier integrity requirements.—*Daniel Horner, Washington*

Fact Sheet

United States Nuclear Regulatory Commission
Office of Public Affairs
Washington DC 20555
Telephone: 301/415-8200 E-mail: opa@nrc.gov

The Accident At Three Mile Island

The accident at the Three Mile Island Unit 2 (TMI-2) nuclear power plant near Middletown, Pennsylvania, on March 28, 1979, was the most serious in U.S. commercial nuclear power plant operating history(1), even though it led to no deaths or injuries to plant workers or members of the nearby community. But it brought about sweeping changes involving emergency response planning, reactor operator training, human factors engineering, radiation protection, and many other areas of nuclear power plant operations. It also caused the U.S. Nuclear Regulatory Commission to tighten and heighten its regulatory oversight. Resultant changes in the nuclear power industry and at the NRC had the effect of enhancing safety.

The sequence of certain events - - equipment malfunctions, design related problems and worker errors - - led to a partial meltdown of the TMI-2 reactor core but only very small off-site releases of radioactivity.

Summary of Events

The accident began about 4:00 a.m. on March 28, 1979, when the plant experienced a failure in the secondary, non-nuclear section of the plant. The main feedwater pumps stopped running, caused by either a mechanical or electrical failure, which prevented the steam generators from removing heat. First the turbine, then the reactor automatically shut down. Immediately, the pressure in the primary system (the nuclear portion of the plant) began to increase. In order to prevent that pressure from becoming excessive, the pilot-operated relief valve (a valve located at the top of the pressurizer) opened. The valve should have closed when the pressure decreased by a certain amount, but it did not. Signals available to the operator failed to show that the valve was still open. As a result, cooling water poured out of the stuck-open valve and caused the core of the reactor to overheat.

As coolant flowed from the core through the pressurizer, the instruments available to reactor operators provided confusing information. There was no instrument that showed the level of coolant in the core. Instead, the operators judged the level of water in the core by the level in the pressurizer, and since it was high, they assumed that the core was properly covered with coolant. In addition, there was no clear signal that the pilot-operated relief valve was open. As a result, as alarms rang and warning lights flashed, the operators did not realize that the plant was experiencing a loss-of-coolant accident. They took a series of actions that made conditions worse by simply reducing the flow of coolant through the core.

Because adequate cooling was not available, the nuclear fuel overheated to the point at which the zirconium cladding (the long metal tubes which hold the nuclear fuel pellets) ruptured and the fuel pellets began to melt. It was later found that about one-half of the core melted during the early stages of the accident. Although the TMI-2 plant suffered a severe core meltdown, the most dangerous kind of nuclear power accident, it did not produce the worst-case consequences that reactor experts had long feared. In a worst-case accident, the melting of nuclear fuel would lead to a breach of the walls of the containment building and release massive quantities of radiation to the environment. But this did not occur as a result of the Three Mile Island accident.

The accident caught federal and state authorities off-guard. They were concerned about the small releases of radioactive gases that were measured off-site by the late morning of March 28 and even more concerned about the potential threat that the reactor posed to the surrounding population. They did not know that the core had melted, but they immediately took steps to try to gain control of the reactor and ensure adequate cooling to the core. The NRC's regional office in King of Prussia, Pennsylvania, was notified at 7:45 a.m. on March 28. By 8:00, NRC Headquarters in Washington, D.C. was alerted and the NRC Operations Center in Bethesda, Maryland, was activated. The regional office promptly dispatched the first team of inspectors to the site and other agencies, such as the Department of Energy and the Environmental Protection Agency, also mobilized their response teams. Helicopters hired by TMI's owner, General Public Utilities Nuclear, and the Department of Energy were sampling radioactivity in the atmosphere above the plant by midday. A team from the Brookhaven National Laboratory was also sent to assist in radiation monitoring. At 9:15 a.m., the White House was notified and at 11:00 a.m., all non-essential personnel were ordered off the plant's premises.

By the evening of March 28, the core appeared to be adequately cooled and the reactor appeared to be stable. But new concerns arose by the morning of Friday, March 30. A significant release of radiation from the plant's auxiliary building, performed to relieve pressure on the primary system and avoid curtailing the flow of coolant to the core, caused a great deal of confusion and consternation. In an atmosphere of growing uncertainty about the condition of the plant, the governor of Pennsylvania, Richard L. Thornburgh, consulted with the NRC about evacuating the population near the plant. Eventually, he and NRC Chairman Joseph Hendrie agreed that it would be prudent for those members of society most vulnerable to radiation to evacuate the area. Thornburgh announced that he was advising pregnant women and pre-school-age children within a 5-mile radius of the plant to leave the area.

Within a short time, the presence of a large hydrogen bubble in the dome of the pressure vessel, the container that holds the reactor core, stirred new worries. The concern was that the hydrogen bubble might burn or even explode and rupture the pressure vessel. In that event, the core would fall into the containment building and perhaps cause a breach of containment. The hydrogen bubble was a source of intense scrutiny and great anxiety, both among government authorities and the population, throughout the day on Saturday, March 31. The crisis ended when experts determined on Sunday, April 1, that the bubble could not burn or explode because of the absence of oxygen in the pressure vessel. Further, by that time, the utility had succeeded in greatly reducing the size of the bubble.

Health Effects

Detailed studies of the radiological consequences of the accident have been conducted by the NRC, the Environmental Protection Agency, the Department of Health, Education and Welfare (now Health and Human Services), the Department of Energy, and the State of Pennsylvania. Several independent studies have also been conducted. Estimates are that the average dose to about 2 million people in the area was only about 1 millirem. To put this into context, exposure from a full set of chest x-rays is about 6 millirem. Compared to the natural radioactive background dose of about 100-125 millirem per year for the area, the collective dose to the community from the accident was very small. The maximum dose to a person at the site boundary would have been less than 100 millirem.

In the months following the accident, although questions were raised about possible adverse effects from radiation on human, animal, and plant life in the TMI area, none could be directly correlated to the accident. Thousands of environmental samples of air, water, milk, vegetation, soil, and foodstuffs were collected by various groups monitoring the area. Very low levels of radionuclides could be attributed to releases from the accident. However, comprehensive investigations and assessments by several well-respected organizations have concluded that in spite of serious damage to the reactor, most of the radiation was contained and that the actual release had negligible effects on the physical health of individuals or the environment.

Impact of the Accident

The accident was caused by a combination of personnel error, design deficiencies, and component failures. There is no doubt that the accident at Three Mile Island permanently changed both the nuclear industry and the NRC. Public fear and distrust increased, NRC's regulations and oversight became broader and more robust, and management of the plants was scrutinized more carefully. The problems identified from careful analysis of the events during those days have led to permanent and sweeping changes in how NRC regulates its licensees - - which, in turn, has reduced the risk to public health and safety.

Here are some of the major changes which have occurred since the accident:

- Upgrading and strengthening of plant design and equipment requirements. This includes fire protection, piping systems, auxiliary feedwater systems, containment building isolation, reliability of individual components (pressure relief valves and electrical circuit breakers), and the ability of plants to shut down automatically;
- Identifying human performance as a critical part of plant safety, revamping operator training and staffing requirements, followed by improved instrumentation and controls for operating the plant, and establishment of fitness-for-duty programs for plant workers to guard against alcohol or drug abuse;
- Improved instruction to avoid the confusing signals that plagued operations during the accident;
- Enhancement of emergency preparedness to include immediate NRC notification requirements for plant events and an NRC operations center which is now staffed 24 hours a day. Drills and

response plans are now tested by licensees several times a year, and state and local agencies participate in drills with the Federal Emergency Management Agency and NRC;

- Establishment of a program to integrate NRC observations, findings, and conclusions about licensee performance and management effectiveness into a periodic, public report;
- Regular analysis of plant performance by senior NRC managers who identify those plants needing additional regulatory attention;
- Expansion of NRC's resident inspector program - first authorized in 1977 - whereby at least two inspectors live nearby and work exclusively at each plant in the U.S to provide daily surveillance of licensee adherence to NRC regulations;
- Expansion of performance-oriented as well as safety-oriented inspections, and the use of risk assessment to identify vulnerabilities of any plant to severe accidents;
- Strengthening and reorganization of enforcement as a separate office within the NRC;
- The establishment of the Institute of Nuclear Power Operations (INPO), the industry's own "policing" group, and formation of what is now the Nuclear Energy Institute to provide a unified industry approach to generic nuclear regulatory issues, and interaction with NRC and other government agencies;
- The installing of additional equipment by licensees to mitigate accident conditions, and monitor radiation levels and plant status;
- Employment of major initiatives by licensees in early identification of important safety-related problems, and in collecting and assessing relevant data so lessons of experience can be shared and quickly acted upon;
- Expansion of NRC's international activities to share enhanced knowledge of nuclear safety with other countries in a number of important technical areas.

Current Status

Today, the TMI-2 reactor is permanently shut down and defueled, with the reactor coolant system drained, the radioactive water decontaminated and evaporated, radioactive waste shipped off-site to an appropropriate disposal site, reactor fuel and core debris shipped off-site to a Department of Energy facility, and the remainder of the site being monitored. The owner says it will keep the facility in long-term, monitored storage until the operating license for the TMI-1 plant expires at which time both plants will be decommissioned. Below is a chronology of highlights of the TMI-2 cleanup from 1980 through 1993.

Date	Event
July 1980	Approximately 43,000 curies of krypton were vented from the reactor building.
July 1980	The first manned entry into the reactor building took place.
Nov. 1980	An Advisory Panel for the Decontamination of TMI-2, composed of citizens, scientists, and State and local officials, held its first meeting in Harrisburg, PA.
July 1984	The reactor vessel head (top) was removed.
Oct. 1985	Defueling began.
July 1986	The off-site shipment of reactor core debris began.
Aug. 1988	GPU submitted a request for a proposal to amend the TMI-2 license to a "possession-only" license and to allow the facility to enter long-term monitoring storage.
Jan. 1990	Defueling was completed.
July 1990	GPU submitted its funding plan for placing $229 million in escrow for radiological decommissioning of the plant.
Jan. 1991	The evaporation of accident-generated water began.
April 1991	NRC published a notice of opportunity for a hearing on GPU's request for a license amendment.
Feb. 1992	NRC issued a safety evaluation report and granted the license amendment.
Aug. 1993	The processing of accident-generated water was completed involving 2.23 million gallons.
Sept. 1993	NRC issued a possession-only license.
Sept. 1993	The Advisory Panel for Decontamination of TMI-2 held its last meeting.
Dec. 1993	Post-Defueling Monitoring Storage began.

Additional Information

Further information on the TMI-2 accident can be obtained from sources listed below. The documents can be ordered for a fee from the NRC's Public Document Room at 301-415-4737 or 1-800-397-4209;

e-mail pdr@nrc.gov. The PDR is located at 11555 Rockville Pike, Rockville, Maryland; however the mailing address is: U.S. Nuclear Regulatory Commission, Public Document Room, Washington, D.C. 20555. A glossary is also provided below.

Additional Sources for Information on Three Mile Island

NRC Annual Report - 1979, NUREG-0690

"Population Dose and Health Impact of the Accident at the Three Mile Island Nuclear Station," NUREG-0558

"Environmental Assessment of Radiological Effluents from Data Gathering and Maintenance Operation on Three Mile Island Unit 2," NUREG-0681

"Report of The President's Commission on The Accident at Three Mile Island," October, 1979

"Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement," NUREG-0600

"Three Mile Island; A Report to the Commissioners and to the Public," by Mitchell Rogovin and George T. Frampton, NUREG/CR-1250, Vols. I-II, 1980

"Lessons learned From the Three Mile Island - Unit 2 Advisory Panel," NUREG/CR-6252

"The Status of Recommendations of the President's Commission on the Accident at Three Mile Island," (A ten-year review), NUREG-1355

"NRC Views and Analysis of the Recommendations of the President's Commission on the Accident at Three Mile Island," NUREG-0632

"Environmental Impact Statement related to decontamination and disposal of radioactive wastes resulting from March 28, 1979 accident Three Mile Island Nuclear Station, Unit 2," NUREG-0683

"Answers to Questions About Updated Estimates of Occupational Radiation Doses at Three Mile Island, Unit 2," NUREG-1060

"Answers to Frequently Asked Questions About Cleanup Activities at Three Mile Island, Unit 2," NUREG-0732

"Status of Safety Issues at Licensed Power Plants" (TMI Action Plan Reqmts.), NUREG-1435

Walker, J. Samuel, Three Mile Island: A Nuclear Crisis in Historical Perspective, Berkeley: University of California Press, 2004.

Other Organizations to Contact

GPU Nuclear Corp, One Upper Pond Road, Parsippany, NJ, 07054, telephone 201-316-7249;

Three Mile Island Public Health Fund, 1622 Locust Street, Philadelphia, PA, 19103, telephone 215-875-3026;

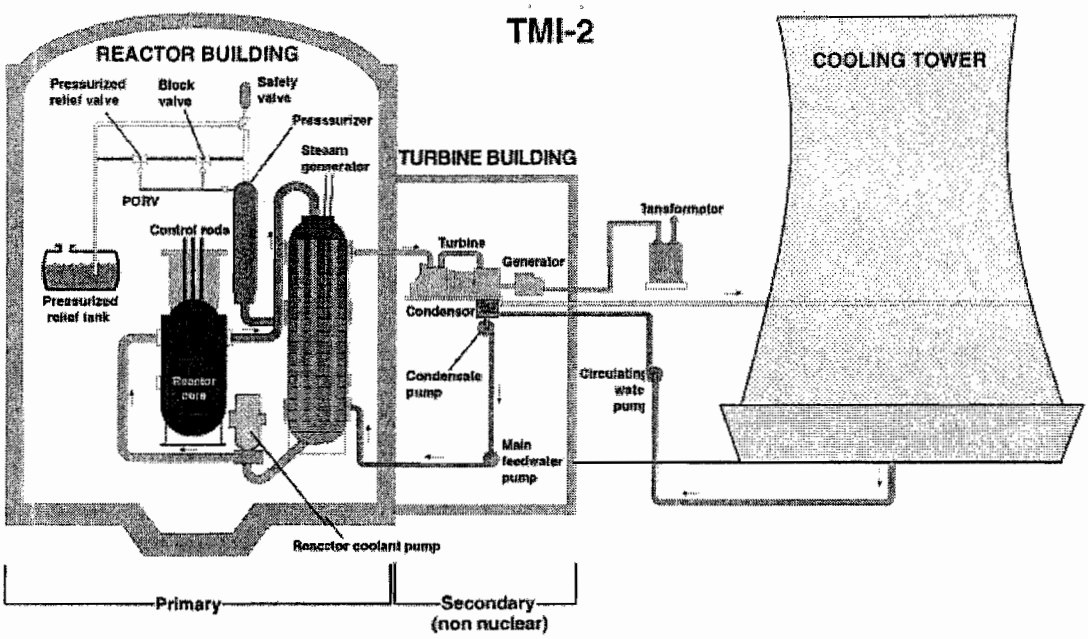
Pennsylvania Bureau of Radiation Protection, Department of Environmental Protection, Rachel Carson State Office Building, P.O. Box 8469, Harrisburg, PA, 17105-8469, telephone 717-787-2480.

March 2004

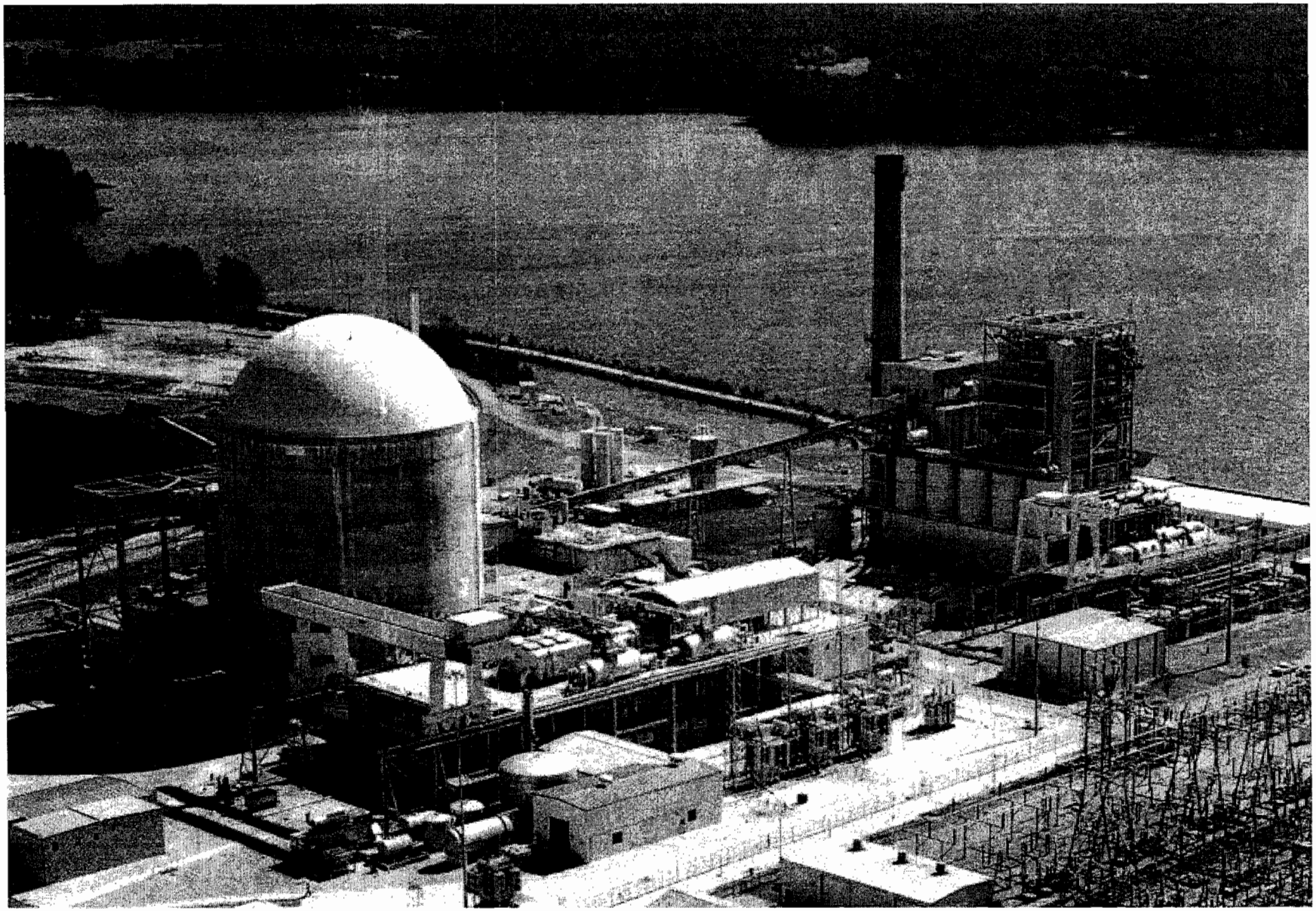
Glossary

Auxiliary feedwater - (see emergency feedwater)
Background radiation - The radiation in the natural environment, including cosmic rays and radiation from the naturally radioactive elements, both outside and inside the bodies of humans and animals. The usually quoted average individual exposure from background radiation is 360 millirem per year.
Cladding - The thin-walled metal tube that forms the outer jacket of a nuclear fuel rod. It prevents the corrosion of the fuel by the coolant and the release of fission products in the coolants.
Aluminum, stainless steel and zirconium alloys are common cladding materials.
Emergency feedwater system - Backup feedwater supply used during nuclear plant startup and shutdown; also known as auxiliary feedwater.
Fuel rod - A long, slender tube that holds fuel (fissionable material) for nuclear reactor use. Fuel rods are assembled into bundles called fuel elements or fuel assemblies, which are loaded individually into the reactor core.
Containment - The gas-tight shell or other enclosure around a reactor to confine fission products that otherwise might be released to the atmosphere in the event of an accident.
Coolant - A substance circulated through a nuclear reactor to remove or transfer heat. The most commonly used coolant in the U.S. is water. Other coolants include air, carbon dioxide, and helium.
Core - The central portion of a nuclear reactor containing the fuel elements, and control rods.
Decay heat - The heat produced by the decay of radioactive fission products after the reactor has been shut down.
Decontamination - The reduction or removal of contaminating radioactive material from a structure, area, object, or person. Decontamination may be accomplished by (1) treating the surface to remove or decrease the contamination; (2) letting the material stand so that the radioactivity is decreased by natural decay; and (3) covering the contamination to shield the radiation emitted.
Feedwater - Water supplied to the steam generator that removes heat from the fuel rods by boiling and becoming steam. The steam then becomes the driving force for the turbine generator.
Nuclear Reactor - A device in which nuclear fission may be sustained and controlled in a self-supporting nuclear reaction. There are several varieties, but all incorporate certain features, such as fissionable material or fuel, a moderating material (to control the reaction), a reflector to conserve escaping neutrons, provisions for removal of heat, measuring and controlling instruments, and protective devices
Pressure Vessel - A strong-walled container housing the core of most types of power reactors.
Pressurizer A tank or vessel that controls the pressure in a certain type of nuclear reactor.
Primary System - The cooling system used to remove energy from the reactor core and transfer that energy either directly or indirectly to the steam turbine.
Radiation - Particles (alpha, beta, neutrons) or photons (gamma) emitted from the nucleus of an unstable atom as a result of radioactive decay.
Reactor Coolant System - (see primary system)
Secondary System - The steam generator tubes, steam turbine, condenser and associated pipes, pumps, and heaters used to convert the heat energy of the reactor coolant system into mechanical energy for electrical generation.
Steam Generator - The heat exchanger used in some reactor designs to transfer heat from the primary (reactor coolant) system to the secondary (steam) system. This design permits heat exchange with little or no contamination of the secondary system equipment.

Turbine - A rotary engine made with a series of curved vanes on a rotating shaft. Usually turned by water or steam. Turbines are considered to be the most economical means to turn large electrical generators.



1. The catastrophic Chernobyl accident in the former Soviet Union, in 1986, was by far the most severe nuclear reactor accident to occur in any country; it is widely believed an accident of that type could not occur in U.S.-designed plants. For more detail on the accident at Chernobyl, see Fact Sheet at http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/fschernobyl.html.



ROBINSON NUCLEAR PLANT

ACRS MEETING

March 4, 2004

Progress Energy

RNP Unique Differences

- Robinson Site Consists of a Fossil Plant (Unit1) and a Nuclear Plant (Unit 2)
- RNP Containment-
  - Grouted Tendons
  - Liner is Insulated (Limit Heat Transfer during postulated DBA)
- 480 Volt Emergency Power (versus 4160 volt)
- Safe Shut Down Diesel (in addition to 2 Emergency Diesel Generators)

Major Equipment Replacement/Upgrade

Within Past 20 Years-

- Steam Generators Replaced (1984)
- Service Water Piping Replaced
  - Inside containment (1988)
  - From booster pumps to containment (1990)
  - North Header (1999)
- Turbine Rotor Replaced (LP 1987, HP 2002)
- Power Uprate (Appendix K, ~ 2% in 2002)
  - No current plans for additional uprate

Major Equipment Replacement/Upgrade

Ongoing or Planned

- Security Upgrades (4Q04)
- RV Head Replacement (RO 23, Fall 2005)
  - RNP Request for relief from NRC Order related to RV Head Inspection withdrawn.
- Dry Fuel Storage (Load 1st Module 3Q05)
- Generator & Exciter Refurbishment (RO 24)

Progress Energy

Operating Experience

	2000	2001	2002	2003	2004
Capacity	103.96	92.18	93.70	103.54	(2/23) 105.88 (proj.) 95.14
Refuel	NA	4/7 to 5/12	10/12 to 11/14	NA	28 day plan- April 20
Exposure (REM)	8.4	124.8	110.6	4.8	(Goal) < 9 Plus RO22

Currently, continuous run of 465 days*. Breaker to Breaker operation between spring 2001 and fall 2002 refueling. Other offline, minimal:

- 6/21/00 to 6/22/00 Manual Trip due to Turbine EH oil leak
- 11/24/02* Turbine taken offline to repair steam leak

All NRC Performance Indicators are Green

Progress Energy

Boric Acid Program

(reference - 06/03 AMR Inspection Report and 09/30 ACRS subcommittee meeting)

- Corporate *"Boric Acid Corrosion Control"* Program has been implemented at Progress Energy PWRs. Procedure guidance includes

  "*All plant personnel should recognize borated system leakage, understand its significance, and initiate corrective action when boric acid residue is detected.*"

  "*If carbon and low-alloy steel components are exposed to boric acid, the components shall be carefully cleaned and visually inspected.*"

Progress Energy

Boric Acid Program

- RNP System Walkdown Procedure revised to include

  "*Boric acid corrosion of carbon steel components can adversely impact component integrity. When boric acid leakage is detected, initiate a work request and/or condition report as appropriate to be evaluated in accordance Boric Acid Corrosion Control Program*"

Commitments/Tracking

- 47 Programs credited for License Renewal. 10 are existing programs and require no changes. 37 Commitments for 27 Enhancements and 10 New Programs have been entered into RNP Commitment Tracking Program
- All Commitments will be either implemented or "transitioned" from LR to Plant Organization for future implementation by July 2004
- The RNP Supervisor of Licensing/Regulatory Programs has overall responsibility for management of commitment tracking

Commitments/Tracking

- Once Implemented
  - Commitments are identified in implementing documents
  - Change controlled by 10 CFR 50.59 process
- Configuration control process will incorporate guidance to ensure that requirements of 10 CFR 54.37(b) are met; Support by
  - License Renewal Training (October 2004)
  - License Renewal DBD (July 2004)
  - UFSAR Supplement (October 2004)

(3)

H.B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2

License Renewal
Safety Evaluation Report

Staff Presentation to the ACRS
SIKHINDRA (S.K.) MITRA
Project Manager
March 4, 2004

Background

- **JUNE 14, 2002: CP&L SUBMITTED LICENSE RENEWAL APPLICATION**
- **SEPTEMBER 30, 2003: ACRS SUBCOMMITTEE BRIEFING ON SER /OI**
- **JANUARY 20, 2004: SER ISSUED**
  - **REQUREMENTS OF PART 54 HAVE BEEN MET**
- **CURRENT LICENSE EXPIRES JULY 31, 2010**
- **REQUEST LICENSE RENEWAL THROUGH JULY 31, 2030**

NRC Audits and Inspections

- THREE INSPECTIONS AND TWO AUDITS
- SCOPING AND SCREENING METHODOLOGY AUDIT
  - SEPTEMBER 17 - 20, 2002
- SCOPING AND SCREENING INSPECTION
  - MARCH 31 – APRIL 4, 2003
- AGING MANAGEMENT PROGRAM AUDIT
  - MAY 28 – 29, 2003
- AGING MANAGEMENT INSPECTION
  - JUNE 9 – 14 and JUNE 23 -27, 2003
- FINAL INSPECTION
  - SEPTEMBER 9 – 10, 2003

Aging Management Program Audit

OBJECTIVE: REVIEW AMPs CONSISTENCY WITH GALL

- DATE OF AUDIT - MAY 28-29, 2003
- AUDIT REPORT DATED AUGUST 12, 2003
- AUDITED ALL THE ATTRIBUTES OF THE AMPs CLAIMED TO BE CONSISTENT WITH GALL
- CONCLUDED AMPs WERE CONSISTENT WITH GALL EXCEPTING:
  - NON-EQ INSULATED CABLES AND CONNECTIONS PROGRAM LACKED DETAIL TO CONCLUDE CONSISTENCY WITH GALL
    - AMP WAS REVISED AND SUBMITTED TO TECHNICAL STAFF FOR REVIEW
    - STAFF FOUND IT ACCEPTABLE

Aging Management Inspection

- **OBJECTIVE: VERIFICATION OF THE ACCURACY OF THE APPLICATION IMPLEMENTATION WITH REGARD TO AMPs**
- **CONDUCTED JUNE 9–27, 2003**
- **OBSERVATION:**
  - **INCOMPLETE INTEGRATION OF FUTURE TASKS INTO ESTABLISHED SITE ACTION REQUEST TRACKING SYSTEM**
- **INSPECTION REPORT (50-261/03-09) ISSUED ON July 31, 2003**

Aging Management Inspection (Continued)

- **THIRD (OPTIONAL) INSPECTION**
- **CONDUCTED SEPTEMBER 9-10, 2003**
- **APPLICANT HAD LOADED FUTURE TASKS INTO ESTABLISHED SITE ACTION REQUEST TRACKING SYSTEM**
- **TRANSITION PLAN FOR COMPLETION OF LICENSE RENEWAL PROJECT WAS ESTABLISHED**
- **INSPECTION REPORT (50-261/03-11) ISSUED ON SEPTEMBER 29, 2003**

Open Items

- TWO OPEN ITEMS AND THIRTY CONFIRMATORY ITEMS ALL OPEN AND COMFIRMATORY ITEMS ARE RESOLVED
- Open Item 2.3.1.6-1
  - STAFF IDENTIFIED THAT DEGRADATION OF THE FEEDRINGS, J-NOZZLES, OR J-NOZZLE WELDS COULD PRODUCE LOOSE PARTS INSIDE THE STEAM GENERATOR SHELL
    - MAY DAMAGE SAFETY-RELATED COMPONENTS, ESPECIALLY DURING TRANSIENTS
  - COMPONENTS BROUGHT INTO SCOPE AND OPEN ITEM IS RESOLVED

Open Items (continued)

- Open Item 2.3.3.8-1
  - FOLLOWING A LAKE ROBINSON DAM FAILURE AND DEPLETION OF CONDENSATE STORAGE TANK INVENTORY, FAILURE OF DEEPWELL PUMPS WOULD CAUSE FAILURE OF THE SAFETY RELATED AUXILIARY FEEDWATER SYSTEM AND PREVENT THE RESIDUAL HEAT REMOVAL NECESSARY TO MAINTAIN A SAFE SHUTDOWN CONDITION
  - THREE DEEPWELL PUMPS, ASSOCIATED PIPING, AND VALVES WERE BROUGHT INTO SCOPE AND OPEN ITEM IS RESOLVED

RESOLUTION OF CONFIRMATORY ITEM 4.6.4 .1 - AGING OF BORAFLEX

- LICENSE AMENDMENT WAS SUBMITTED TO ELIMINATE CREDIT OF THE BORAFLEX PANELS FROM RNP TECHNICAL SPECIFICATIONS
- STAFF REVIEWED THE AMENDMENT APPLICATION AND APPROVED THE APPLICANT REQUEST
- DOCUMENTED IN LICENSE AMMENDMENT 198, ISSUED ON DECEMBER 22, 2003, AND SER SECTION 4.6.4

TLAA - REACTOR VESSEL NEUTRON EMBRITTLEMENT

- Analysis of PTS projected to end of PEO
- Staff performed independent calculations

ITEMS	LIMIT (°F) (MAXIMUM)	RNP (°F)
CIRCUMFERENTIAL WELDS	300	275
PLATES/FORGINGS/AXIAL WELDS	270	235

PTS = Pressurized Thermal Shock PEO = Period of Extended Operation

REACTOR VESSEL UPPER SHELF ENERGY (USE)

- ANALYSIS OF USE PROJECTED AT THE END OF PEO
- STAFF PERFORMED INDEPENDENT CALCULATION

REACTOR VESSEL UPPER SHELF ENERGY (USE)	LIMIT (MINIMUM) FT-LBS	RNP FT-LBS
WELDS/FORGINGS	50	56
PLATE MATERIALS	42 (EMA)	45
NOZZLE FORGING	50	53
NOZZLE WELDS	50	52

EMA = Equivalent Margin Analysis

AP1000 Status

March 4, 2004
ACRS Full Committee Meeting

John Segala, Senior Project Manager
Office of Nuclear Reactor Regulation

Overview

- Purpose
  - Provide status of the staff's review
  - Discuss major schedule milestones
  - Provide overview of remaining Draft SER open items
- Conclusion
  - On schedule to issue Final SER by September 13, 2004

AP1000 Review Chronology

- March 2002 - Completed pre-application review
- March 28, 2002 - Westinghouse (W) submitted DC application
- June 25, 2002 - NRC accepted the application for docketing
- June 16, 2003 - NRC issued DSER with 174 Open Items
- FSER Review Progressing

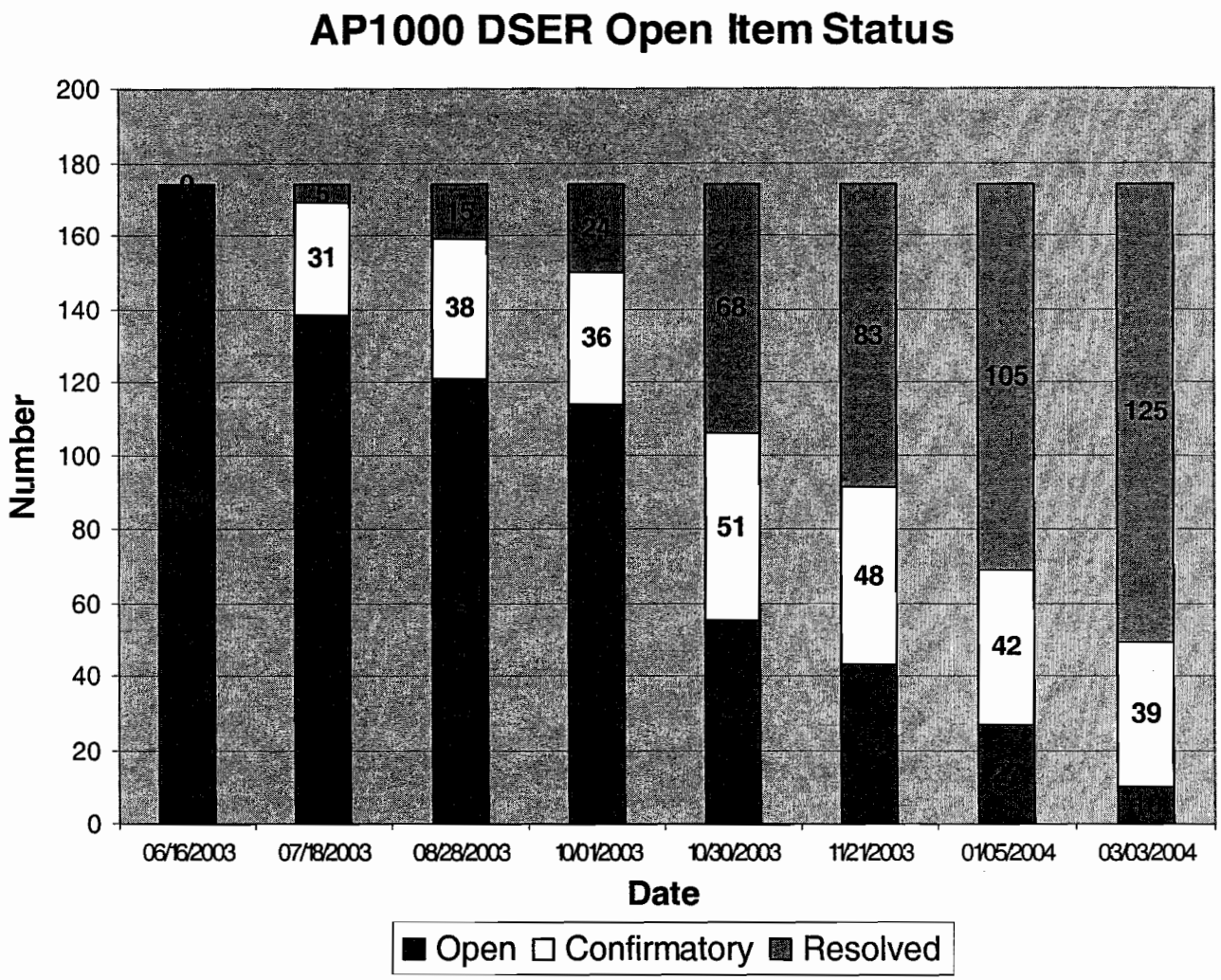
Schedule Milestones

- March 31, 2004 - W provides acceptable responses to all open items
- May 25, 2004 – No Open Item FSER to ACRS
- May 31, 2004 - W submits final AP1000 design control document
- June 25, 2004 - ACRS Future Plant Design Subcommittee Meeting
- July 7-9, 2004 - Full ACRS Committee Meeting
- September 13, 2004 - Final SER and FDA issued

AP1000 DSER Open Item Resolution

Working to Resolve Open Items

- 10 open
- 164 technical resolution completed



Remaining Open Items

- Security (2 Open Items)
  - New COL Action Item
    - Deferred Security Plan to the COL applicant
  - Staff is currently reviewing the ITAAC

Remaining Open Items (Continued)

- Aerosol Removal Coefficients (3 Open Items)
  - Need to determine if AP1000 Removal Coefficients are applicable
  - Sandia National Laboratory Contract
    - Monte Carlo uncertainty analysis
    - 200 runs of MELCORE for DEDVI line break
    - Removal Coefficient over time
  - Staff currently reviewing draft report
  - Staff will run independent dose calculations with W and Sandia's Removal Coefficients

Remaining Open Items (Continued)

- Leak Before Break (1 Open Item)
  - W using LBB for Main Steam piping
  - AP1000 does not have a diverse means of detecting main steam line leakage

Remaining Open Items (Continued)

- NRC Open Items (4 Open Items)
  - Review of final AP1000 Design Control Document Revision
  - Review of Tier 2* information
  - Review of COL Action Items
  - Documentation of AP600 FSER information

Conclusion

- On schedule to issue Final SER by September 13, 2004
- Questions/Comments?

V C Summer Nuclear Station

ACRS Presentation

Al Paglia

Jamie LaBord

Bob Whorton

Purpose

- Background/History
- Application
- Issues of Interest:
  - Hot Leg Crack
  - Head Inspections
  - Sump Blockage Bulletin
  - Thermal Fatigue
  - Ground Water Issues
  - SWPH Settlement
- Commitment Tracking/Living Program

Background/History

- 1000 MWe 3 Loop Westinghouse PWR
- Initial License granted August 1982
- SCE&G is 2/3 owner and licensee
- Santee Cooper is 1/3 owner
- Steam Generator Replacement – 1994
- Up-rate 2775 MWt to 2900 MWt – 1996
- NRC Indicators and Findings all Green

Application

- Application developed in accordance with Regulatory Guide 1.188, utilizing guidance from NEI 95-10
- Format in accordance with NUREG 1800, Standard Review Plan, with comparisons to NUREG 1801, GALL, as appropriate

Hot Leg Crack

- "A" Hot Leg weld replaced with a spool piece utilizing Alloy 690 weld materials
- Root Cause of crack attributed to high residual stresses resulting from original installation weld repairs
- NDE results of all other loop nozzle welds showed no recordable indications

Head Inspections

- Upper Head
  - RF-14 – 100% bare metal inspection
- Lower Head
  - RF-14 inspection – 360 degree 100% bare metal
- No Active Leaks or Degradation

Sump Blockage

- Sump inspections RF-14
  - Original installation gaps
  - Level instrument replacement gaps
- Future Plans
  - Evaluate current sump design\surfac area
  - Modification (if necessary) within NRC established schedule for GSI-191

Thermal Fatigue

- WESTEMS utilized for cycle counting including high usage components
- The year 2002 CUF for the Pressurizer surge line is 0.38
  - Changes made to operating procedures to accumulation of usage on surge line nozzle
- Year 2002 CUF for the normal charging line is 0.46 and the alternate charging is 0.47
- VCS committed to re-compute the CUF for NUREG/CR-6260 locations using guidelines of NUREG/CR-6583 and NUREG/CR-5704

VCSNS Groundwater Evaluations

- Groundwater initially identified (2002) as "mildly acidic, but non-aggressive"
- Recent analyses (October 2003) from new wells indicate that groundwater is "non-aggre" – thus minimal effects on buried component

	pH	Cl	SO
Old Wells	**4.8 – 5.3**	**< 10 ppm**	**< 10 pp**
New Wells	**6.0 – 7.1**	**< 25 ppm**	**< 185 pp**
NUREG-1801 (GALL)	**< 5.5**	**> 500 ppm**	**> 1500 ppm**

Service Water Pump House & Intake Structure Settlement

- Excessive settlement observed during initial construction
- Subsequent re-analysis revised projected settlement & rebound estimates
- Settlement monitoring conducted semi-annually
- Results show stability for last 20+ years

Commitment Tracking/Living Program

- All commitments and associated action items have been entered into the station tracking program
- Commitment implementation guidance is being incorporated into a License Renewal Design Basis Document and Station procedures
- Configuration Control processes will incorporate guidance to ensure continuing compliance with requirements of Part 54

QUESTIONS

ACRS License Renewal Full Committee

V.C. Summer Nuclear Station License Renewal Application

Final Safety Evaluation Report
March 4, 2003

Rajender Auluck
Senior Project Manager

5

Overview

- Application submitted by letter dated August 6, 2002
- Westinghouse pressurized water reactor
- Plant located in Fairfield County, South Carolina
- Current license expires August 6, 2022 - Requests renewal through August 6, 2042
- Draft SER issued October 9, 2003
- ACRS License Renewal Subcommittee meeting held on December 3, 2003
- Final SER issued January 29, 2004

Staff Conclusions

The Applicant has met the requirements for license renewal, as required by 10 CFR 54.29:

- Actions have been identified and have been or will be taken such that there is reasonable assurance that activities will continue to be conducted in the renewal term in accordance with the current licensing basis
- The applicable requirements of 10 CFR Part 51 have been satisfied
- Matters raised under 10 CFR 2.758 have been addressed

Scoping and Aging Management

- Scoping and screening methodology is adequately described and justified in the LRA and satisfies the requirements of 10 CFR 54.21(a)(2)
- Scoping and screening review results found that the SSCs within the scope of license renewal have been identified, as required by 10 CFR 54.4(a) and those subject to an AMR have been identified, as required by 10 CFR 54.21(a)(1)
- Aging management review found that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation, as required by 10 CFR 54.21(a)(3)

Scoping and Aging Management

Audit and Inspections

- Scoping and Screening Methodology Audit
  - January 28-31, 2003
- Scoping and Screening Inspection
  - May 12-16, 2003
- Aging Management Program Audit
  - July 16-17, 2003
- Aging Management Review Inspection
  - August 4-8, and August 18-22, 2003
- Third Inspection
  - November 19-20, 2003

Aging Management

Aging Management Programs

- 45 AMPs credited for license renewal
  - 34 AMPs are consistent with GALL
  - 11 AMPs are non-GALL programs
  - 26 AMPs – Existing Programs
  - 16 AMPs – New Programs
  - 3 AMPs – TLAA Programs
- 3 AMPs added as a result of staff review

Aging Management

Aging Management of In-Scope Inaccessible Concrete

	Aggressive Limit	V.C. Summer
pH	<5.5	4.8 - 5.3
Chlorides	>500 ppm	<10 ppm
Sulphates	>1500 ppm	<10 ppm

- Applicant has initiated additional site groundwater studies.
- Additional provisions to be added to existing plant programs and procedures.

TLAAs

- The applicant has identified the appropriate TLAAs and had demonstrated or is committed to demonstrate that the TLAAs:
  - Will remain valid for the period of extended operation
  - Have been projected to the end of the period of extended operation, or
  - The aging effects will be adequately managed for the period of extended operation

TLAAs

Reactor Vessel Upper Shelf Energy (USE)

Reactor Vessel Beltline Material	Screening Criteria USE FT-LBS	Staff Calculated USE FT-LBS
Limiting Beltline Plate Material	≥ 50	53
Limiting Weld	≥ 50	59

Pressurized Thermal Shock

Limiting Beltline Material	RT_{PTS} Criterion (°F)	Staff Calculated RT_{PTS} (°F)
Base Metal Intermediate Shell Plate A9154-1	≤ 270	158
Axial Weld 4P4784	≤ 270	110

TLAAs

Metal Fatigue

- Reactor coolant system components at V.C. Summer designed to Class 1 requirements of the ASME Code.
- Three components may exceed the design basis fatigue usage factor during the period of extended operations.

Commitment

- Transients will be tracked by Thermal Fatigue Management Program (TFMP)
- Perform evaluation of NUREG/CR-6260 components for environmental fatigue prior to the period of extended operation
- Components with CUFs projected to exceed 1.0 will be either re-analyzed or replaced prior to exceeding cycles of transients tracked by TFMP

Commitment Tracking System

- SER Appendix A lists the applicant's license renewal commitments
- Fulfillment of commitments will be confirmed by the staff with Inspection Procedure 71003
- Commitments are tracked using the station tracking program
- Implementation guidance being incorporated into a License Renewal Design Basis Document and Station Procedures

License Conditions and Environmental Review

- Two standard license conditions:
  - Following issuance of the renewed license, the applicant will include the UFSAR Supplement in the next UFSAR update, as required by 10 CFR 50.71(e)
  - Future inspection activities identified in the UFSAR Supplement will be completed prior to the period of extended operation
- No new plant-specific license conditions
- Staff's environmental evaluation documented in NUREG-1437, Supplement 12, published on February 27, 2004

Staff Conclusions

The Applicant has met the requirements for license renewal, as required by 10 CFR 54.29:

- Actions have been identified and have been or will be taken such that there is reasonable assurance that activities will continue to be conducted in the renewal term in accordance with the current licensing basis
- The applicable requirements of 10 CFR Part 51 have been satisfied
- Matters raised under 10 CFR 2.758 have been addressed

ACRS Review of Research Quality

George E. Apostolakis
apostola@mit.edu

510th ACRS Meeting
March 4, 2004

7

The Analytic-Deliberative Decision-Making Process

- *Analysis* uses rigorous, replicable methods, evaluated under the agreed protocols of an expert community - such as those of disciplines in the natural, social, or decision sciences, as well as mathematics, logic, and law - to arrive at answers to factual questions.

- *Deliberation* is any formal or informal process for communication and collective consideration of issues.

  National Research Council, *Understanding Risk*, 1996.

- Application to our problem:
  - The proposed framework is intended to help us focus on high-level issues in our deliberations.
  - The members' judgment always prevails.

Formal Analysis

- What is important? (*Objectives*)
- To what extent are the objectives satisfied? (*Performance Measures,* i)
- What is the relative importance of the performance measures? (*Weights,* w_i)
- How does the project rate with respect to each of the performance measures? (*Utility Functions on Constructed Scales,* u_{ij})
- What is the overall rating of the project? (*Performance Index,* PI)

$$PI = \sum_{i=1}^{N_{PM}} w_i \left(\sum p_{ij} u_{ij}\right)$$

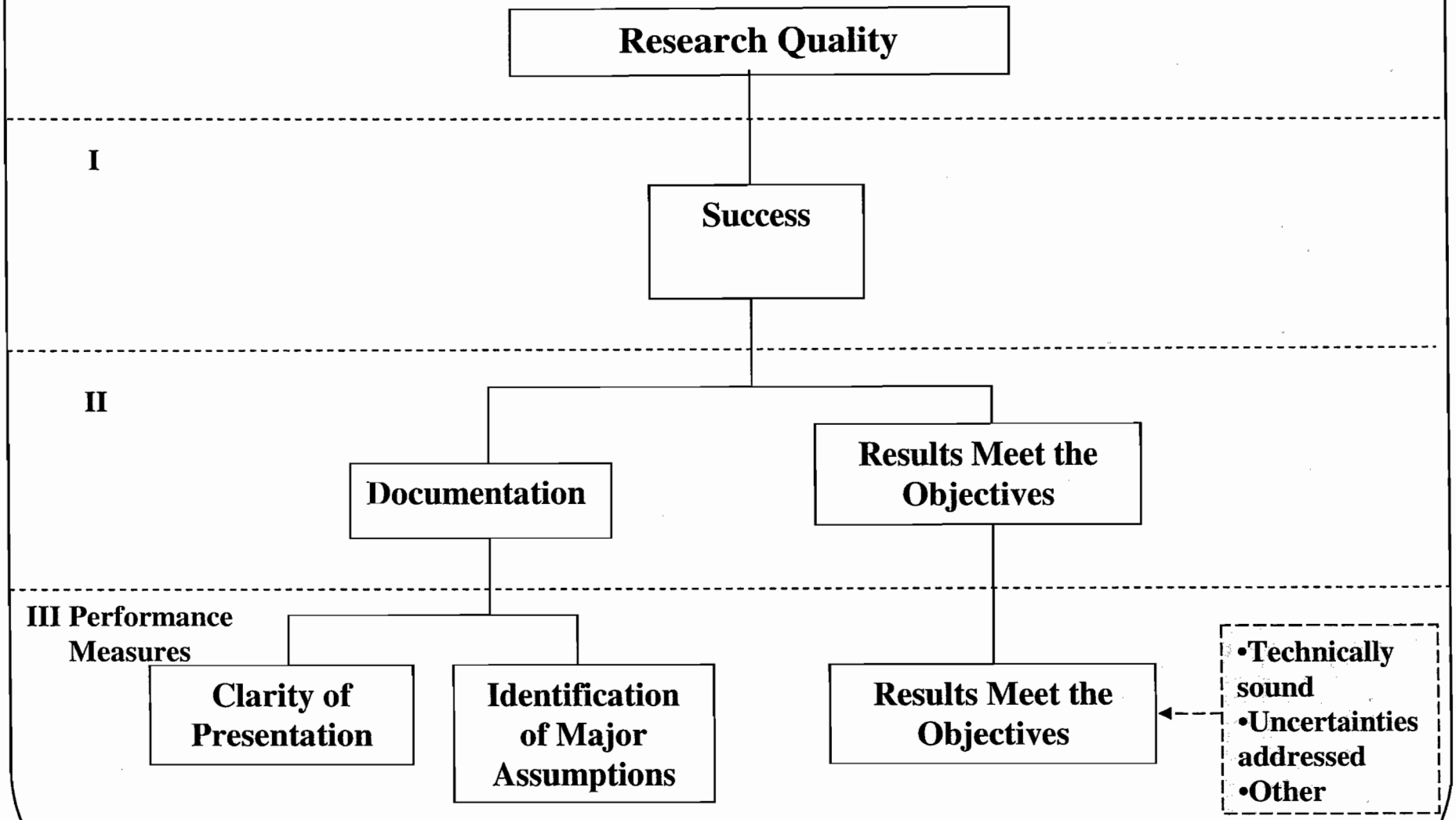
Utility of level j of the constructed scale for PM i: u_{ij}

Probability that level j will be achieved (for starting projects only): p_{ij}

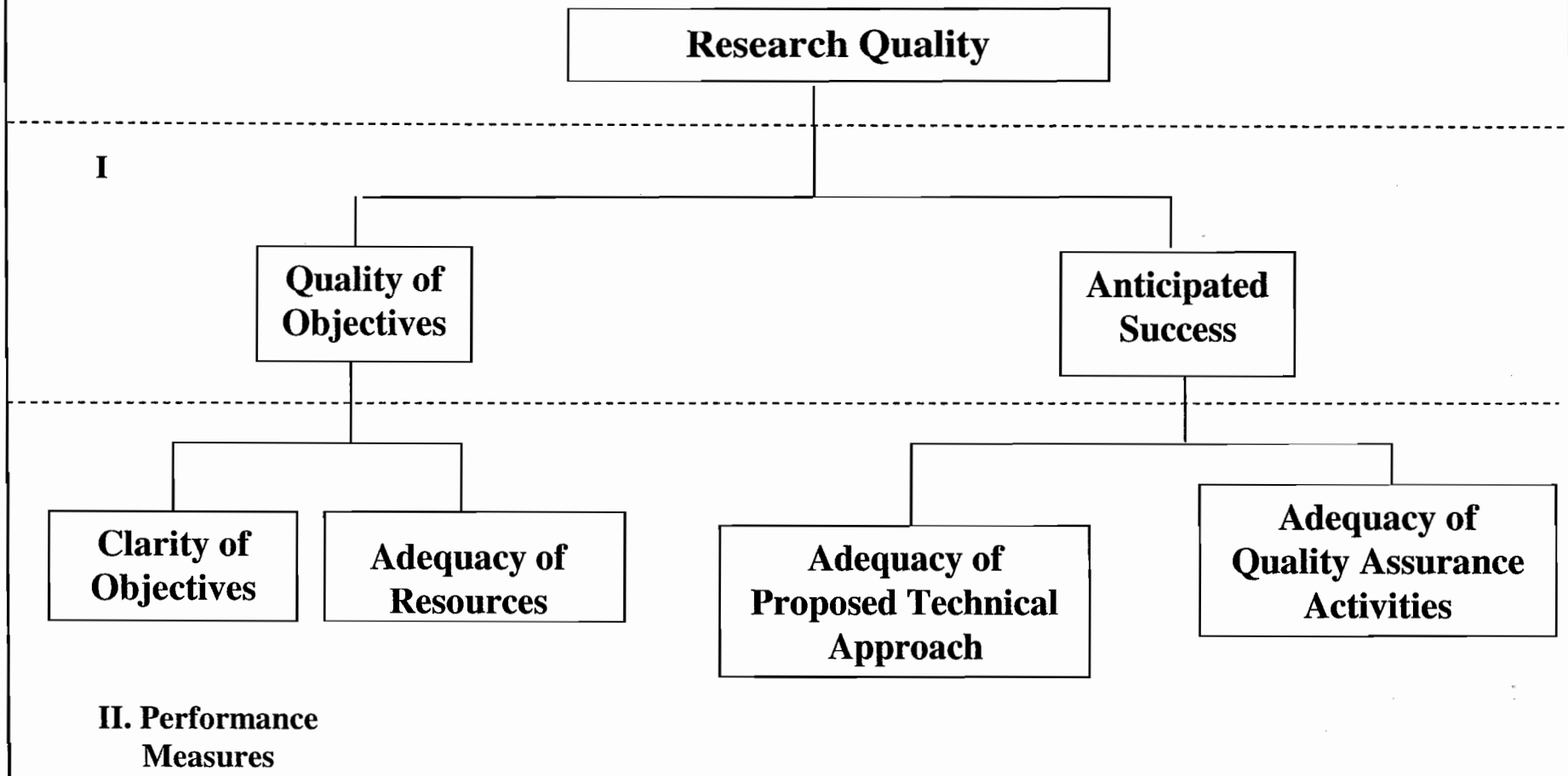
Remarks

- **The decision-making framework must be problem-specific.**
  - **Should large and small projects be evaluated using the same metrics and the same relative weights?**
  - **Should finished and starting projects be evaluated using the same metrics and the same relative weights?**

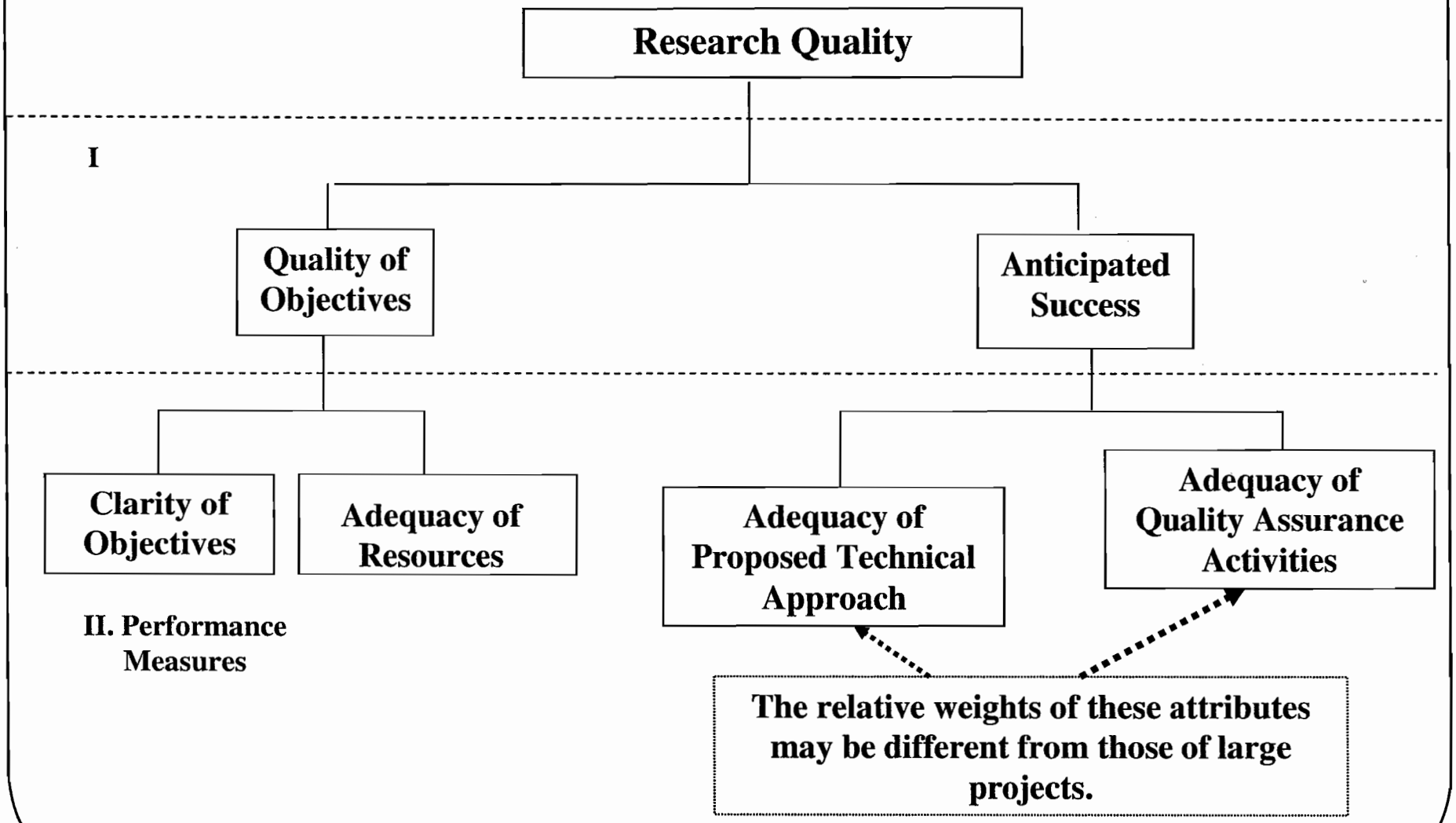
The Value Tree for Finished Projects



The Value Tree for Large Starting Projects



The Value Tree for Small Starting Projects



The Weights

- **Recall that**

$$PI = \sum_{i=1}^{N_{PM}} w_i \left(\sum p_{ij} u_{ij}\right)$$

- **The weights are scaling factors that sum to unity**

$$\sum_{1}^{N_{PM}} w_i = 1$$

- **They represent trade-offs between PMs. They can be assessed directly or using structured approaches, such as SMART and AHP. The ACRS has the final approval.**

Example of Weight Elicitation (Finished Projects): Tier II

- **How would you weigh *Documentation* vs. *Results Meet the Objectives* with respect to *Success*?**
- **Set the weight of *Documentation* equal to 10.**

➢ **Suppose the answer is 25.**

➢ **The normalized weights are:**

✓ ***Documentation:*** $\frac{10}{35} = 0.29$

✓ ***Results Meet the Objectives:*** **0.71**

Example of Weight Elicitation (Finished Projects): Tier III

- **Compare *Clarity of Presentation* and *Identification of Major Assumptions* wrt *Documentation*.**

➢**Suppose the answer is: *Clarity of Presentation* 10; *Identification of Major Assumptions* 15**

➢**Normalized weights wrt *Documentation*:**

✓***Clarity of Presentation*:** $\frac{10}{25} = 0.40$

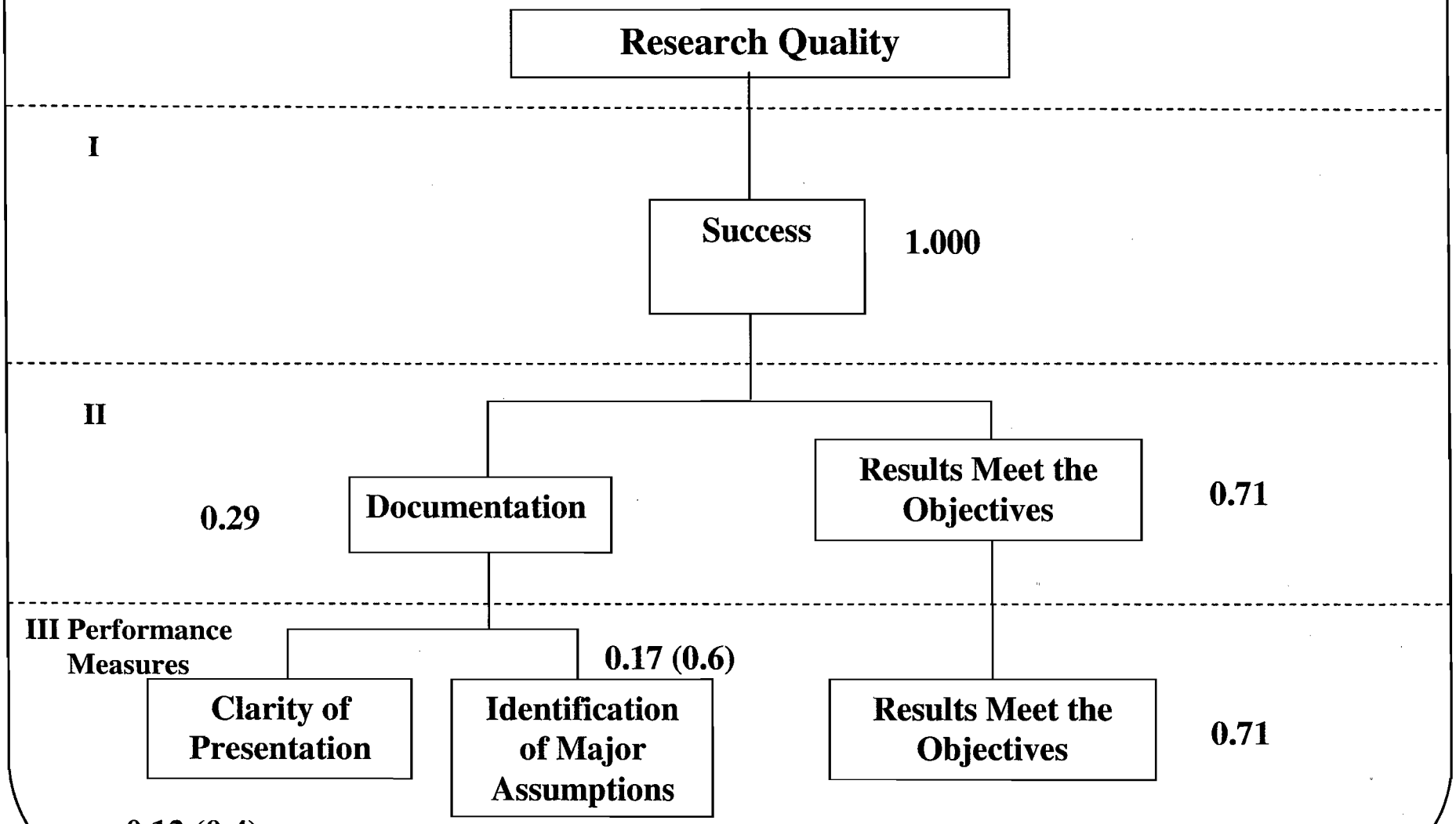
✓***Identification of Major Assumptions*: 0.60**

▪**Overall weights (wrt *Research Quality*):**

✓***Clarity of Presentation*: 0.4x0.29 = 0.12**

✓***Identification of Major Assumptions*: 0.6x0.29 = 0.17**

Overall Weights (Finished Projects)



Scale and Utility Values for *Clarity of Presentation*

- O Outstanding 10 points
- E Excellent 8 points
- S Satisfactory 5 points
- M Marginal 3 points
- U Unacceptable 0 points

First Consistency Checks:

- **Should *Outstanding* be twice more important than *Satisfactory*?**
- **Should *Outstanding* be about three times more important than *Satisfactory*?**

Consistency Checks between Performance Measures

- Suppose *Clarity of Presentation* is judged to be at the O level. Then, it contributes to the overall Performance Index the amount of 10x0.12 = 1.2.
- Suppose *Results Meet the Objectives* is judged to be at the M level. Then, it contributes to the overall Performance Index the amount of 3x0.71 = 2.13.

- Do we find these results acceptable? Would we agree that, under the given circumstances, the contribution of poor presentation should be about one-half of the contribution of meeting the objectives?

Overall Evaluation (Finished Projects)

- **Suppose that the ACRS judges the project to be at the following levels of the constructed scales:**

➢ ***Clarity of Presentation:*** **S $\Rightarrow$ u = 5**

➢ ***Identification of Major Assumptions:*** **M $\Rightarrow$ u = 3**

➢ ***Results Meet the Objectives:*** **E $\Rightarrow$ u = 8**

✓**PI = 0.12x5 + 0.17x3 + 0.71x8 = 6.8**

✓**Therefore, this project is between S and E (closer to E).**

ACRS/ACNW MEETING WITH EDO AND PROGRAM OFFICE DIRECTORS

Jack Strosnider, Deputy Director
Office of Nuclear Regulatory Research

March 5, 2004

(8)

High-Priority Topics for 2004 and 2005

- **Spent Fuel Pool Risk**
- **Materials Research Program**
- **Non-Light-Water-Reactor Issues**
- **Advanced Reactor Licensing Framework**

Spent Fuel Pool Risk

- Response to terrorist attacks assessed using new, detailed, more realistic models
- Integral analyses of operating BWR show more margin than earlier estimates
- Practical mitigation strategies identified to further reduce risk
- Interim results presented to ACRS; staff has also met with NAS to discuss ongoing work
- Final results of pilot plant study to be presented to ACRS in September 2004

Material Research Program

- Draft research plan developed, sent to Commission in February 2004 (SECY-04-0030)
  - Enhance capabilities/tools needed to evaluate emerging issues in radiation protection and assess potential impacts on NRC programs
    - Major goal: in-house maintenance of models/codes for assessment of health effects
- Preliminary plans to discuss implementation with ACNW in Fall 2004

Non-LWR Issues

- **Issues developed in pre-application reviews of gas-cooled reactors:**
  - **1) Expectations for safety; 2) defense in depth; 3) use of international standards; 4) probabilistic approach to licensing basis; 5) source term; 6) containment vs. confinement; 7) emergency preparedness.**
  - **Detailed in SECY-03-0047; June 2003 SRM approved staff recommendation to include #2, 4, 5, 7 in framework; more information requested on #1, 6.**
- **Commission Paper in preparation to address #1, 6, to be discussed with ACRS in April meeting.**
- **Further work on other items to be folded into Advanced Reactor Framework effort**

Advanced Reactor Licensing Framework

- First step in developing risk-informed, technology-neutral structure for future reactors
- Framework will guide development of regulations and design-specific implementation
- Schedule for ACRS interactions:
  - June 2004: Subcommittee meeting to solicit input on technical issues associated with preliminary results
  - November 2004: S/C and full ACRS briefings on final draft framework prior to public review and comment; letter will be requested. Information briefing on preliminary implementation work.
  - Mid-2005: Brief ACRS on resolution of public comments and final proposed framework.

Other Significant Topics

- Several other issues may be brought to ACRS & ACNW in next 18 months, e.g.,
  - Proactive materials degradation program (ACRS)
  - Risk-informed, performance-based fuel performance criterion (ACRS)
  - Follow-up on key items in ACRS report on RES programs
  - Package Performance Study (ACNW)
    - Options discussed in SECY-04-0029, February 2004
- RES is providing technical support to program offices in other key areas.
- Further information on schedule to be developed.

Summary and Conclusions

ACRS/ACNW Meeting with EDO and Program Office Directors March 5, 2004

Jim Dyer, Director

Office of Nuclear Reactor Regulation

Overview

- Risk-Informed Initiatives
  - PRA Quality
  - 50.46 Rulemaking
- Emergent Technical Issues
  - Unanticipated Effects of Power Uprates
  - PWR Sump Performance Issues
  - Electrical Grid Reliability
- New Reactors

PRA Quality

- Commission directed staff to take actions to stabilize PRA quality expectations and requirements
- Staff will meet with:
  - Public on 2/24/04 and 3/24/04
  - ACRS on 3/25/05 and 4/15/05
- Staff plan to be provided to the Commission by July 2004

50.46 Rulemaking

- Commission directed staff to prepare rule which will risk-inform:
  - Maximum LOCA break size
  - LBLOCA/LOOP coincidence
- Staff requested Commission direction and additional guidance on policy issues to facilitate resolution of identified technical issues associated with a proposed rule.

Unanticipated Effects of Power Uprates

- Flow-induced vibration problems:
  - Steam dryers
  - Steam and feedwater equipment
- BWR Owners Group leading industry response
- NRC actions include:
  - Review and monitoring of events and industry activities
  - Issuance of information notices
  - Updating review standard
  - Considering further generic actions

PWR Sump Performance

- Two-phase generic communication approach:
  - Bulletin
  - Generic Letter
- Providing guidance to industry:
  - Regulatory Guide
  - Evaluation Methodology Guidance
  - NRC-Sponsored Research

Electrical Grid Reliability

- Offsite power system assumptions used in licensing reviews may be challenged by the existing environment
- August 14, 2003 blackout has raised concerns regarding GDC 17 and station blackout rule
- NRR is reviewing concerns regarding grid reliability for short term and long term actions
- Staff will be requesting ACRS review of recommended actions

New Reactors

- Continued need for ACRS assistance on:
  - AP1000
  - ESBWR
  - ACR-700
  - Early site permit reviews
- Benefits of early engagement on issues
- Appreciate ACRS schedule flexibility

Summary

- Tackling diverse issues in rapidly changing environment
- Using PBPM process; meeting goals
- Ensuring safety is primary goal
- Looking for efficiency improvement opportunities

ACRS/ACNW Joint Meeting with EDO and Program Office Directors

Martin J. Virgilio
Director NMSS

March 05, 2004

OVERVIEW

- Policy & Planning
- Top Issues in Next Two Years
  - High-Level Waste
  - Decommissioning
  - Transportation
  - Risk-Informing NMSS

Policy & Planning

- Recent Meetings to Improve Productivity
  - Value-Added
- Forecasting & Scheduling
  - Rolling calendar
- Pre-decisional Topics

High-Level Waste

- Risk Baseline Report
- Risk-Informed Inspections
- Igneous
- Other Pre-Licensing Activities

Decommissioning

- Risk-Informed, Performance-Based Approach
- Future Briefings (planned)
  - License Termination Rule Implementation
  - Intentional Mixing of Soil

Key Transportation Issues

- Package Performance Study (RES)
- Emerging Technical Issues
  - Moderator Exclusion
  - Burnup Credit
  - High-Burnup Fuel

Risk-Informing in NMSS

- Risk-Informed Decision Making (RIDM) Guidance
  - Consideration of risk guidelines in decisions
  - Quality of risk assessments
  - Balancing risks against other traditional factors important to the Commission
- NMSS Risk Guidelines
  - Defining Population at Risk
  - Worker Risk Guidelines
- Emergent Risk-Informing Activities

Summary

- Success in Policy & Planning
- Challenging Issues in Next Two Years
- Risk-Informed, Performance-Based Approaches

(9)

G:PPHandout

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
510	12	12.1

Title **PLANNING & PROCEDURES/ FUTURE ACRS ACTIVITIES**

Authors
JOHN T. LARKINS

List of Documents Attached

PLANNING & PROCEDURES MINUTES

12

Instructions to Preparer	From Staff Person
1. Paginate Attachments 2. Punch holes 3. Place Copy in file box	JOHN T. LARKINS

March 4, 2004

G:PlanPro(ACRS):ppmins.510

INTERNAL USE ONLY

SUMMARY MINUTES OF THE
ACRS PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
MARCH 3, 2004

The ACRS Subcommittee on Planning and Procedures held a meeting on March 3, 2004, in Room T-2B3, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss mattes related to the conduct of ACRS business. The meeting was convened at 8:30 a.m. and adjourned at 10:10 a.m.

ATTENDEES
M. Bonaca
G. Wallis
S. Rosen

ACRS Staff
J. T. Larkins
H. Larson
R. P. Savio
S. Duraiswamy
S. Steele
M. Snodderly
M. Sykes
B.P. Jain
R. Caruso
M. Weston

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

Member assignments and priorities for ACRS reports and letters for the March ACRS meeting are attached (pp. 8-10). 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 March ACRS meeting be as shown in the attachment (pp. 8-10).

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through May 2004 is attached (pp. 8-10). 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

RECOMMENDATION

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

3) ACRS Meeting with the NRC Commissioners

The ACRS is scheduled to meet with the NRC Commissioners between 1:30 and 3:30 p.m. on Thursday, May 6, 2004, to discuss items of mutual interest. The following topics proposed by the Committee during the February 2004 ACRS meeting have been sent to SECY for Commissioners' feedback. (p. 11)

A.) Overview (MVB)
B.) PRA Quality (GEA)
C.) NRC Safety Research Program Report (DAP)
D.) Interim Review of the AP1000 Design (TSK)
E.) ESBWR Pre-Application Review (TSK)
F.) Risk-Informing 10 CFR 50.46 (WJS)
G.) PWR Sump Performance (JDS)

RECOMMENDATION

The Subcommittee recommends that the ACRS Executive Director keep the Committee informed of the Commissioners' feedback on the list of topics proposed by the Committee.

4) ACRS Report on the NRC Safety Research Program

An advance copy of the 2004 ACRS report on the NRC Safety Research Program was sent to the Commission on February 26, 2004. Any feedback from the Commissioners will be provided to Dr. Powers for incorporation, as appropriate. Subsequently, the final report will be published as NUREG-1635, Vol. 6.

The leadership, dedication, and hard work provided by Dr. Powers in coordinating the preparation of this report resulted in significant savings of the Committee's time. In order to maintain consistency and minimize the time spent by the Committee on future reports, Dr. Powers should continue to take the lead in coordinating the preparation of future ACRS reports.

RECOMMENDATION

The Subcommittee recommends that the Committee assign lead responsibility to Dr. Powers in coordinating the preparation of the future (biennial) reports to the Commission on the NRC Safety Research Program. Also, the Committee should decide whether it wants to write a report this year on a specific research topic (e.g., safeguards and security).

5) Revision to ACRS Action Plan

As agreed to by the Committee during its January 29-30, 2004 retreat, the ACRS Action Plan that was issued in 2001 is being revised. A significant discussion of the proposed revision to the Action Plan will be the planned ACRS pro-active initiatives. Members should provide comments on items to be included or dropped from the current Action Plan to Maggalean Weston by March 12, 2004. A proposed revision to the Action Plan will be provided to the Planning and Procedures Subcommittee for consideration during its April 14, 2004 meeting. Subsequent to incorporating the Subcommittee's comments, this Plan will be provided to the ACRS members for comment.

RECOMMENDATION

The Subcommittee recommends that the ACRS members provide comments on items to be added to or deleted from the current Action Plan by March 12, 2004, and that the ACRS Executive Director provide a revised copy of the Action Plan at least a week before the April 14, 2004 meeting. In addition, copies of the current Action Plan should be provided to the members during the March 2004 meeting.

6) Proposed Generic Letter 2004-XX, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors"

The staff plans to issue the subject Generic Letter (GL)(pp. 12-27) for public comment in March 2004. This GL is to request the licensees to submit information to the NRC concerning the status of their compliance with 10 CFR 50.46 (b)(5), which requires long-term reactor core cooling, and with the additional plant-specific licensing basis requirements listed in the GL. This request is based on the identified potential susceptibility of PWR recirculation sump screens to debris blockage during design basis accidents requiring recirculation operation of the emergency core cooling system (ECCS) or containment spray system (CSS) and the potential for additional adverse effects due to debris blockage of flow paths necessary for ECCS and CSS recirculation and containment drainage.

We understand that the staff plans to provide the ACRS with the draft final version of this GL after reconciliation of public comments. In view of the significance of this issue and

the expected discussion of the issue of PWR sump performance at the ACRS/Commission meeting on May 6, 2004, the Committee needs to decide whether to review this GL prior to issuance for public comment or after reconciliation of public comments.

RECOMMENDATION

The Subcommittee agrees with the recommendation by Dr. Wallis that the Committee review the draft final versions of this GL after reconciliation of public comments. Also, the Thermal-Hydraulic Phenomena Subcommittee should hold a meeting to review the draft final GL and the NEI guidance document prior to referring item to the full Committee for consideration.

7) ACRS Member Notebooks

Prior to the March 2004 meeting, ACRS members received a CD containing the normal material included in the meeting notebooks. If this is an acceptable procedure, we will issue a CD each month prior to the meeting, in lieu of providing a three ring binder.

RECOMMENDATION

The Subcommittee recommends that the members decide as to whether they would like to receive a CD instead of the hard copy notebook.

8) Visit to a Nuclear Plant and Regional Office

Each year the members visit a nuclear plant and the NRC Regional Office and meet with the licensee and the Regional staff to discuss items of mutual interest. The Committee should decide on the plant to be visited and the date. We have been rotating between regions and this year's visit would normally be to Region III, however, some members have proposed plants outside of Region III. Plants for consideration are (1) Dresden, (2) LaSalle, and (3) Quad Cities.

RECOMMENDATION

The Subcommittee recommends that the members visit Dresden Nuclear Plant and the Regional Office on June 8-9, 2004.

9) Ethics Refresher Training

Mr. John Szabo, OGC, has agreed to provide ethics refresher training to the ACRS members between 1:00 and 2:00 p.m. on April 16, 2004, and answer any questions the members may have on this subject. If the members would like to send questions to Mr. Szabo in advance of the April meeting, they can do so through the ACRS Executive Director. Part of this hour will also be used to cover computer security issues and other administrative matters.

RECOMMENDATION

The Subcommittee recommends that since the agenda for the April meeting is too full, the ACRS Executive Director consider postponing this training to the May 2004 ACRS meeting.

10) Review of NRC Codes

A member of the public sent an anonymous e-mail (pp. 28-30) to Dr. Wallis, expressing concerns about the process used for reviewing NRC codes. Specifically, that person is concerned about the process for reviewing the TRACE Code. Key points made by that person include:

- Several issues are overlooked when NRC codes are under review. However, they are not overlooked when NRC reviews codes submitted by other organizations.

- Extreme adversarial environment has been present at the NRC for the past 30 years. Free and open discussion of important technical issues has not been possible for all these years.

- The ACRS Thermal-Hydraulic Phenomena Subcommittee and its consultants have individual agendas and do not listen to the very people who know the most about the subject matter presented.

- Those organizations who submit codes for NRC review have very specialized people who know exactly what is important for each and every application of their codes and experimental data. The ACRS and its consultants, on the other hand, generally do not have the time, or more importantly the inclination, to digest the material to the depth necessary to understand the important issues. The Thermal-Hydraulic Phenomena Subcommittee and its consultants and the material on which they focus and the manner on which they discuss the material are the subject of many jokes and not-so kind comments all over the industry.

- It is accepted procedure that computer codes must be verified before the models and method are validated. Generally, the codes developed under NRC funding have never undergone verification. Additionally, almost all of the validation or assessment calculation done with the NRC codes have not been done under an approved and qualified procedure.

- It does not appear that verification of the TRACE code will be performed. Also, it is not clear that the NRC has a qualified and approved QA plan in place for the TRACE code. Such plans are required by the NRC for commercial organizations.

- If the documentation for the numerical solution methods used in TRACE are studied in detail, the results will show the basic SETS solution method is based on less-than-exact methodologies.

- There is a basic problem that has never been addressed, i.e., the numerical method does not solve for the void fraction in a way that can be theoretically justified.

Mr. Caruso, ACRS staff engineer, has sent this e-mail to Mr. Rosenthal of RES. Since this seems to be an allegation, we should probably forward this to the EDO for action.

RECOMMENDATION

The Subcommittee recommends that the ACRS Executive Director refer this matter to the EDO for action.

11) License Amendment Request by Duke Power to Insert Mixed Oxide Lead Test Assemblies

On February 27, 2003, Duke Energy Corporation filed a license amendment to revise the McGuire and Catawba Technical Specifications to allow insertion of four mixed oxide (MOX) lead test assemblies at either the McGuire or the Catawba Nuclear Station. Subsequently, the Blue Ridge Environmental Defense League (BREDL) and the Nuclear Information and Resources Service filed petition to intervene, and requests for hearings.

The Atomic Safety and Licensing Board (ASLB) decided in favor of the interveners on certain issues noted below. In an Order dated February 18, 2004 (pp. 31-44) the Commission:

- Reversed the ASLB decision to allow access to BREDL to safeguards information.

- Exercised its supervisory authority and overturned another order of the ASLB that gave BREDL representatives a right to attend a safeguards-related meeting between the staff and Duke Power.

- Established a filing schedule for the filing of security-related contingencies by BREDL.

In an e-mail (p. 45) dated February 25, 2004, Dr. Powers states that during the ACRS review of the Duke's license amendment, the interveners may raise the issues that have been decided already by the Commission. The Committee should discuss how the DFO and the Subcommittee Chairman should react to the issues that may be raised by interveners during the meeting, especially on those issues that the Commission has already taken a position. He suggests that the ACRS Chairman inform the Commissioners of the potential for the public to raise the issue at the ACRS meetings and assure the Commissioners that the ACRS does not want to get involved in safeguards and security issues related to this matter on which the Commission has already taken a position.

RECOMMENDATION

The Subcommittee recommends that the Committee discuss the issue raised by Dr. Powers and decide on a course of action.

12) LINK Technologies, Inc., Report

At the request of Mr. Rosen, LINK Technologies, Inc. has prepared a report that includes recommendations for enhancing the NRC training material for inspecting a licensee's corrective action program and explores the possibility of implementing Performance Indicators in the Reactor Oversight Process for addressing the corrective action programs. Copies of the LINK report ~~has been~~ was distributed ~~to the members this morning~~.

RECOMMENDATION

The Subcommittee recommends that a representative from LINK Technologies be invited to provide a presentation to the Human Factors Subcommittee during its meeting on April 22, 2004. In addition, as agreed to by the Committee during its January 29-30, 2004 retreat, the Human Factors Subcommittee should also discuss corrective action programs, inspection and training programs, and other cross-cutting issues of the Reactor Oversight Processes during its meeting on April 22, 2004.

ANTICIPATED WORKLOAD
MARCH 3-6, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Bonaca	All Members	Larkins	Meeting with the EDO and Office Directors **(ACNW Members will participate)**	--	--	--
	Leitch	Sykes	Final review of the License Renewal Application for the Virgil C. Summer Nuclear Station	A	To support the accelerated staff schedule	**Draft**
	--	Savio/Major	Safeguards and Security **[March 3, 2004, 11:00 a.m. - 6:00 p.m.]**	--	--	--
Ford	Wallis	Jain	Resolution of Certain Items Identified by the ACRS in NUREG-1740 related to DPO on steam generator tube integrity issues	B	To provide Committee's views	**Draft**
Kress	Wallis	El-Zeftawy	Interim review of the AP1000 design	A	To identify issues of concern to the ACRS	**Draft**
Leitch	Bonaca	Jain	Final review of the License Renewal Application for the H.B. Robinson Plant	A	To support the staff schedule	**Draft**
Powers	--	Nourbakhsh/ Duraiswamy	Response to SRM on divergence in regulatory approaches between U.S. and other countries	A	To respond to the Commission SRM	--
	Apostolakis/ Shack/ Wallis	Nourbakhsh/ Duraiswamy	Criteria for evaluating the effectiveness (Quality) of the NRC Research Programs	B	To respond to the staff's request	--
Wallis	--	Caruso	Response to the 12/22/03 EDO response to the 9/30/03 ACRS report on Reg. Guide 1.82, Rev. 3	B	To provide Committee's views on the EDO response	**Draft**

ANTICIPATED WORKLOAD
APRIL 15-17, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	--	Snodderly	Staff's action plan for implementing phased approach for improving PRA quality	A	To support the staff schedule	--
Bonaca	Leitch	Sykes	Final Review of the License Renewal Application for the Ginna Nuclear Plant	A	To support the staff schedule	--
	All Members	Larkins	Preparation for meeting with the Commissioners - May 6, 2004 (1:30—3:30 p.m.)	--	--	--
Ford	--	Weston	Generic Communication regarding pressurizer dissimilar metal weld cracking issues	A	To support the staff schedule	--
Kress	--	El-Zeftawy	Options and recommendations for functional performance requirements and criteria for the containments of non-LWRs	A	To support the staff schedule for submitting this matter to the Commission	--
Leitch	Bonaca	Jain	Subcommittee Report — Interim Review of the License Renewal Application for the Dresden and Quad Cities Nuclear Plants	--	--	--
Shack	Wallis	Snodderly/Sykes	Draft Commission Paper regarding the staff's response to the March 31, 2003 SRM on risk-informing 10 CFR 50.46	A	To provide ACRS views for consideration by the Commission during its deliberation of this matter	--
Sieber	Apostolakis	Sykes	Subcommittee Report on Digital I&C matters	--	--	--

ANTICIPATED WORKLOAD
MAY 6-8, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	--	Snodderly	Status of the pilot/trial use of Reg. Guide 1.200 (Formerly DG-1122) regarding PRA Quality - **Information Briefing**	A	To support the staff schedule	--
	--	Weston	Risk Management Technical Specifications	B	To provide feedback to the staff	--
Bonaca	All Members	Larkins	Meeting with the NRC Commissioners (May 6, 1:30 - 3:30 p.m.)	--	--	--
Powers	--	Caruso	MOX Lead Test Assemblies for Catawba	A	To support the staff schedule	--
Rosen	--	El-Zeftawy	Corrective action programs, inspection and training programs, and other ROP cross-cutting issues	**B**	Provide ACRS views to the Commission (ACRS proactive initiative)	--
Sieber	--	Weston	Proposed final revision to Appendix E relating to NRC approval of changes to emergency action levels and exercise requirements for co-located licenses	Possible Larkinsgram	To support the staff schedule	--

S. Duraiswamy

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

February 19, 2004

MEMORANDUM TO: Annette L. Vietti-Cook, Secretary
Office of the Secretary of the Commission

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: ADVISORY COMMITTEE ON REACTOR SAFEGUARDS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION, MAY 6, 2004 - PROPOSED TOPICS

The ACRS is scheduled to meet with the NRC Commissioners between 1:30 and 3:30 p.m. on Thursday, May 6, 2004, to discuss items of mutual interest. Topics proposed by the ACRS are as follows:

1. Overview - Mario V. Bonaca
2. PRA Quality - George E. Apostolakis
3. NRC Safety Research Program Report - Dana A. Powers
4. Interim Review of the AP1000 Design - Thomas S. Kress
5. ESBWR Pre-application Review - Thomas S. Kress
6. Risk-Informing 10 CFR 50.46 - William J. Shack
7. PWR Sump Performance - John D. Sieber

After the Commission decides on the topics to be discussed, we will allocate appropriate time for individual topics. If necessary, we can combine items (4) and (5) to accommodate Commissioners' interest in expanded discussions of specific topics.

I would appreciate Commissioners' feedback on the topics proposed above by March 3, 2004.

cc: M. V. Bonaca

ISSUANCE OF PROPOSED GENERIC LETTER 2004-XX, "POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS"
FOR
PUBLIC COMMENT

At the request of the Office of NRR, the Committee to Review Generic Requirements (CRGR) held a meeting on February 24, 2004, to review a proposed generic letter, GL 2004-XX, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized - water Reactors" (attached). Following CRGR endorsement, the proposed generic letter will be issued for public comment. The final generic letter is scheduled to be issued in August 2004, to coincide with the release of the industry guidance document by the NEI.

Background

On June 9, 2003, having completed its technical assessment of GSI-191, the NRC issued Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors." As a result of the emergent issues discussed therein, the bulletin requested an expedited response from PWR licensees as to the status of their compliance on a mechanistic basis, with regulatory requirements concerning the ECCS and CSS recirculation functions. Addressees who were unable to assure regulatory compliance pending further analysis were asked to describe any interim compensatory measures that have been implemented or will be implemented to reduce risk until the analysis could be completed. All licensees have since responded to Bulletin 2003-01. The staff, In response to questions from CRGR responded that except Davis Besse, all licensees have chosen to implement interim compensatory measures pending further evaluation. Davis Besse is the only plant that stated that it meets the regulatory requirement.

Some of the key issues identified by CRGR are summarized below:

There is no bridge between the Bulletin 2003-01 and the proposed generic letter. What new information will be collected from the licensee. Is the staff announcing a new staff position since some plants use 50% screen blockage as their current licensing basis. Is the methodology for a mechanistic analysis (of the susceptibility of the ECCS and CSS recirculation functions to the adverse effects of post-accident debris blockage and operation with debris laden fluids) that is acceptable to the staff available to the licensee?. How the chemical effects will be addressed in the methodology? Will the acceptable methodology for a mechanistic analysis be available to the licensees by the time this generic letter is issued in August 2004? What is the staff's cost-benefit analysis for requesting information in the generic letter? The staff's current cost benefit analyses underestimates cost for modifying sump screen; does not include costs associated with resolving issues such as down-stream (of the sump screen) clogging effects (i.e., pump and throttle valve clogging) and chemical effects.

Staff's response

The staff responded that there is no change in staff's position, however, there is a fine line; the generic letter is basically a request for information. If the licensee, including those with 50% screen blockage current licensing basis, can not show compliance with the current regulations (e.g., 10CFR 50.46 (b) (5)) using acceptable mechanistic method, then he needs to bring the plant in compliance by performing modifications. Although, methodology for acceptable mechanistic method does not exist today but the NEI guidance regarding the methodology is under development and the staff will endorse it, if found acceptable. If the staff does not accept the NEI guidance, then the staff will have its own guidance for the industry ready concurrent with the issuance of the generic letter. Therefore, an acceptable mechanistic method will be available to the industry, one way or the other.

The staff conceded that the issues of chemical effect, down-stream blockage effect, and redefinition of break size associated with large break LOCA are the wild cards. The staff agreed to revisit their current cost-benefit analyses for a reality check. The staff agreed to revise the text of the proposed generic letter to provide a bridge over the Bulletin 2003-01 and incorporate other enhancements as suggested by the CRGR.

The staff will resubmit the final draft of the proposed generic letter to the CRGR prior to its release for public comment.

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

(Date to be issued for comment)

NRC GENERIC LETTER 2004-XX: POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS

Addressees

All holders of operating licenses for pressurized-water nuclear power reactors, except those who have 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 generic letter to:

(1) Request that addressees submit information to the NRC concerning the status of their compliance with 10 CFR 50.46(b)(5), which requires long-term reactor core cooling, and with the additional plant-specific licensing basis requirements listed in this generic letter. This request is based on the identified potential susceptibility of pressurized-water reactor (PWR) recirculation sump screens to debris blockage during design basis accidents requiring recirculation operation of the emergency core cooling system (ECCS) or containment spray system (CSS) and the potential for additional adverse effects due to debris blockage of flowpaths necessary for ECCS and CSS recirculation and containment drainage.

(2) Require addressees to provide the NRC a written response in accordance with 10 CFR 50.54(f).

Background

In 1979, as a result of evolving staff concerns related to the adequacy of PWR recirculation sump designs, the NRC opened Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance." To support the resolution of USI A-43, the NRC undertook an extensive

ATTACHMENT 1

research program, the technical findings of which are summarized in NUREG-0897, "Containment Emergency Sump Performance," dated October 1985. The resolution of USI A-43 was subsequently documented in Generic Letter (GL) 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," dated December 3, 1985. Although the staff's regulatory analysis concerning USI A-43 did not support imposing new sump performance requirements upon licensees of operating PWRs or boiling-water reactors (BWRs), the staff recommended in GL 85-22 that all affected reactor licensees replace the 50-percent blockage assumption (under which most nuclear power plants had been licensed) with a comprehensive, mechanistic assessment of plant-specific debris blockage potential for future modifications related to sump performance, such as thermal insulation changeouts. The staff also updated the NRC's regulatory guidance, including Section 6.2.2 of the Standard Review Plan (NUREG-0800) and Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," to reflect the USI A-43 technical findings documented in NUREG-0897.

Following the resolution of USI A-43 in 1985, several events occurred that challenged the conclusion that no new requirements were necessary to prevent the clogging of ECCS strainers at operating BWRs:

- On July 28, 1992, at Barsebäck Unit 2, a Swedish BWR, the spurious opening of a pilot-operated relief valve led to the plugging of two containment vessel spray system suction strainers with mineral wool and required operators to shut down the spray pumps and backflush the strainers.

- In 1993, at Perry Unit 1, two events occurred during which ECCS strainers became plugged with debris. On January 16, ECCS strainers were plugged with suppression pool particulate matter, and on April 14, an ECCS strainer was plugged with glass fiber from ventilation filters that had fallen into the suppression pool. On both occasions, the affected ECCS strainers were deformed by excessive differential pressure created by the debris plugging.

- On September 11, 1995, at Limerick Unit 1, following a manual scram due to a stuck-open safety/relief valve, operators observed fluctuating flow and pump motor current on the A loop of suppression pool cooling. The licensee later attributed these indications to a thin mat of fiber and sludge which had accumulated on the suction strainer.

In response to these ECCS suction strainer plugging events, the NRC issued several generic communications, including Bulletin 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated February 18, 1994, Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," dated October 17, 1995, and Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," dated May 6, 1996. These bulletins requested that BWR licensees implement appropriate procedural measures, maintenance practices, and plant modifications to minimize the potential for the clogging of ECCS suction strainers by debris accumulation following a loss-of-coolant accident (LOCA). The NRC staff has concluded that all BWR licensees have sufficiently addressed these

bulletins.
However, findings from research to resolve the BWR strainer clogging issue have raised questions concerning the adequacy of PWR sump designs. In comparison to the technical findings of the USI A-43 research program concerning PWRs, the new research findings demonstrate that the amount of debris generated by a high-energy line break (HELB) could be greater, that the debris could be finer (and, thus, more easily transportable), and that certain combinations of debris (e.g., fibrous material plus particulate material) could result in a substantially greater head loss than an equivalent amount of either type of debris alone. These new research findings prompted the NRC to open Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance." The objective of GSI-191 is to ensure that post-accident debris blockage will not impede or prevent the operation of the ECCS and CSS in recirculation mode at PWRs during LOCAs or other HELB accidents for which sump recirculation is required.

On June 9, 2003, having completed its technical assessment of GSI-191 (summarized below in the Discussion section of this generic letter), the NRC issued Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors." As a result of the emergent issues discussed therein, the bulletin requested an expedited response from PWR licensees as to the status of their compliance on a mechanistic basis, with regulatory requirements concerning the ECCS and CSS recirculation functions. Addressees who were unable to assure regulatory compliance pending further analysis were asked to describe any interim compensatory measures that have been implemented or will be implemented to reduce risk until the analysis could be completed. All licensees have since responded to Bulletin 2003-01.

In response to Bulletin 2003-01, PWR licensees that were unable to ensure regulatory compliance implemented compensatory measures to reduce risk or otherwise enhance the capability of the ECCS and CSS recirculation functions. During the process of resolving the potential concerns identified in this generic letter, the revised analysis of sump performance may affect addressees' understanding of their facilities' ECCS and CSS recirculation capabilities. In accordance with GL 91-18, Revision 1, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," dated October 8, 1997, addressees may find it necessary to reevaluate the adequacy of their compensatory measures in light of the new information and take further action as appropriate and necessary.

The NRC has developed a Web page to keep the public informed of generic activities on PWR sump performance (http://www.nrc.gov/reactors/operating/ops-experience/pwr-sump-performance.html). This page provides links to information on PWR sump performance issues, along with documentation of NRC interactions with industry (industry submittals, meeting notices, presentation materials, and meeting summaries). The NRC will continue to update this Web page as new information becomes available.

Discussion

In the event of a HELB inside the containment of a PWR, energetic pressure waves and fluid jets would impinge upon materials in the vicinity of the break, such as thermal insulation,

coatings, and concrete, causing them to become damaged and dislodged. Debris could also be generated through secondary mechanisms, such as severe post-accident temperature and humidity conditions, flooding of the lower containment, and the impact of containment spray droplets. Through transport methods such as entrainment in the steam/water flows issuing from the break and containment spray washdown, a fraction of the generated debris and foreign material in the containment would be transported to the pool of water formed on the containment floor. Subsequently, if the ECCS or CSS pumps were to take suction from the recirculation sump, the debris suspended in the containment pool would begin to accumulate on the sump screen or be transported through the associated system. The accumulation of this suspended debris on the sump screen could create a roughly uniform covering on the screen, referred to as a debris bed, which would tend to increase the head loss across the screen through a filtering action. If a sufficient amount of debris were to accumulate, the debris bed would reach a critical thickness at which the head loss across the debris bed would exceed the net positive section head (NPSH) margin required to ensure the successful operation of the ECCS and CSS pumps in recirculation mode. A loss of NPSH margin for the ECCS or CSS pumps as a result of the accumulation of debris on the recirculation sump screen, referred to as sump clogging, could result in degraded pump performance and eventual pump failure. Debris could also plug or wear close tolerance components within the ECCS or CSS systems. The effect of this plugging or wear may cause a component to degrade to the point where it may be unable to perform its designated function (i.e. pump fluid, maintain system pressure, or pass and control system flow.)

Assessing the likelihood of the ECCS and CSS pumps at domestic PWRs experiencing a debris-induced loss of NPSH margin during sump recirculation was the primary objective of the NRC's technical assessment of GSI-191. The NRC's technical assessment culminated in a parametric study that mechanistically treated phenomena associated with debris blockage using analytical models of domestic PWRs generated with a combination of generic and plant-specific data. As documented in Volume 1 of NUREG/CR-6762, "GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," dated August 2002, the GSI-191 parametric study concludes that recirculation sump clogging is a credible concern for domestic PWRs. As a result of limitations with respect to plant-specific data and other modeling uncertainties, however, the parametric study does not definitively identify whether or not particular PWR plants are vulnerable to sump clogging when phenomena associated with debris blockage are modeled mechanistically.

The methodology employed by the GSI-191 parametric study is based upon the substantial body of test data and analyses that are documented in technical reports generated during the NRC's GSI-191 research program and earlier technical reports generated by the NRC and the industry during the resolution of the BWR strainer clogging issue and USI A-43. These pertinent technical reports, which cover debris generation, transport, accumulation, and head loss, are incorporated by reference into the GSI-191 parametric study:

- NUREG/CR-6770, "GSI-191: Thermal-Hydraulic Response of PWR Reactor Coolant System and Containments to Selected Accident Sequences," dated August 2002.

- NUREG/CR-6762, Vol. 3, "GSI-191 Technical Assessment: Development of Debris Generation Quantities in Support of the Parametric Evaluation," dated August 2002.

- NUREG/CR-6762, Vol. 4, "GSI-191 Technical Assessment: Development of Debris Transport Fractions in Support of the Parametric Evaluation," dated August 2002.

- NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," dated October 1995.

In light of the credibility of the concerns identified above, the NRC staff has determined that it is appropriate to request that addressees submit information to confirm their compliance with NRC regulations and other regulatory requirements pertaining to post-accident debris blockage, using analysis methods that mechanistically account for debris generation and transport, post accident equipment and systems operation with debris laden fluid or, as applicable, a description of plans to restore compliance.

In addition to demonstrating the potential for debris to clog containment recirculation sumps, operational experience and the NRC's technical assessment of GSI-191 have also identified three integrally related modes by which post-accident debris blockage could adversely affect the sump screen's design function of intercepting debris that could impede or prevent the operation of the ECCS and CSS in recirculation mode.

First, as a result of the 50-percent blockage assumption, most PWR sump screens were designed assuming that relatively small structural loadings would result from the differential pressure associated with debris blockage. Consequently, PWR sump screens may not be capable of accommodating the increased structural loadings that would occur due to mechanistically determined debris beds that cover essentially the entire screen surface. Inadequate structural reinforcement of a sump screen may result in its deformation, damage, or failure, which could allow large quantities of debris to be ingested into the ECCS and CSS piping, pumps, and other components, potentially leading to their clogging or failure. The ECCS strainer plugging and deformation events that occurred at Perry Unit 1 (further described in Information Notice (IN) 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," dated April 26, 1993, and LER 50-440/93-011, "Excessive Strainer Differential Pressure Across the RHR Suction Strainer Could Have Compromised Long Term Cooling During Post-LOCA Operation," submitted May 19, 1993), demonstrate the credibility of this concern for screens and strainers that have not been designed with adequate reinforcement.

Second, in some PWR containments, the flowpaths by which containment spray or break flows return to the recirculation sump may include "choke-points," where the flowpath becomes so constricted that it could become blocked with debris following a HELB. Examples of potential choke-points are drains for pools, cavities, isolated containment compartments, and constricted drainage paths between physically separated containment elevations. Debris blockage at certain choke-points could hold up substantial amounts of water required for adequate recirculation or cause the water to be diverted into containment volumes that do not drain to the recirculation sump. The holdup or diversion of water assumed to be available to support sump recirculation could result in an available NPSH for ECCS and CSS pumps that is lower than the analyzed value, thereby reducing assurance that recirculation would successfully function. A reduced available NPSH directly concerns sump screen design because the NPSH margin of

the ECCS and CSS pumps must be conservatively calculated to determine correctly the required surface area of passive sump screens when mechanistically determined debris loadings are considered. Although the parametric study (NUREG/CR-6762, Volume 1) did not analyze in detail the potential for the holdup or diversion of recirculation sump inventory, the NRC's GSI-191 research identified this phenomenon as an important and potentially credible concern. A number of LERs associated with this concern have also been generated, which further confirms its credibility and potential significance:

- LER 50-369/90-012, "Loose Material Was Located in Upper Containment During Unit Operation Because of an Inappropriate Action," McGuire Unit 1, submitted August 30, 1990.
- LER 50-266/97-006, "Potential Refueling Cavity Drain Failure Could Affect Accident Mitigation," Point Beach Unit 1, submitted February 19, 1997.
- LER 50-455/97-001, "Unit 2 Containment Drain System Clogged Due to Debris," Byron Unit 2, submitted April 17, 1997.
- LER 50-269/97-010, "Inadequate Analysis of ECCS Sump Inventory Due to Inadequate Design Analysis," Oconee Unit 1, submitted January 8, 1998.
- LER 50-315/98-017, "Debris Recovered from Ice Condenser Represents Unanalyzed Condition," D.C. Cook Unit 1, submitted July 1, 1998.

Third, debris blockage at flow restrictions within the ECCS recirculation flowpath downstream of the sump screen is a potential concern for PWRs. Debris that is capable of passing through the recirculation sump screen may have the potential to become lodged at a downstream flow restriction, such as a high-pressure safety injection (HPSI) throttle valve or fuel assembly inlet debris screen. Debris blockage at such flow restrictions in the ECCS flowpath could impede or prevent the recirculation of coolant to the reactor core, thereby leading to inadequate core cooling. Similarly, debris blockage at flow restrictions in the CSS flowpath, such as a containment spray nozzle, could impede or prevent CSS recirculation, thereby leading to inadequate containment heat removal. Debris may also accumulate in close tolerance sub-components of pumps and valves. The effect may either be to plug the sub-component thereby rendering the component unable to perform its function or to wear critical close tolerance sub-components to the point at which component or system operation is degraded and unable to fully perform its function. Considering the recirculation sump screen's design function of intercepting potentially harmful debris, it is essential that the screen openings are adequately sized and that the sump screen's current configuration is free of gaps or breaches which could compromise the ECCS and CSS recirculation functions. It is also essential that system components are designed and evaluated to be able to operate with debris laden fluid as necessary post-LOCA.

To assist in determining on a plant-specific basis whether compliance exists with 10 CFR 50.46(b)(5) and additional plant-specific licensing basis requirements concerning the ECCS and CSS recirculation functions, addressees may use the guidance contained in Regulatory Guide 1.82 (RG 1.82), Revision 3, "Water Sources for Long-Term Recirculation

Cooling Following a Loss-of-Coolant Accident," dated November 2003. In addition, the NRC staff is reviewing generic industry guidance and will issue a safety evaluation report endorsing portions or all of the generic industry guidance, if found acceptable. Once approved, this guidance may also be used to assist in determining the status of regulatory compliance. Individual addressees may also develop alternative approaches to those named in this paragraph for determining the status of their regulatory compliance; however, additional staff review may be required to assess the adequacy of such approaches.

Applicable Regulatory Requirements

NRC regulations in Title 10, of the *Code of Federal Regulations* Section 50.46,(10 CFR 50.46), require that the ECCS must satisfy five criteria, one of which is to provide the capability for long-term cooling of the reactor core following a LOCA. The ECCS must have the capability to provide decay heat removal, such that the core temperature is maintained at an acceptably low value for the extended period of time required by the long-lived radioactivity remaining in the core. For PWRs licensed to the General Design Criteria (GDCs) in Appendix A to 10 CFR Part 50, GDC 35 specifies additional ECCS requirements.

Similarly, for PWRs licensed to the GDCs in Appendix A to 10 CFR Part 50, GDC 38 provides requirements for containment heat removal systems, and GDC 41 provides requirements for containment atmosphere cleanup. Many PWR licensees credit a CSS, at least in part, with performing the safety functions to satisfy these requirements, and PWRs that are not licensed to the GDCs may similarly credit a CSS to satisfy licensing basis requirements. In addition, PWR licensees may credit a CSS with reducing the accident source term to meet the limits of 10 CFR Part 100 or 10 CFR 50.67.

Criterion XVI (Corrective Action) of Appendix B to 10 CFR Part 50 states that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. For significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions.

If, in the course of preparing a response to the requested information, an addressee determines that its facility is not in compliance with the Commission's requirements, the addressee is expected to take appropriate action in accordance with requirements of Appendix B to 10CFR Part 50 and the plant technical specifications to restore the facility to compliance.

Applicable Regulatory Guidance[1]

Regulatory Guide 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," November 2003.

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[1] The NRC staff is currently reviewing evaluation guidance developed by the industry. The NRC staff will document its review in a safety evaluation which licensees can reference as regulatory guidance.

Requested Information

All addressees are requested to provide the following information:

(1) Within 60 days of the date of this generic letter, provide the following information:

(a) A statement of whether or not you plan to perform a mechanistic analysis of the susceptibility of the ECCS and CSS recirculation functions for your reactor to adverse effects of post-accident debris blockage and operation with debris laden fluids identified in this generic letter. If a mechanistic analysis will be performed, state the planned methodology to be used and the planned completion date. If a mechanistic analysis will not be performed, please provide justification that ECCS and CSS recirculation functions will not be adversely effected by post-accident debris blockage

(b) A statement of whether or not you plan to perform a containment walkdown surveillance in support of the analysis of the susceptibility of the ECCS and CSS recirculation functions to the adverse effects of debris blockage identified in this generic letter. Provide justification if no containment walkdown surveillance will be performed. If a containment walkdown surveillance will be performed, state the planned methodology to be used and the planned completion date. If a containment walkdown surveillance has already been performed, state the methodology used, the completion date, and the results of the surveillance.

2. No later than April 1, 2005, provide the following information:

(a) A submittal that contains the methodology that was used to perform a mechanistic analysis of the suscepibility of the ECCS and CSS recirculation functions to the adverse effects of post-accident debris blockage and operation with debris laden fluids. The submittal may reference a guidance document (e.g. Regulatory Guide 1.82, industry guidance) or other methodology previously submitted to the NRC. The documents to be submitted or referenced include the methodology for conducting a supporting containment walkdown surveillance used to identify potential debris sources and other pertinent containment characteristics.

(b) A general description of and implementation schedule for all corrective actions, including any plant modifications that may be necessary to ensure compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. Provide justification for any corrective action that will not be completed by the end of the refueling outage listed above or December 30, 2007.

(c) Provide confirmation that the ECCS and CSS recirculation functions under mechanistically determined debris loading conditions are or will be in compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. This submittal should also address the configuration of the plant that will exist once all modifications required for regulatory compliance

(e) A description of any existing or planned programmatic controls that will ensure that, in the future, potential sources of debris introduced into containment (e.g., insulations, signs, coatings, and foreign materials) will be assessed for potential adverse effects on the ECCS and CSS recirculation functions. Addressees may reference their responses to GL 98-04 to the extent that their responses address these specific foreign material control issues.

Required Response

In accordance with 10 CFR 50.54(f), the subject PWR addressees are required to submit written responses to this generic letter. This information is sought to verify licensees' compliance with current licensing basis for the subject PWR addressees. The addressees have two options:

(1) addressees may choose to submit written responses providing the information requested above within the requested time periods, or

(2) addressees who choose not to provide information requested or cannot meet the requested completion dates are required to submit written responses within 15 days of the date of this generic letter. The responses must address any alternative course of action proposed, including the basis for the acceptability of the proposed alternative course of action.

The required written responses should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, 11555 Rockville Pike, Rockville, Maryland 20852, under oath or affirmation under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, a copy of a response should be submitted to the appropriate regional administrator.

The NRC staff will review the responses to this generic letter and will notify affected addressees if concerns are identified regarding compliance with NRC regulations and their current licensing bases. The staff may also conduct audits and/or inspections to determine addressees' effectiveness in addressing the generic letter.

Reasons for Information Request

As discussed above, recent research and analysis suggests that (1) the potential for the failure of the ECCS and CSS recirculation functions as a result of debris blockage is not adequately addressed in most PWR licensees' current safety analyses, and (2) the ECCS and CSS recirculation functions at a significant number of operating PWRs could become degraded as a result of the potential effects of debris blockage or extended operation with debris laden fluids identified in this generic letter. An ECCS that is incapable of providing long-term reactor core cooling through recirculation operation would be in violation of 10 CFR 50.46. A CSS that is incapable of functioning in recirculation mode may not comply with GDCs 38 and 41 or other plant-specific licensing requirements or safety analyses. Therefore, the information requested in this generic letter is necessary to permit the assessment of plant-specific compliance with NRC regulations.

The NRC staff will also use the requested information to (1) determine whether a sample auditing approach is acceptable for verifying that addressees have resolved the concerns identified in this generic letter, (2) assist in determining which addressees would be subject to the proposed sample audits, (3) provide confidence that any nonaudited addressees have addressed the concerns identified in this generic letter, and (4) assess the need for and guide the development of any additional regulatory actions that may be necessary to address the adequacy of the ECCS and CSS recirculation functions under mechanistically determined debris loading conditions determined using NRC approved methodology.

Related Generic Communications

- Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors," June 9, 2003.
- Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," May 6, 1996.
- Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in the Suppression Pool Cooling Mode," October 17, 1995.
- Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," May 11, 1993.
- Bulletin 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," February 18, 1994.
- Generic Letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," July 14, 1998.
- Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," October 7, 1997.
- Generic Letter 85-22, "Potential For Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," December 3, 1985.
- Information Notice 97-13, "Deficient Conditions Associated With Protective Coatings at Nuclear Power Plants," March 24, 1997.
- Information Notice 96-59, "Potential Degradation of Post Loss-of-Coolant Recirculation Capability as a Result of Debris," October 30, 1996.
- Information Notice 96-55, "Inadequate Net Positive Suction Head of Emergency Core Cooling and Containment Heat Removal Pumps Under Design Basis Accident Conditions," October 22, 1996.

- Information Notice 96-27, "Potential Clogging of High Pressure Safety Injection Throttle Valves During Recirculation," May 1, 1996.
- Information Notice 96-10, "Potential Blockage by Debris of Safety System Piping Which Is Not Used During Normal Operation or Tested During Surveillances," February 13, 1996.
- Information Notice 95-47, "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage," October 4, 1995.
- Information Notice 95-47, Revision 1, "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage," November 30, 1995.
- Information Notice 95-06, "Potential Blockage of Safety-Related Strainers by Material Brought Inside Containment," January 25, 1995.
- Information Notice 94-57, "Debris in Containment and the Residual Heat Removal System," August 12, 1994.
- Information Notice 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," April 26, 1993.
- Information Notice 93-34, Supplement 1, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," May 6, 1993.
- Information Notice 92-85, "Potential Failures of Emergency Core Cooling Systems Caused by Foreign Material Blockage," December 23, 1992.
- Information Notice 92-71, "Partial Plugging of Suppression Pool Strainers at a Foreign BWR," September 30, 1992.
- Information Notice 89-79, "Degraded Coatings and Corrosion of Steel Containment Vessels," December 1, 1989.
- Information Notice 89-79, Supplement 1, "Degraded Coatings and Corrosion of Steel Containment Vessels," June 29, 1990.
- Information Notice 89-77, "Debris in Containment Emergency Sumps and Incorrect Screen Configurations," November 21, 1989.
- Information Notice 88-28, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," May 19, 1988.

Backfit Discussion

Under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10CFR 50.54(f), this generic letter transmits an information request for the purpose of verifying compliance with existing applicable regulatory requirements (see the Applicable Regulatory Requirements section of this generic letter). Specifically, the required information will enable the NRC staff to determine whether the emergency core cooling system (ECCS) and containment spray system (CSS) at reactor facilities are able to perform their safety functions following all postulated accidents for which ECCS or CSS recirculation is required while taking into account the adverse effects of post-accident debris blockage and operation with debris laden fluids. No backfit is either intended or approved by the issuance of this generic letter, and the staff has not performed a backfit analysis.

Small Business Regulatory Enforcement Fairness Act

The NRC has determined that this generic letter is not subject to the Small Business Regulatory Enforcement Fairness Act of 1996.

Federal Register Notification

The NRC published a notice of opportunity for public comment on this generic letter in the *Federal Register* on ..... . In addition, the NRC has provided opportunities for public comment at several public meetings. As the resolution of this matter progresses, the NRC will continue to provide opportunities for further public involvement.

Paperwork Reduction Act Statement

This generic letter contains information collections that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget (OMB) under approval number XXXX-XXXX, which expires on XXX XX, XXXX.

The burden to the public for these mandatory information collections is estimated to average 1000 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the necessary data, and completing and reviewing the information collections. Send comments regarding this burden estimate or any other aspect of these information collections, including suggestions for reducing the burden, to the Records Management Branch, Mail Stop T-6 E6, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail to INFOCOLLECTS@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

The NRC may neither conduct nor sponsor, and an individual is not required to respond to, an information collection unless the requesting document displays a currently valid OMB control number.

If you have any questions about this matter, please contact the technical contacts or lead project manager listed below.

Bruce A. Boger, Director
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

Technical Contacts:

Ralph Architzel, NRR
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Lead Project Manager:

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301-415-1445
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From: Ralph Caruso
To: Graham B. Wallis
Date: 2/20/04 4:26PM
Subject: Re: The TRACE Code

Graham,

No, this name does not ring any bells at all. I think it might be a good idea to pass this along to RES, if you have no objection. I will also forward it to John Larkins. I think you should not send out your reply till we have had some time to think about this. It might be something that should be taken as an allegation, and we should talk to the agency allegation people before we put out something publically.

I have your response, and I still think that we need to talk, first. I am out all next week, but I will forward this to RES, and have them start to think about how they want to respond.

Ralph

>>> Graham B. Wallis <Graham.B.Wallis@Dartmouth.EDU> 02/20/04 04:21PM >>>
Do you know this guy? I have written a reply which I will forward.
G.

--- Forwarded Message from Fritz Wuehler <fritz@spamexpire-200402.rodent.frell.eu.org> ---

>From: Fritz Wuehler <fritz@spamexpire-200402.rodent.frell.eu.org>
>Comments: This message did not originate from the Sender address above. It was remailed automatically by anonymizing remailer software. Please report problems or inappropriate use to the remailer administrator at <abuse@remailer.frell.eu.org>.
>To: Graham.Wallis@Dartmouth.EDU
>Subject: The TRACE Code
>Precedence: anon
>Date: Fri, 20 Feb 2004 20:58:53 +0100

Professor Wallis:

There are several issues that need to be addressed relative to the TRACE code. Generally, these issues are overlooked when NRC codes are under review. They are not however overlooked when the NRC reviews codes submitted by other organizations for review and approval. The issues are discussed following the next paragraph. The next paragraph is an aside that however needs to be addressed by the NRC and the ACRS T/H Subcommittee.

It is a very unfortunate situation in that the extreme adversarial environment that has been present at the NRC for the past thirty years or so makes this form of communication necessary. Free and open discussion of the technical issues that are truly important has not been possible for all these years. The ACRS T/H Subcommittee and its Consultants have been filled with persons who have individual agendas and who do not listen to the very people who know the most about the subject matter that is presented. Those organizations that submit codes for review have hundreds of thousands of very specialized and focused man-hours invested in their products. These people know exactly what is important for each and every application of their codes and experimental data. The ACRS and it consultants, on the other hand,

generally do not have the time, or more importantly the inclination, to digest the material presented to the depth necessary to understand the important items that really matter in any application. The personal agendas of the T/H Subcommittee and its Consultants, as reflected in the ACRS transcripts, almost never are important to the practical issues of an application. Quite frankly, the T/H Subcommittee and its Consultants and the material on which they focus and the manner on which they discuss the material are the subject of many jokes and not-so-kind comments all over the industry.

The issues that are being overlooked relative to the TRACE code at this stage in its development include:

1. Independent verification of the coding.

2. A fundamental issue associated with the numerical methods used in the code.

Independent Verification

It is accepted procedure that software in computer codes must be verified before the models and methods are validated. Verification is the process of ensuring that the equations used in the code have been correctly coded. Validation is the process of ensuring that the correct equations have been chosen and coded. Verification must always precede validation, and that is the methodology applied to codes submitted to the NRC by all commercial organizations.

Generally, the computer codes developed under NRC funding have never undergone verification. Additionally, the validation procedure applied to these codes has not measured up to the standards required by the NRC for commercial organizations. Almost all the so-called "validation" or "assessment" calculations done with the NRC codes have not been done under an approved and qualified procedure with "frozen" versions of the software.

I have not seen that verification of the coding in the TRACE code is to be performed. To proceed to validation without verification invalidates the validation process. Additionally, it is not clear that the NRC has a qualified and approved Q/A plan in place for TRACE. Such plans are required of commercial organizations by the NRC.

Issues with the Numerical Methods

If the documentation for the numerical solution methods used in TRACE, both the code manuals and papers in the literature, are studied in detail the results will show that the basic SETS solution method is based on less-than-exact methodologies. Many solution orders for the equations were simply experimented with until one that "works" was discovered. While this approach is less than satisfactory from a theoretical view, it might be called an "engineering solution".
The following discussion is based on the documentation given in the (1) TRAC-P manual of NUREG/CR-5673, LA-12031-M, 1993, and (2) the draft of the manual for the first F-90 version of TRAC/M given in the LA-UR-00-910, 2000. The latter document is basically the former only

re-arranged. The latter document is also, I think, a rough first cut at the TRAC part of the TRACE manual. The latter document, (2), will be cited in the following discussion, although the exact same material can be found in the former.

Generally, no multi-step method actually satisfies the original FDEs, and the many approximations used in the TRAC SETS method are somewhat out the the ordinary relative to almost all other numerical solution methods for transient compressible fluid flow. The lack of satisfying the original FDEs and the many approximations can only be appreciated after digging through the manuals in a very, very detailed study. But these are not the main issues here, however.

There is a basic problem that has never been addressed and it is kind of hard to dig out of the documentation. This basic problem is as follows. The numerical method does not solve for the void fraction in a way that can be theoretically justified. A reference to section 2.1.8.2.4 at the bottom of page 2-31 of (2) is the only clue in the manual for how the void fraction at the new-time level is obtained. Section 2.1.8.2.4 is on page 2-61 of (2), 30 pages from where it is needed. The material on page 2-61 states that a system of equations is set up to obtain the new-time level values for the void fraction and the other EOS variables. The system is based on the solution of products of void and density and void-density-energy from the mass and energy stabilizer step of the SETS method plus the EOS with pressure and temperature as independent variables. The discussion given in the manuals is correct and the system of non-linear equations can easily be discovered and the iterative Newton-Raphson method applied to the solution of the system to get the pressure, phase temperatures, and void fraction.

But, here is the basic issue. As discussed on page 2-61, the system of non-linear equations is treated as a system of un-coupled linear equations and a one-shot step, without iteration is all that is done. Note that coupled linear equations require iteration to obtain a solution. Most importantly, all the quantities determined by this one-shot rough estimate are discarded except for the void fraction. This means that none of the results from the one-shot evaluation will satisfy the equations from which they were obtained and only the void fraction from this "solution" is retained. Thus, just as in the case of the RELAP5 code, the non-linear EOS is not satisfied. Additionally one must wonder exactly what the "void fraction" "calculated" in the TRAC SETS manner actually represents.

CC: Jack Rosenthal; John Larkins

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED 02/18/04

COMMISSIONERS

SERVED 02/18/04

Nils J. Diaz, Chairman
Edward McGaffigan, Jr.
Jeffrey S. Merrifield

In the Matter of	)	
	)	
DUKE ENERGY CORPORATION	)	
	)	Docket Nos. 50-413-OLA, 50-414-OLA
(Catawba Nuclear Station,	)	
Units 1 and 2)	)	
	)	

CLI-04-06

MEMORANDUM AND ORDER

In this license amendment proceeding to authorize the use of four lead test assemblies of mixed oxide fuel in one of Duke Energy Corporation's Catawba commercial nuclear reactors, the Commission grants the NRC Staff's petition for interlocutory review. We reverse the Licensing Board's decision to provide a hearing petitioner, the Blue Ridge Environmental Defense League ("BREDL"), access to confidential NRC "safeguards" information. As we see the case, there has been no showing that BREDL has a "need to know" the information -- *i.e.*, no showing that access to the information is indispensable to BREDL's opportunity to frame litigable contentions.

We also take this opportunity to exercise our general supervisory authority to overturn a recent (unpublished) Board order, dated February 4, 2004, giving BREDL's representatives a right to attend a confidential, safeguards-related, meeting between the NRC staff and the licensee. NRC licensing boards have no power to superintend the NRC staff's regulatory reviews or, in particular, to direct the Staff to admit particular individuals or groups to non-adjudicatory meetings.

I. BACKGROUND

On February 27, 2003, Duke Energy Corporation filed a license amendment request to revise the McGuire and Catawba Technical Specifications to allow insertion of four mixed oxide ("MOX")[1] lead test assemblies at either the McGuire or the Catawba Nuclear Station. After publication of a notice of opportunity for hearing in the *Federal Register*,[2] the Nuclear Information and Resource Service and BREDL filed petitions to intervene and requests for hearing. Neither Duke nor the NRC Staff contested the standing of the two organizations to seek a hearing.

To formulate contentions about security, BREDL's counsel requested access to Duke's September 15, 2003, security-related submittal to the NRC. This document, which contains confidential safeguards information,[3] includes a revision to the Duke Energy Corporation Nuclear Security and Contingency Plan and a related request for exemption from certain requirements in 10 C.F.R. Parts 11 and 73 associated with the proposed use of MOX fuel at Catawba. It contains, in effect, the special security arrangements Duke plans to put in place during the time it is storing the unirradiated MOX test assemblies at Catawba.

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[1]MOX is a mixture of uranium and plutonium oxides. As a part of the United States - Russian Federation plutonium disposition program, the U.S. Department of Energy plans to dispose of weapons grade plutonium by converting it to MOX fuel and using the fuel in commercial nuclear reactors. Current plans are to test four assemblies by placing them into the 193-assembly core in one of the reactors at Catawba. After irradiation of the test assemblies, they will be tested to verify their properties. A later license amendment request is contemplated for "batch use" of the fuel.

[2]*See* "Duke Energy Corporation et al., Catawba Nuclear Station, Units 1 and 2; McGuire Nuclear Station, Units 1 and 2; Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing," 68 Fed. Reg. 44,107 (July 25, 2003).

[3] Safeguards information is protected from public disclosure under the authority of section 147 of the Atomic Energy Act, 42 U.S.C. § 2167. The designation covers, among other things, "security measures (including security plans, procedures and equipment) for the physical protection" of special nuclear material, byproduct material, source material, and safety-significant equipment at nuclear reactors. NRC regulations establish specific requirements for protecting and accessing safeguards-designated information. *See* 10 C.F.R. § 73.21.

Duke requested the Board to enter a protective order upon execution of non-disclosure affidavits by BREDL's attorney, Diane Curran, and expert witness, Edwin Lyman.[4] The Board entered such an order on December 15, 2003,[5] and Ms. Curran and Dr. Lyman thereafter viewed Duke's MOX-related security submittal.

Subsequently, the present controversy developed when BREDL's attorney requested access to additional safeguards and classified documents from the NRC Staff and the NRC Staff declined to provide them. BREDL sought, among other things, certain orders the NRC issued in 2003 to modify licenses at reactor facilities,[6] including safeguards and classified information about the design basis threat for commercial nuclear reactors and individual Category I facilities.[7]

BREDL believes the safeguards information it requested is necessary to formulate contentions on Duke's MOX-related security plan submittal.[8] BREDL considers the documents to contain the "law," or standard, on which to base its disputes with Duke. According to BREDL,

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[4]NIRS's representative indicated that NIRS would not file any security-related contentions; thus, NIRS is not included in the protective order.

[5]*See Duke Energy Corp.* (Catawba Nuclear Station, Units 1 and 2), unpublished "Memorandum and Order (Protective Order Governing Duke Energy Corporation's September 15, 2003 Security Plan Submittal)" (Dec. 15, 2003).

[6] A public version of these orders appears in the *Federal Register. See, e.g., All Power Reactor Licensees, Order Modifying Licenses (Effective Immediately)*, 68 Fed. Reg. 24,517 (May 7, 2003). But, just as NRC regulations do not specify the precise number of intruders or type of weapons a facility must protect against (the design basis threat), neither do the public versions of the NRC's 2003 security orders. NRC security regulations, for the most part, contain general requirements that are implemented through details contained in confidential (safeguards) licensee security plans. *See, e.g.,* 10 C.F.R. §§ 73.1, 73.55. Similarly, the NRC's security orders are accompanied by confidential, safeguards-designated attachments setting out sensitive security-related details. It is those details that BREDL seeks.

[7]Category I facilities are licensed to possess formula quantities of strategic special nuclear material. There currently are two such facilities in the United States. *See* "Final Rule: Material Control and Accounting Requirements," 67 Fed. Reg. 78,130-31 (Dec. 23, 2002).

[8]BREDL filed its non-security-related contentions on October 21, 2003.

Duke's exemption requests cannot be evaluated without consideration of the requirements from which the exemptions are sought or the substitute standard that Duke proposes to satisfy instead. Without obtaining access to the additional safeguards information it seeks (*i.e.*, information besides the safeguards Duke security submittal that BREDL already has), BREDL believes that it would have to base its security contentions on a "sheer guess" of what the standards might be.

The NRC staff and Duke opposed BREDL's request for additional documents. The Staff maintains that it will review Duke's security submittal on the basis of currently applicable standards only -- 10 C.F.R. §§ 73.5 and 11.9 -- which are available in the *Code of Federal Regulations.*[9] The Staff represents that it will not itself use the requested safeguards documents in its planned review of Duke's license amendment application. Thus, says the Staff, BREDL does not need the material. Duke concentrates on the information in the security submittal itself and states that BREDL should be able to formulate security contentions from that submittal and the publicly available regulatory requirements.

On motion by BREDL, a quorum of the Board[10] ordered the Staff to provide access to the requested safeguards documents, but not to the classified documents, by February 2, 2004.[11] The Board also granted BREDL an extension of time to file its security-related

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[9]Exemptions for regulations dealing with physical protection of plants and materials appear at 10 C.F.R. § 73.5, and exemptions from criteria and procedures for determining eligibility for access to or control over special nuclear material appear at 10 C.F.R. § 11.9.

[10]The non-participating Board member was not available to attend the oral argument on the safeguards issue.

[11]*See* unpublished "Memorandum (Providing Notice of Granting BREDL Motion for Need to Know Determination and Extension of Deadline for Filing Security-Related Contentions)" (Jan. 29, 2004) (The unredacted version of the order is sealed because it contains safeguards information.) Specifically, the Board ordered the Staff to produce the following items:

1) Three Orders for Modification of License that the NRC issued for Catawba on

(continued...)

contentions, allowing them to be filed no later than 14 days after the Staff makes the disputed safeguards documents available for inspection. On Jan. 30, 2004, the NRC staff requested a temporary stay of the Board's order pending review of a petition for interlocutory review to be filed on the same day. The Commission granted a "housekeeping stay"[12] of the order until February 13, 2004, and later extended the stay to February 18, 2004.

In the meantime, on February 4, 2004, the Board issued an order (unpublished), at the request of BREDL, providing BREDL's attorney and expert access to a closed meeting between the NRC staff and Duke to discuss requests for additional information on Duke's security submittal. In reaction to the Board's February 4 order, the Staff canceled the proposed meeting.

On January 30, 2004, the Staff filed a petition for interlocutory review of the Board's January 29 "disclosure" order. Duke supports and BREDL opposes the Staff's petition. Just last week, on February 11, the Staff also sought review of the Board's February 4 "meeting" order. We grant review and reverse both of the Board's orders.

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[11](...continued)
April 29, 2003, including the revised Design Basis Threat (DBT) for radiological sabotage, the training order and the fatigue order; 2) The access authorization order that the NRC issued for Catawba on January 7, 2003; and 3) Any regulatory guidance associated with these orders.

Id., slip op. at 3, 16-17. (The Board also issued a public version of this order. *See* "Memorandum (Providing Notice of Granting BREDL Motion for Need to Know Determination and Extension of Deadline for Filing Security-Related Contentions)" (Jan. 29, 2004)).

[12]*See, e.g., Yankee Atomic Electric Co.* (Yankee Nuclear Power Station), 1996 WL 627, 640 (NRC) (Oct. 2, 1996).

II. DISCUSSION

A. Interlocutory Review

The Commission's longstanding general policy disfavors interlocutory review.[13] But we do undertake such review when a Board ruling either threatens "immediate and serious irreparable impact" or "affects the basic structure of the proceeding in a pervasive or unusual manner."[14] And sometimes we review interlocutory decisions as an exercise of our inherent supervisory authority over ongoing adjudicatory proceedings.[15]

The NRC Staff filed its petition for interlocutory review of the Board's January 29 disclosure order on the ground that the disputed ruling threatens serious and irreparable impact which could not be alleviated through a petition for review of the Board's final decision.[16] The Staff states that this case warrants interlocutory review to avoid irreparable harm to the public and to the nation's common defense and security because compliance with the Board's order -- unnecessarily and unlawfully -- would provide BREDL access to documents that cover much of the Commission's post-September 11, 2001 work in the area of nuclear security. Further, says the Staff, the requested documents reveal sensitive information that is pertinent to all operating nuclear power plants. Duke agrees with the Staff and adds that release of the documents will

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[13]*See Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), CLI-00-13, 52 NRC 23, 28-29 (2000); *Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant), CLI-00-11, 51 NRC 297, 299 (2000).

[14]10 C.F.R. § 2.786(g). *See, e.g., Duke Cogema Stone & Webster* (Savannah River Mixed Oxide Fuel Fabrication Facility), CLI-02-07, 55 NRC 205, 214 n. 15 (2002); *Georgia Power Co.* (Vogtle Electric Generating Plant, Units 1 and 2), CLI-94-5, 39 NRC 190, 193 (1994).

[15]*See, e.g., Advanced Medical Systems, Inc.* (One Factory Row, Geneva OH 44041), ALAB-929, 31 NRC 271, 279 (1990).

[16]*See* 10 C.F.R. § 2.786(g)(1). This provision applies to certified questions and rulings referred by the presiding officer. Even absent a referral or certification, the Commission will consider a petition for interlocutory review if one of the standards in 10 C.F.R. § 2.786(g) is met.

also have a pervasive effect on the proceeding by opening the door to contentions which will needlessly broaden the proceeding. BREDL opposes the Staff's petition for review.

As disclosure of the safeguards information at issue here would be effectively irreversible later, the Commission agrees that review is necessary now. Review at the end of the case would be meaningless because the Commission cannot later, on appeal from a final Board decision, rectify an erroneous disclosure order.[17] A bell cannot be unrung. "Because the adverse impact of that release would occur now, the alleged harm is immediate."[18] Accordingly, we will review the Board's decision now. The Staff's petition for review, and the parties' briefs in response to it, discuss the issues adequately. No additional briefs are necessary.

As for the Staff's petition for review of the Board's February 4 "meeting" order, we are convinced that the Board lacked authority to issue the order. That order has already had the effect of canceling the meeting to which it was originally addressed. A failure by the Commission to review the February order now could lead to a continuing impact on how the Staff conducts its non-adjudicatory duties related to review of the license amendment that is the subject of this proceeding. Hence, for the reasons we give near the end of this opinion, we exercise our general supervisory authority over adjudications to summarily reverse the February 4 order.

B. "Need to Know"

To obtain access to safeguards information, a person must have an "established need to know"[19] and must provide assurance of trustworthiness. Here, BREDL's attorney and its expert have security clearances beyond the minimum requirement for access to safeguards

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[17] *See Vogtle*, CLI-94-5, 39 NRC at 193.

[18] *Id.*

[19] *See* 10 C.F.R. § 73.21(c).

information. Thus, the NRC staff was not reluctant to give them access to Duke's safeguards-protected security submittal outlining Duke's security proposals for its MOX amendment. Hence, there is no question here of clearances or trustworthiness. The only issue is BREDL's "need to know" the additional safeguards information it seeks.

NRC regulations define "need to know," in the safeguards setting, as a finding that it is *necessary* for a recipient to have the safeguards information to perform official duties (here, to participate in an NRC hearing):

> *Need-to-know* means a determination made by a person having responsibility for protecting Safeguards Information that a proposed recipient's access to Safeguards Information is necessary to the performance of official, contractual, or licensee duties of employment.[20]

Plainly, under this "necessity" definition, "need to know" is a much narrower standard than general relevance. A party's mere desire to have information or its belief that the information is needed to provide context or background may have little or no bearing on a "need-to-know" determination, which must distinguish between "wants" and needs. Also, a party's need to know may be different at different stages of an adjudicatory proceeding, depending on the purpose of the request for information.

In this case, we have examined Duke's security submittal and we find that BREDL does not require access to the additional information it seeks to formulate security contentions. In other words, the mandatory "necessary" element of "need to know" is missing here. This proceeding has a limited scope, focusing on the lawfulness and safety of Duke's proposed MOX amendment. Duke has already provided its security plan for implementing that amendment, including safeguards information. More general security information related to the

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[20] 10 C.F.R. § 73.2. In 10 C.F.R. § 95.5, "need to know" is defined, for purposes of *classified* information, as allowing access when a person "requires" it to perform or assist in a government function. The Board order that we are reviewing indicates that the Board will later have to make a decision about some classified information that BREDL has requested.

Catawba plant-at-large -- the kind of information in the NRC orders that the Board has ordered disclosed to BREDL -- is not, in our judgment, "necessary" to allow BREDL to participate meaningfully in this license amendment proceeding.

The current proceeding has nothing to do with the NRC's post-September 11 general security orders.[21] It is not those orders, but Duke's MOX-related security submittal, that details the particular security measures that will be taken as a consequence of the presence of the MOX fuel assemblies at issue here. Duke's security submittal seeks exemptions from certain requirements of 10 C.F.R. §§ 73.45 and 73.46, and it provides explanations for those exemption proposals. Duke's MOX-related security proposal and its exemption request may be appropriate subjects for BREDL's security contentions. In requesting the amendment at issue in this proceeding, Duke has not asked for an exemption from any of the terms of the orders that are the subject of this dispute. We see no reason why BREDL cannot evaluate Duke's proposed incremental changes to its security plan related to the presence of MOX fuel assemblies and decide whether to challenge Duke's proposed security arrangements as inadequate to accommodate the use of MOX fuel at Catawba.

The Board's need-to-know determination is flawed because it succumbs to BREDL's general argument that it needs more information about the context, or baseline, against which it will measure Duke's security submittal. But a desire to obtain safeguards materials for "context" is an insufficient basis for access to safeguards information. Rather, the touchstone for a demonstration of "need to know" is whether the information is indispensable. Here, as the pleadings before us represent, neither Duke nor the NRC staff has any intention of measuring Duke's security arrangements for MOX against last year's general security orders issued to

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[21] *Cf. Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation, LBP-03-05, 57 NRC 233 (2003) (license applications are measured against regulatory standards, not against enforcement orders).

reactors. Indeed, those orders do not impose immutable requirements, but are subject to change depending on updated assessments of the terrorist threat. All parties to this adjudication, including BREDL, may safely assume, as a baseline, that Duke's Catawba facility will comply with all applicable general security requirements, both those prescribed in NRC rules and those prescribed by NRC order. That's not at issue in this MOX license amendment case. At stake here is the appropriate increment -- the appropriate heightening of security measures -- necessitated by the proposed presence of MOX fuel assemblies at the Catawba reactor site. While these security enhancements are safeguards information, BREDL has been given access to that information and thus is in a position to measure Duke's security proposals against the requirements of Part 73. After doing so, BREDL (or its technical expert) should be able to identify credible vulnerabilities, if any, and present corresponding contentions to the Board.

As a policy matter, the Commission has a strong interest in limiting access to safeguards and security information. We must limit distribution of safeguards information to those having an actual and specific, rather than a perceived, need to know. Anything less would breach our duty to the public and to the nation, for the likelihood of inadvertent security breaches increases proportionally to the number of persons who possess security information, regardless of security clearances and everyone's best efforts to comply with safeguards requirements.[22] The Commission is well aware of the delicate balance between fulfilling our mission to protect the public and providing the public enough information to help us discharge that mission. In this case, however, we find BREDL's lack of "need to know," within the meaning of our regulations, determinative. Thus, we grant the Staff's petition for review and reverse the Board's January 29 order directing disclosure of safeguards documents.

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[22] *See Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), CLI-02-25, 56 NRC 340, 356 (2002).

C. Board Authority over Staff Meetings

The Board's effort, in its February 4 order, to grant BREDL a ticket of admission to a closed meeting between the NRC staff and Duke was inappropriate. Our licensing boards have wide powers over adjudications, as for example, when they determine who can participate in hearings, where and when such hearings should take place, and which issues are litigable. But NRC staff reviews, which frequently proceed in parallel to adjudicatory proceedings, fall under the direction of Staff management and the Commission itself, not licensing boards. If, as the Board here apparently believed, the NRC staff unreasonably has closed a meeting, or has acted in violation of Commission open meeting policies, that is a matter to be addressed through normal agency channels, outside the adjudication. We long have held that licensing boards do not sit to correct NRC staff misdeeds or to supervise or direct NRC staff regulatory reviews.[23]

The licensing boards' sole, but very important, job is to consider safety, environmental, or legal issues raised by license applications. Licensing boards simply have no jurisdiction over non-adjudicatory activities of the Staff that the Commission has clearly assigned to other offices unless the Commission itself grants that jurisdiction to Board. In this case the Commission has made no extraordinary grant of authority to the Board beyond the routine authority to oversee the adjudicatory aspects of Duke's amendment application. Accordingly, we summarily reverse the Board's February 4 order opening to outsiders a confidential meeting that the NRC staff decided to close in order to protect safeguards information.

D. General Guidance

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[23] *See e.g., Baltimore Gas & Elec. Co.* (Calvert Cliffs Nuclear Power Plant, Units 1 and 2), CLI-98-25, 48 NRC 325, 349 (1998); *Curators of the University of Missouri*, CLI-95-1, 41 NRC 71, 121 (1995).

The Commission notes that the number and types of security issues that intervenors or hearing petitioners have raised continue to increase in the wake of the September 11, 2001 terrorist attacks. This is understandable, and we by no means wish to discourage citizens or groups from raising their security concerns before NRC licensing boards or before the Commission itself. But there is a potential for licensing boards to reach inconsistent conclusions regarding the need for dissemination of safeguards information. Accordingly, to promote both uniformity of decisions and fairness to litigants as well as to protect information, we take this opportunity to offer guidance to licensing boards (and presiding officers) in their "need to know" determinations.

First, as is evident from the text of our regulations,[24] it is appropriate for NRC Staff experts to make the initial "need to know" decisions. When a licensee or intervenor disputes those decisions, licensing boards, while exercising their own judgment, should give considerable deference to the Staff's judgments. The Commission has confidence in our Staff, which is well trained and is experienced in NRC licensing and enforcement proceedings, and intimately familiar with both NRC safeguards regulations and the licensing or enforcement matter at hand.

Second, if a licensing board does overturn a Staff need-to-know finding, it is imperative that access to safeguards documents be as narrow as possible. Again, we rely on the text of our regulations, which says that the disclosure must be "necessary" or "required." [25] This standard entails thorough examination of safeguards materials and, at times, release of only

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[24] *See* 10 C.F.R. §§ 73.2, 73.21, 95.5.

[25] *See* 10 C.F.R. §§ 73.2, 73.21, 95.5.

portions of documents or redacted versions of documents; *i.e.*, a "sanitized" version of a document.[26]

Finally, it is important in some cases (although not this one) that the Board assure itself of the reliability of the requestor of the safeguards information.[27] This ordinarily requires special procedures for attorneys and experts. Boards therefore should restrict access to qualified, "cleared" representatives of intervenors.[28] And boards should stress that such representatives may handle the confidential information only under the conditions and restrictions laid out in NRC regulations.

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[26]*See, Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-410, 5 NRC 1398, 1405, *review denied,* CLI-77-23, 6 NRC 455 (1977). Facially, the Board's disclosure order in the instant case appears overbroad, even if it were otherwise justifiable, because it directs that BREDL's representatives be given access to entire documents.

[27]BREDL has requested access to the security information only for its attorney and its technical advisor, both of whom hold "L"-level security clearances.

[28]*Id.* at 1406.

III. CONCLUSION

For the foregoing reasons, the Commission: (1) *reverses* the Board's January 29 order granting BREDL's motion for "need to know" determination and the Board's February 4 order granting BREDL access to a closed meeting; (2) *directs* BREDL to file any security-related contentions no later than March 3, 2004; and (3) *provides* that responses to any such contentions shall be filed no later than 14 days after the contentions are filed.[29]

IT IS SO ORDERED.

For the Commission

/RA/

Annette L. Vietti-Cook
Secretary of the Commission

Dated at Rockville, Maryland,
this 18th day of February, 2004

[29]The NRC Staff also requested a stay pending disposition of its petition for review. Because of time constraints, we granted an emergency housekeeping stay to enable our review of the Staff's motion and its petition for review. In view of our decision to grant the petition for review, and to reverse the Board, we need take no further action on the Staff's stay motion.

From: John Larkins
To: Powers, Dana A
Date: 2/25/04 5:42PM
Subject: Re: Planned Subcommittee Meeting and Committee Meeting on LTAs

Thanks, I will put this item on the P&P for next week, but the Committee should not involve themselves in policy matters that the Commission has taken a position on.
>>> "Powers, Dana A" <dapower@sandia.gov> 02/25/04 02:29PM >>>

John,Ralph,

We will in the near future have some meetings on the LTAs. The Commission has recently over-ruled the ALSRB on the issue of providing intervenors with access to security information concerning the storage of the LTAs. This has not set well with the intervenors. I would not be at all surprised that in our solicitation for public comments concerning LTAs that this issue will be raised with the expectation that ACRS would want to enter this fray. As far as I can see, it is strictly one of policy and the Commission has made the policy. ACRS lacks any expertise to critique or endorse this policy. Consequently, I think we ought to chat at our next opportunity about how the designated federal official and the chair of the meeting ought to react to attempts by the public to raise the issue. Mario may want to inform the Commission of the potential for the public to raise the issue at our future meetings and assure the Commission that ACRS (or certainly this member of the ACRS) does not want to get involved in the issue.

Dana

CC: RXC; SXD1