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

SL-0522

July 6, 2004

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

Dear Chairman Diaz:

SUBJECT: SUMMARY REPORT - 513th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, JUNE 2-4, 2004, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 513TH meeting, June 2-4, 2004, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports, letter, and memoranda:

REPORTS:

Reports to Nils J. Diaz, Chairman, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Security of Nuclear Facilities, dated June 10, 2004 (National Security Information - Secret)

- Draft Final 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components For Nuclear Power Reactors," dated June 15, 2004

LETTER:

Letter to Luis A. Reyes, Executive Director for Operations, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Digital Instrumentation and Controls Research Program, dated June 9, 2004

MEMORANDA:

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

- Proposed Revisions to Standard Review Plan Sections 5.2.3, 5.3.1 and 5.3.3, dated June 9, 2004

- Draft Regulatory Guide, DG-1130, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants," (Proposed Revision 2 to Regulatory

Guide 1.152), dated June 3, 2004

Memorandum to Bruce A. Boger, Director, Division of Inspection Program Management, NRR, from John T. Larkins, Executive Director, ACRS:

- Deferral of ACRS Review of Draft SRP Chapter 13.0 "Conduct of Operation," Section 13.1.2 - 13.1.3, "Operating Organization" Revision and Supporting Documents Until After Public Comment, dated June 7, 2004

HIGHLIGHTS OF KEY ISSUES

1. Draft Final 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components For Nuclear Power Reactors"

The Committee met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss the draft final 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors." The staff's briefing focused on major changes to the rulemaking package since its briefing to the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment on February 19, 2004. The staff also discussed the conditions in Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," on the acceptance of NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline." NEI then provided its perspective on the draft final rule. NEI expressed some frustration because it had just been made aware of some changes to the rulemaking package, and it disagreed with the staff's proposal to issue Regulatory Guide 1.201 for trial use.

Committee Action:

The Committee issued a report to the Chairman on this matter, dated June 15, 2004, recommending that the final 10 CFR 50.69 be issued. The Committee concluded that Regulatory Guide 1.201 should be issued for trial use.

2. Revised License Renewal Review Process

The Committee heard presentations by and held discussion with representatives of the NRC staff regarding the proposed enhancements to the current license renewal application review process that are expected to increase the overall efficiency of the process while maintaining an appropriate level of rigor to ensure adequate plant safety during the period of extended operation. The proposed enhancements include additional onsite reviews to improve the flow of information and limit the need for written correspondence on minor issues and clarifications.

Before finalizing these proposed enhancements, the staff will evaluate the pilot applications of these enhancements at four plants. Insights gained during these pilot applications will be considered before the staff makes enhancements to the Generic License Renewal Guidance Documents.

Committee Action:

This was an information briefing, and no Committee action was taken.

3. Meeting With the NRC Commissioners

The Committee met with the NRC Commissioners on June 2, 2004 to discuss the following items:

- PWR Sump Performance
- PRA Quality for Decisionmaking
- Risk-Informing 10 CFR 50.46
- ACRS 2004 Report on the NRC Safety Research Program
- ESBWR Pre-Application Review
- Interim Review of the AP1000 Design

The Committee is awaiting the Staff Requirements Memorandum (SRM) from the Commission regarding this meeting.

Committee Action:

The Committee will address any issues raised by the Commission in the SRM.

4. Digital Instrumentation and Control System (I&C) Research Activities

The Committee heard presentation by and held discussion with the representatives of the Office of Nuclear Regulatory Research (RES) regarding the current research activities for developing tools and guidance for the risk-assessment of digital I&C systems. The staff provided an overview of the information presented to the Plant Operations Subcommittee on March 26, 2004. The staff described the research program objectives included in SECY-01-0155, NRC Digital Instrumentation and Control Research Plan, and focused attention on the current research studies under way to address each of these objectives.

The staff provided additional details of the ongoing developmental studies at the University of Virginia, the University of Maryland, and Brookhaven National Laboratory that address the following issues:

- Digital Systems have different failure modes, and are much more challenging to model. More quantitative methods are needed.

- Digital systems are being retrofitted into current generation of nuclear power plants and they need to be reviewed in a risk-informed manner.

- The NRC does not have guidance on what is acceptable and what is not in modeling of digital system reliability.

Committee Action:

The Committee issued a letter to the NRC Executive Director for Operations on this matter, dated June 9, 2004. The Committee expressed support for the effort of the Digital I&C Research Program to develop more quantitative measures of digital system reliability. The letter also contained personal opinions and recommendations submitted by Dr. George

Apostolakis for staff consideration.

5. NRC Staff's Response to the ACRS Report on the AP1000 Design

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding the staff's response to the ACRS interim letter dated March 17, 2004 on the AP1000 design certification review. The ACRS in its letter commented on seven technical issues. These issues are automatic depressurization system (ADS)-4 squib valve function, assurance of long-term cooling (strainer blockage), code deficiencies, range of pi-group values, in-vessel retention/fuel-coolant interactions (FCI), organic iodine production, and catastrophic failure of a free-standing steel containment.

In addition, Westinghouse representatives briefed the Committee on three issues that had been raised by the ACRS: in-vessel retention/FCI, organic iodine production, and catastrophic failure of a free -standing steel containment. The staff described its plans to review Westinghouse's organic iodine production sensitivity study.

Committee Action:

This briefing was for information only. The ACRS Future Plant Designs Subcommittee plans to hold a meeting on June 25, 2004 to follow-up on the organic iodine production issue and any remaining items as a result of the review of the Final Safety Evaluation Report (FSER). The Committee also plans to include a session during its meeting on July 7-9, 2004 to discuss the FSER.

6. Proposed Revisions to Standard Review Plan (SRP) Sections and Process and Schedule for Revising the SRP

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding proposed revisions to SRP Sections 5.2.3, "Reactor Coolant Pressure Boundary Materials;" 5.3.1, "Reactor Vessel Materials;" and 5.3.3, "Reactor Vessel Integrity." The staff stated that the proposed revisions to the above SRP sections include primarily editorial changes and updates to the references, and do not include any technical changes. The staff also discussed the process, schedule, estimated resources for updating the other SRP sections. Those SRP sections that involve safety-significant issues and stakeholder/Commission interest will have a high priority for update. Updates to the SRP will be accomplished in accordance with the Office of Nuclear Reactor Regulation (NRR) Office Instruction (OI) LIC-200, "Standard Review Plan Process." The staff plans to update about 35 SRP sections in each FY 2005 and FY 2006. The staff will "bundle" the relevant SRP Sections and submit the bundles to the ACRS for review. During FY 2005, the staff plans to submit 13 "bundles" to the ACRS for review.

Committee Action:

The Committee decided not to review the proposed revisions to SRP Sections 5.2.3, 5.3.1, and 5.3.3. The ACRS Executive Director issued a memorandum to the NRC Executive Director for Operations (EDO), dated June 9, 2004, informing the EDO of the Committee's decision. As suggested by the Committee, the staff has agreed to include a recommendation in the memorandum transmitting future SRP revisions to the ACRS with regard to the need for the

Committee's review of those sections along with the reasons therefor.

7. Metrics for Evaluating the Quality of the NRC Research Programs

RES is required to have an independent evaluation of the quality of its research programs. This evaluation is mandated by the Government Performance and Results Act and needs to be in place during the next fiscal year. The Committee has agreed to assist RES in assessing the quality of the NRC research programs. The Committee has previously approved the strategy for the review of the quality of selected research projects. This strategy will be tested during FY 2004 and refined in FY 2005. During the June 2-4, 2004 ACRS meeting, the Committee discussed a process for developing the quantitative metrics (numerical grades) to be used for evaluating the quality of selected NRC research projects.

Committee Action:

The Committee plans to use a decisionmaking framework (value tree) in evaluating the quality of selected NRC research projects. In FY 2004, the Committee will assess the quality of the NRC research projects on PWR Sump Performance and on MACCS Code.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENT

- The Committee discussed the EDO's response dated May 18, 2004 to comments and recommendations included in the ACRS interim letter dated March 17, 2004, regarding the ACRS Reviews of the Westinghouse Electric Company Application for Certification of the AP1000 Plant Design.

  The Committee decided that it was satisfied with the EDO's response. However, the ACRS Subcommittee on Future Plant Designs plans to discuss the organic iodine production issue during a meeting on June 25, 2004.

- The EDO's May 18, 2004 response concerning the AP1000 also included several commitments that arose from the AP1000 review, but which are generic in applicability, and their resolution does not impact the completion of the AP1000 licensing activity.

  The staff committed to brief the ACRS Subcommittee on Thermal-Hydraulic Phenomena on experimental studies being conducted at the ATLATS facility at Oregon State University to support thermal-hydraulic code improvements, and the planned separate-effects tests at the APEX-AP1000 facility, which will help to isolate and measure upper plenum pool entrainment. The staff will brief the Subcommittee on the new models that will be developed and implemented in the TRACE code as deficiencies are identified.

  The Committee plans to review the procedure that the staff will develop to define an appropriate Pi group range for scaling integral test facilities.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from May 6, 2004 through June 1, 2004, the following Subcommittee meetings were held:

- Planning and Procedures - June 1, 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.

- Materials and Metallurgy - June 1, 2004

  The purpose of this meeting was to follow up on the July 2003 meeting on vessel head degradation regarding work that the MRP and NRC staff indicated they would be working on.

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The Committee plans to review insights gained from the trial use of Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance."

- The Committee plans to review the draft final Regulatory Guide, DG-1130, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants," after reconciliation of public comments.

- The Committee plans to review the draft final SRP Chapter 13.0, "Conduct of Operation," after reconciliation of public comments.

- The Committee plans to hold discussions with the staff regarding Safeguards and Security matters during its September 8-11, 2004 meeting.

PROPOSED SCHEDULE FOR THE 514TH ACRS MEETING

The Committee agreed to consider the following topics during the 514th ACRS meeting to be held on July 7-9, 2004:

- Final Safety Evaluation Report Associated with the AP1000 Design Certification
- Draft Final Generic Letter on Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at PWRs
- Risk-Informing 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors"
- Differences in Regulatory Approaches and Requirements Between U.S. and Other Countries

- Proposed Generic Communication on the Use of Ultrasonic Flow Measurement Devices for Measuring Feedwater Flow Rates in Nuclear Plants
- Status of the ACRS Members' Assessment of the Quality of Selected NRC Research Projects

Sincerely,

/RA/

Mario V. Bonaca
Chairman

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OFFICE	ACRS	ACRS	ACRS	ACRS	ACRS
NAME	HJLarson	SDuraiswamy	RCaruso	JTLarkins	JTLarkins for MVB
DATE	6/23/04	6/23/04	7/1/04	7/1/04	7/6/04

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June 2-4, 2004

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LETTER:

Letter to Luis A. Reyes, Executive Director for Operations, NRC, from Mario V. Bonaca, Chairman, ACRS:

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APPENDICES

I. *Federal Register Notice*
II. Meeting Schedule and Outline
III. Attendees
IV. Future Agenda and Subcommittee Activities
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513th ACRS Meeting
June 2-4, 2004

CERTIFIED

MINUTES OF THE 513th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
June 2-4, 2004
ROCKVILLE, MARYLAND

The 513th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on June 2-4, 2004. Notice of this meeting was published in the *Federal Register* on May 19, 2004 (65 FR 28951) (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, Dr. Dana A. Powers, Dr. Victor H. Ransom, Dr. William J. Shack, and Mr. John D. Sieber. Mr. Graham M. Leitch did not attend this meeting. 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. Draft Final 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors" (Open)

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

Dr. George Apostolakis, the cognizant Committee member for this issue, introduced the topic at approximately 8:35 am. Dr. Apostolakis said the purpose of the briefing was to review the draft final rulemaking package for 10 CFR 50.69. He said 10 CFR 50.69 had been developed to permit licensees to implement an alternative regulatory framework with respect to special treatment. "Special treatment" refers to those requirements that provide increased assurance beyond normal industrial practice that structures, systems, and components perform their design basis functions. Dr. Apostolakis reminded the Committtee of its most recent action on this matter was to review and comment upon draft rule language for 10 CFR 50.69 and proposed industry guidance in Revision B to NEI 00-04. The Committee's letter dated March 19, 2002, had several conclusions and recommendations, which included the following: (1) criteria used by the integrated decisionmaking panel for categorizing SSCs should be made explicit and should include consideration of risk metrics that supplement core damage frequency and large early release frequency, such as late containment failure and inadvertent release of radioactive material, (2) aging phenomena and the management of degradation must be considered in the IDP deliberations concerning affected SSCs and passive system components, (3) guidance for performing uncertainty analyses should be provided, and (4) justification for increasing failure rates in NEI 00-04 Revision B was weak and better justification was required.

NRC Staff Presentation

Tim Reed, Office of Nuclear Reactor Regulation (NRR), the main presenter introduced the subject, along with introducing Donald Harrison and Tom Scarbrough, NRR. Mr. Reed said the objective of the briefing was to gain the Committee's endorsement on the draft final rule. The discussions focused on changes to the draft final rule since the February 2004 Subcommittee briefing and remaining issues associated with RG 1.201.

Mr. Harrison then discussed the associated guidance contained in RG 1.201. He said RG 1.201 conditionally endorses NEI 00-04. Mr. Harrison said that implementation limitations depended on the types of analyses used in categorization. These implementation limitations were also dependent on the availability of PRA standards and guidance as discussed in the staff's plan for a phased approach to PRA quality. He then discussed uncertainty considerations and common cause failure and degradation mechanism considerations. Programs designed to prevent degradation are to be carried through and monitoring programs are to be in place to support the assumed reliability numbers. One technical objection remained. The staff did not agree with the RISC-3 reliability reduction factor assumed as part of the risk sensitivity recommended in NEI 00-04.

Mr. Harrison then discussed IDP membership. This issue was raised during the February 2004 Subcommittee meeting. The rule requires that the IDP be staffed with expert, plant-knowledgeable members. NEI 00-04 Section 9.1 provides additional information on panel makeup and training. Mr. Harrison said that at the ASME/ANS joint meeting IDP qualifications were identified as an action item for future consideration. The next topic was the rationale for issuing RG 1.201 for trial issue. RG 1.201 was being issued for trial use because of the reliability reduction factor issue, Code Case N-660 guidance had not been finalized, and staff expects to learn from piloting RG 1.201. Mr. Harrison concluded by saying that NEI is addressing staff comments and that the staff will continue to work NEI to address staff comments and develop the final version of NEI 00-04 that is endorseable with few, if any, conditions.

During the above discussions, the NRC staff, a representative of NEI, and the ACRS Members made the following points:

- Dr. Apostolakis asked if RG 1.201 being issued for trial use meant that it would be revised in a year. The NRC staff indicated that was the case, that lessons learned from the Surry and Wolf Creek pilot applications would be incorporated into RG 1.201. Dr. Apostolakis followed up by asking if any changes to the rule were anticipated as a result of the pilot applications. The NRC staff said that none were anticipated.

- Dr. Apostolakis asked if the NRC staff was using existing uncertainty studies such as NUREG-1150 to identify major contributors. Mr. Harrison said he thought so but that he wanted to consider more recent work.

- Dr. Apostolakis asked what assures us that the process is conservative. Mr. Harrison said there are several checks and balances. Sensitivity studies are required. Second, delta risk calculations are performed to make sure that any risk increase is small. Third, there is a corrective action feedback loop to verify the failure rates.

- Dr. Shack asked if masking an importance factor with the conservatisms was being considered. Mr. Harrison said the objective was to try and get at what is driving the answer high or low. Those factors are then adjusted to see what effect it has. If there is a component and an adjustment and it does not move, that is a confirmation. If it does move than the IDP needs to consider it.

- Mr. Rosen asked what the scope of the 50.69 evaluation was. Mr. Harrison said it was in a process review for a couple of systems. Mr. Harrison continued that once the rule goes out, it would be a process approval. The review will demonstrate how the process works for the systems chosen. Mr. Rosen then confirmed that the licensee would still have to comply with the rule when the rule is issued. Mr. Harrison confirmed that the licensee would have to submit a license amendment.

- Dr. Ford acknowledged that the rule addresses common cause failure and known degradation mechanisms. Dr. Ford said that he could not find guidance in NEI 00-04 on how to address this requirement. Mr. Harrison said the staff's position was that to maintain those programs that address known degradation mechanisms, the programs will pass through unaffected.

- Mr. Pietrangelo, NEI, cautioned the Committee on allowing the same rule language in the treatment requirements that the implementer has just been exempted from.

- Mr. Pietrangelo said the issue the staff had was not with the reliabiliy reduction factor and how it's established. The issue is how you monitor against that factor. That's the remaining technical issue. Mr. Pietrangelo said more discussion is needed. Based on a single failure, on a low safety-significant SSC, the rule is going to require a corrective action, as well as condition adverse to quality which, Mr. Pietrangelo stated, means an extent of condition evaluation on the failure. He said the staff is requesting that one immediately changed treatment and test everything else upon failure of one low safety-significant SSC. He said that this was an over reaction and that industry would not do that.

<u>Committee Action:</u>

The Committee issued a report to the Chairman on this matter dated June 15, 2004. The Committee recommended that the final 10 CFR 50.69 be issued. The Committee concluded that Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," should be issued for trial use.

III. <u>Revised License Renewal Review Process</u> (Open)

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

The staff provided an informational briefing outlining the proposed enhancements to the current license renewal application review process that are expected to increase the overall efficiency of the process while maintaining an appropriate level of rigor to ensure adequate plant safety during the period of extended operation. The proposed enhancements to the process include additional onsite reviews to improve the flow of information and limit the need for written correspondence on minor issues and clarifications.

Before finalizing these proposed changes, the staff will complete pilot evaluations at four plants. Insights gained during these reviews will be considered before the staff makes final enhancements to the license renewal review guidance documents.

<u>Committee Action</u>

This was an information briefing only, no Committee action was required.

IV. Digital Instrumentation and Control System Research Activities (Open)

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

The Office of Nuclear Regulatory Research (RES) provided information on current research activities for developing tools and staff guidance for the risk assessment of digital I&C systems. The staff provided an overview of the information presented to the Plant Operations Subcommittee on March 26, 2004. The staff presentation described the research program objectives described in SECY-01-0155, NRC Digital Instrumentation and Control Research Plan, and focused attention on the current research studies underway to address each of these objectives.

The staff provided additional detail of the ongoing developmental studies at the University of Virginia, the University of Maryland, and Brookhaven National Laboratory that address the following issues:

- Digital Systems have different failure modes, and are much more challenging to model. More quantitative methods are needed.
- Digital systems are being retrofitted into current generation of nuclear power plants and they need to be reviewed in a risk-informed manner.
- The NRC does not have guidance on what is acceptable and what is not in modeling of digital system reliability.

Committee Action

In a letter dated June 9, 2004, the Committee supported the effort of the Digital I&C Research Program to develop more quantitative measures of digital system reliability. The letter also contained personal opinions and recommendations submitted by Dr. George Apostolakis for staff consideration.

V. NRC Staff Response to the ACRS Report on the AP1000 Design (Open)

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

Dr. Thomas Kress, Future Plant Designs Subcommittee Chairman, stated that the purpose of this meeting was to hold discussions with the NRC staff and Westinghouse representatives regarding their response to the ACRS comments and recommendations included in the March 17, 2004 ACRS interim letter on the AP1000 design.

Mr. John Segala, Project Manager, Office of NRC Nuclear Reactor Regulation (NRR), stated that 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 CFR, and for standard design certification in accordance with Subpart B of 10 CFR Part 52. Accordingly, the NRC staff has reviewed the design certification application and will issue its draft safety evaluation report (DSER) on June 16, 2003.

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.

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.

During the 510th meeting of the ACRS, the Committee met with the NRC staff and Westinghouse representatives and discussed the status of the open items as well as issues previously raised by the ACRS. The Committee reviews have not addressed security matters and their impacts on the AP1000 design.

On March 17, 2004, the Committee issued a letter to the Executive Director for Operations. The Committee outlined seven issues in which the ACRS had comments related to the certification of the AP1000 design. These seven issues are:

Issue 1- Automatic Depressurization System (ADS)-4 Squib Valve Function.
Issue 2- Assurance of Long-Term Cooling (Strainer Blockage).
Issue 3- Code Deficiencies.
Issue 4- Range of Pi-Group Values.
Issue 5- In-Vessel Retention/Fuel-Coolant Interactions (FCI).
Issue 6- Organic Iodine Production.
Issue 7- Catastrophic Failure of a free-Standing Steel Containment.

On May 18, 2004, the NRC staff responded to the above 7 issues.

Issue 1. The staff indicated that the AP1000 design control document (DCD) will include an ITAAC that ensures the ADS-4 squib valves will perform their function. Tests or type tests will be performed to demonstrate the capability of the ADS-4 squib valves to operate under their design conditions. The staff concludes that the performance characteristics of the ADS-4 squib valves will be adequately verified.

Issue 2. The staff stated that Westinghouse had revised the COL action item in the DCD to include the evaluation of chemical debris. This COL action item will capture any impact of chemical effects on the ability of the affected components to accommodate anticipated debris loadings identified during the resolution of GSI-191. The staff concludes that Westinghouse has resolved the concerns related to additional debris that can be caused by chemical reactions in the containment.

Issue 3. The staff indicated that for the AP1000, an acceptable solution was to conservatively bound the calculations. The TRACE code was not used in the AP1000 review, but is currently being assessed using APEX-AP1000 test data. Currently, there are two ongoing experimental studies to help correct code deficiencies. New thermal-hydraulic models will be developed and implemented into TRACE as deficiencies are identified. The experimental programs, along with the associated TRACE assessment will be described to the ACRS thermal-hydraulic Subcommittee at a future meeting.

Issue 4. The staff indicated that an acceptance range of 0.5 to 2.0 for various Pi-groups was selected. The staff notes that the appropriate Pi-groups is generic and does not represent an issue that is specific only to AP1000 designs. The staff, however, as a long term effort will work to develop and document a procedure to define an appropriate Pi-group range for scaling integral test facilities.

<u>Materials</u>. The ACRS commented that ongoing and future studies may suggest material and environmental changes that will be addressed at the COL stage. The staff notes that any future and environmental changes that would be requested at the COL stage would have to follow the change processes set forth in paragraph VIII.C of the DCD.

<u>Severe Accidents</u>. The ACRS commented and questioned the technical justification for the aerosol removal coefficient (lambda) for containment. The staff performed an independent dose analysis with the median aerosol removal coefficient values from the staff's uncertainty analysis , along with other analysis parameters and the bounding hypothetical atmospheric dispersion factors provided by Westinghouse, and the results are within the dose criteria of 10 CFR 50.34 and General Design Criteria 19. The staff concludes that , while the staff and Westinghouse diverge on values for the intermediate steps in the dose calculations, the overall conclusion that the AP1000 dose results are acceptable.

Issue 5. The staff notes that it has performed a reasonably large number of sensitivity analysis and found that the ex-vessel FCI for the AP1000 is of no greater concern than that for the AP600 design. The ex-vessel FCI analysis for the AP600 indicated that the containment integrity would not be challenged by the FCI load. The staff plans to provide additional information.

Issue 6. Westinghouse plans to provide the staff with additional information regarding the pH of the water film on the inside of the containment wall where acidification could produce organic iodine. The staff plans to provide such information to the ACRS along with the staff's evaluation.

Issue 7. The staff states that even if an event progressed to an intermediate or late release, it would likely involve a vented release rather than a catastrophic containment failure, since the AP1000 design will include the capability to vent the containment. The non-safety related containment spray function could also be effective in reducing re-suspended fission products following containment failure.

Westinghouse representatives briefed the Committee on three of the ACRS's issues, namely, in-vessel retention/FCI, organic iodine production, and catastrophic failure of a free-standing steel containment. On April 30, 2004, Westinghouse provided the NRC staff with supporting information regarding the responses to the ACRS issues 5 through 7 above. On May 11, 2004, Westinghouse provided revised information to the NRC staff regarding Issue 6, "Organic Iodine Production." The staff indicated its plans to review Westinghouse's organic iodine production sensitivity study.

Committee Action

This briefing was for information only. The ACRS Future Plant Designs Subcommittee plans to hold a meeting on June 25, 2004 to follow-up on the organic iodine production issue and any remaining items as a result of the review of the final safety evaluation report (FSER). The Committee also plans to hold a session during its meeting on July 7-9, 2004 to discuss the

VI. Proposed Revisions to Standard Review Plan (SRP) Sections and Process and Schedule for Revising the SRP (Open)

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

Dr. Ford, cognizant Subcommittee Chairman, provided a preamble, stating that during this session the NRC staff will brief the Committee with regard to the proposed changes to SRP Sections 5.2.3, "Reactor Coolant Pressure Boundary Materials," 5.3.1, "Reactor Vessel Materials," and 5.3.3, "Reactor Vessel Integrity." Since there are no technical changes involved in the proposed revisions to these SRP Sections, the staff recommends that the ACRS not review these proposed revisions. In addition, the staff will brief the Committee the process and plan to update SRP Sections in FY 2005 and FY 2006.

Proposed Changes to SRP Sections 5.2.3, 5.3.1, and 5.3.3 - Mr. Robert Kuntz, NRR

Mr. Kuntz, NRR, provided a summary of proposed changes to SRP Section 5.2.3, 5,3,1, and 5.3.3. He stated that the proposed changes to these sections do not involve any technical changes. The technology for light water reactor applications in the areas covered by these sections have remained essentially unchanged. Since there are no technical changes involved, the staff believes that an ACRS review of the proposed revisions to the above SRP Sections are not needed.

Dr. Ford stated that the current SRP Sections 5.2.3, 5.3.1, and 5.3.3 were written several years ago. Since then, there have been problems with the nickel-based alloy materials used in PWRs and BWRs. He asked why adequate guidance is not provided for use by the staff in reviewing the nickel-based alloy materials. Mr. Wickman, NRR, stated that there is an SRP section that provides a cautionary note about the use of nickel-based alloys.

In response to another question from Dr. Ford, Mr. Wickman stated that the SRP does not create new regulatory requirements. New requirements are created by modifications to regulations. The purpose of the SRP is to provide guidance to the staff for reviewing license applications.

Stating that there have been accepted regulatory positions on stress corrosion cracking. Dr. Shack stated that the staff should reference such positions in the SRP. Mr. Wickman replied that an accepted regulatory position could be referenced in the SRP. Mr. Matthews added that only those regulatory requirements that have been approved after going through the normal NRC process will be referenced in the SRP.

After further discussions, the staff agreed to meet with Dr. Ford, if needed, to discuss the details of the changes made to SRP Sections 5.2.3, 5.3.1, and 5.3.3 as well as the primary objectives of the SRP.

Process for Revising the SRP - Mr. Robert Kuntz, NRR

Mr. Kuntz, discussed briefly the process for revising the SRP. Key points made by Mr. Kuntz include the following:

- In an October 31, 2003 Staff Requirements Memorandum (SRM), the Commission asked the staff to provide the status, approach, and plans for maintaining a current and effective set of guidance documents (including SRP).
- Prior to the SRM, NRR had begun preliminary work on an SRP update plan that included scoping process, prioritization process, and scheduling.
- The objective of the scoping process is to determine the extent of the update and estimate the resources required to complete the update.
- The purpose of the prioritization process is to create a prioritized list of SRP sections that should be updated each fiscal year. Prioritization will be performed using three criteria: safety significance, recent industry activities, and stakeholders/Commission interest.
- Updating the entire SRP will require approximately 35 FTE.

Several members commented that the update to the SRP should be a continuous process. As soon as new regulatory requirements are established, the SRP should be updated to reflect such requirements. If the SRP is updated once every few years, the staff reviewers will not have the benefit of the new regulatory requirements. Mr. Kuntz stated that the staff is attempting to address this issue in the NRR Office Instruction, LIC-200, "Standard Review Plan Process."

Plan for Updating SRP Sections - Miss Aida Rivera, NRR

Miss Rivera, discussed the plan for updating the SRP Sections. Key points made by Miss. Rivera include the following:

- Updates to the SRP will be performed in accordance with the NRR Office Instruction LIC-200.
- NRR plans to update about 35 SRP Sections in FY 2005 and FY 2006.
- Relevant SRP Sections will be "bundled" prior to submitting to the ACRS for review.
- During FY 2005 the staff plans to submit 13 bundles of SRP updates for ACRS review.

Dr. Powers suggested that when submitting updated SRP Sections for ACRS review, the staff make a recommendation as to whether there is a need for ACRS review of these Sections along with the reasons therefor. Mr. Matthews, NRR, agreed to do so.

Committee Action

The Committee decided not to review the proposed revisions to SRP Section 5.2.3, 5.3.1, and 5.3.3. The ACRS Executive Director issued a memorandum dated June 9, 2004, informing the NRC Executive Director for Operations of the Committee's decision.

VII. Metrics for Evaluating the Quality of the NRC Research Programs (Open)

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

RES is required to have an independent evaluation of the quality of its research programs. This evaluation is mandated by the Government Performance and Results Act and needs to be in place during the next fiscal year. The Committee has agreed to assist RES in assessing the quality of the NRC research programs. The Committee has previously approved the strategy for the review of the quality of selected research projects. This strategy will be tested during FY 2004 and refined in FY 2005. During the June 2-4, 2004 ACRS meeting, the Committee discussed a process for developing the quantitative metrics (numerical grades) to be used for evaluating the quality of selected NRC research projects.

Committee Action:

The Committee plans to use a decisionmaking framework (value tree) in evaluating the quality of selected NRC research projects. In FY 2004, the Committee will assess the quality of the NRC research projects on PWR Sump Performance and on MACCS Code.

VIII. Executive Session (Open)

A. Reconciliation of ACRS Comments and Recommendations/EDO Commitments

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

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

- The Committee discussed the EDO's response dated May 18, 2004 to comments and recommendations included in the ACRS interim letter dated March 17, 2004, regarding the ACRS Reviews of the Westinghouse Electric Company Application for Certification of the AP1000 Plant Design.

  The Committee decided that it was satisfied with the EDO's response. However, the ACRS Subcommittee on Future Plant Designs plans to discuss the organic iodine production issue during a meeting on June 25, 2004.

- The EDO's May 18, 2004 response concerning the AP1000 also included several commitments that arose from the AP1000 review, but which are generic in applicability, and their resolution does not impact the completion of the AP1000 licensing activity.

  The staff committed to brief the ACRS Subcommittee on Thermal-Hydraulic Phenomena on experimental studies being conducted at the ATLATS facility at Oregon State University to support thermal-hydraulic code improvements, and the planned separate-effects tests at the APEX-AP1000 facility, which will help to isolate and measure upper plenum pool entrainment. The staff will brief the Subcommittee on the new models that will be developed and implemented in the TRACE code as deficiencies are identified.

  The Committee plans to review the procedure that the staff will develop to define an appropriate Pi group range for scaling integral test facilities.

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 June 1, 2004. The following items were discussed:

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

Member assignments and priorities for ACRS reports and letters for the July ACRS meeting were attached. 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 October 2004 were considered. The objectives were to:

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

- Revision to ACRS Action Plan

A copy of the revised Action Plan was sent to the members following the May ACRS meeting. Members were requested to provide their comments to Mrs. Weston by June 4, 2004. The current revision to the Action Plan reflects incorporation of the limited comments received from the members.

- Visit to a Nuclear Plant and Regional Office

Several members (Bonaca, Ford, Leitch, Ransom, Rosen, Sieber, and Wallis) are scheduled to visit the D.C. Cook Nuclear Plant on Wednesday, June 9, 2004, and the NRC Region III Office on Thursday, June 10, 2004. Reservations have been made at the Hyatt Lisle Hotel, (630-852-1234) 1400 Corporetum Drive, Lisle, Illinois.

- Tour of Test Facilities Used for the ACR-700 Design

During the May 2004 ACRS meeting, Drs. Ford, Kress, Ransom, and Wallis expressed interest in touring the Chalk River Facility used for the ACR-700 design and participating in a joint meeting of the ACRS Subcommittees on Future Plant Designs and Materials and Metallurgy to discuss various aspects of the ACR-700 design, including materials issues. The tour and the meeting were originally scheduled for July 25-30, 2004. However, a workshop regarding the AP1000 design has been scheduled by the NRC staff on July 26-29, 2004, to be held in China. Dr. Kress was invited to join the NRC Panel to participate in this workshop. During the 512th ACRS meeting in May 2004, the Committee approved Dr. Kress' participation in such workshop. Accordingly, the trip to the Chalk River Facility in Canada will be postponed, possibly till August, September, or October 2004.

- EDO Response to the Anonymous E-mail Regarding the NRC Staff's Process for Reviewing the TRACE Computer Code

A member of the public sent an anonymous e-mail to Dr. Wallis on February 20, 2004, criticizing the process being used by the NRC staff in the development and review of the TRACE computer code. As agreed to by the Committee during its March 2004 meeting, the ACRS Executive Director referred this matter to the EDO for action.

In a memorandum dated April 16, 2004, to the ACRS Executive Director, the EDO addressed the issues raised in the anonymous e-mail. In the response, the EDO recommended that the concerns expressed in the anonymous e-mail be discussed by the ACRS Subcommittee on Thermal-Hydraulic Phenomena during its review of the TRACE code.

- Vermont Yankee Power Uprate

In letters dated March 15 and 31, 2004 to the NRC Chairman, the State of Vermont Public Service Board requested that the ongoing NRC review of the Vermont Yankee power uprate request by Entergy include several new facets, including an "independent assessment" of the plant.

On May 4, 2004, NRC Chairman Diaz responded to the Vermont Public Service Board. He explained that the NRC decided to conduct a detailed engineering inspection and which will be appropriate for addressing its oversight responsibilities. This is also responsive to the concern expressed by the Vermont Public Service Board.

The Chairman also noted that the NRC had been developing a new engineering inspection program which it intends to pilot at selected plants. The new program incorporates the best practices of existing and past engineering inspections and would be appropriate to conduct this new engineering inspection at Vermont Yankee.

The ACRS will also review the Vermont Yankee power uprate request. The NRC staff will provide the results of its review efforts, including relevant inspection findings, to the ACRS for review. After the ACRS completes its review, it will make an independent recommendation regarding whether the proposed power uprate amendment should be approved.

- Interview of Candidates for Potential Membership on the ACRS (Closed)

During the June ACRS meeting, the ACRS Member Candidate Screening Panel and the ACRS members interviewed five candidates for potential membership with the ACRS.

ACRS Assessment of the Quality of the NRC Research Programs

In a letter dated April 26, 2004, the ACRS outlined a strategy for assessing the quality of the NRC research projects. Out of eight projects provided by RES, the Committee selected the following projects for assessment during the remainder of FY 2004 and made assignments as noted below.

- Sump Blockage - Mr. Rosen (Chair), Kress, and Wallis
- MACCS Code - Dr. Kress (Chair), Apostolakis, and Sieber

In a letter dated May 20, 2004, Dr. Paperiello, RES Director, stated that RES appreciates the Committee's willingness to assist RES in assessing the quality of the NRC research projects.

Since the Committee has committed to provide a summary report on its assessment of the above two research projects at the end of this fiscal year, it would be helpful if Mr. Rosen and Dr. Kress provided a status report on the progress of their Panels in assessing the quality of the assigned projects in July 2004.

- Member Issues

Mr. Sieber suggested that instead of sending a CD, which contained background materials for the ACRS full Committee meetings, the cognizant staff engineers e-mail the meeting documents to the members.

Requirements of the Federal Advisory Committee Act (FACA) requires that all documents provided to the Committee, including all reports, letters, agendas, and studies prepared by or for the Committee be maintained for the duration of the Committee. Having all documents for each meeting stored in a single CD will be a more efficient way of complying with the FACA requirement. As has been the practice, the staff engineers will e-mail the status reports and background documents to the members before each meeting.

- Travel Request

Mr. Rosen requested Committee approval and support to attend the NEI Fire Protection Information Forum to be held in Florida between September 18-23, 2004.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 514th ACRS Meeting, July 7-9, 2004.

The 513th ACRS meeting was adjourned at 3:15 p.m. on June 4, 2004.

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

July 29, 2004

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

FROM: Mario V. Bonaca
Chairman

SUBJECT: CERTIFIED MINUTES OF THE 513th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), JUNE 2-4, 2004

I certify that based on my review of the minutes from the 513th 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.

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

June 21, 2004

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador *Sherry Meador*
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 513th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - JUNE 2-4, 2004

Enclosed are the proposed minutes of the 513th 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

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–346]

FirstEnergy Nuclear Operating Company, Davis-Besse Nuclear Power Station; Environmental Assessment and Finding of No Significant Impact

The U.S. Nuclear Regulatory Commission (NRC) is considering withdrawal of an exemption from title 10 of the Code of Federal Regulations (10 CFR) part 50, Appendix R, subsection III.L.1 for Facility Operating License No. NPF–3, issued to FirstEnergy Nuclear Operating Company (FENOC or the licensee), for operation of the Davis-Besse Nuclear Power Station (DBNPS), located in Ottawa County, Ohio. Therefore, as required by 10 CFR 51.21, the NRC is issuing this environmental assessment and finding of no significant impact.

Environmental Assessment

Identification of the Proposed Action

The proposed action would withdraw an exemption to 10 CFR part 50, Appendix R, subsection III.L.1, regarding the plant's capability to achieve cold shutdown within 72 hours by the alternative shutdown process, independent of offsite power.

The proposed action is in accordance with the licensee's application dated December 17, 2003.

The Need for the Proposed Action

The action is proposed because the licensee has now determined that DBNPS can achieve cold shutdown within 72 hours by the alternative shutdown process independent of offsite power; therefore, the exemption is no longer required.

Environmental Impacts of the Proposed Action

The NRC has completed its evaluation of the proposed action and concludes that the proposed exemption withdrawal does not involve radioactive wastes, release of radioactive material into the atmosphere, solid radioactive waste, or liquid effluents released to the environment.

The DBNPS systems were evaluated in the Final Environmental Statement (FES) dated October 1975 (NUREG 75/097). The proposed exemption withdrawal will not involve any change in the waste treatment systems described in the FES.

The proposed action will not significantly increase the probability or consequences of accidents, no changes are being made in the types of effluents that may be released offsite, and there is no significant increase in occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed action.

With regard to potential nonradiological impacts, the proposed action does not have a potential to affect any historic sites. It does not affect nonradiological plant effluents and has no other environmental impact. Therefore, there are no significant nonradiological environmental impacts associated with the proposed action.

Accordingly, the NRC concludes that there are no significant environmental impacts associated with the proposed action.

Environmental Impacts of the Alternatives to the Proposed Action

As an alternative to the proposed action, the staff considered denial of the proposed action (*i.e.*, the "no-action" alternative). Denial of the application would result in no change in current environmental impacts. The environmental impacts of the proposed action and the alternative action are similar.

Alternative Use of Resources

The action does not involve the use of any different resource than those previously considered in the DBNPS FES dated October 1975.

Agencies and Persons Consulted

On April 16, 2004, the staff consulted with Ohio State official, C. O'Claire of the Ohio Emergency Management Agency, regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated December 17, 2003 (ADAMS ML033600026). Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, Public File Area O1–F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management System (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1–800–397–4209 or 301–415–4737, or by e-mail to *pdr@nrc.gov.*

Dated in Rockville, Maryland, this 12th day of May, 2004.

For the Nuclear Regulatory Commission.

Jon B. Hopkins,

Senior Project Manager, Project Directorate III, Section 2, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 04–11297 Filed 5–18–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting Notice

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on June 2–4, 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, June 2, 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.–10:30 a.m.: Draft Final 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Nuclear Energy Institute (NEI) regarding the draft final 10 CFR 50.69, and draft final Regulatory Guide DG–1121, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significances," which endorses NEI 00–04, "10 CFR 50.69 SSC Categorization Guideline."

10:45 a.m.–11:45 a.m.: Revised License Renewal Review Process (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the revised process for the staff's review of the license renewal applications.

12:45 p.m.–1:15 p.m.: Preparation for Meeting with the NRC Commissioners (Open)—The Committee will discuss the following topics scheduled for the ACRS meeting with the NRC Commissioners: PWR Sump Performance, PRA Quality for Decisionmaking, Risk-Informing 10 CFR 50.46, NRC Safety Research Program Report, Economic Simplified Boiling Water Reactor (ESBWR) Pre-Application Review, and Interim Review of the AP1000 Design.

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

4 p.m.–5:30 p.m.: Digital Instrumentation and Control System Research Activities (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and their contractors regarding NRC research activities in the area of digital instrumentation and control (I&C) systems and related matters.

5:45 p.m.–6:45 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting.

Thursday, June 3, 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.–10:30 a.m.: NRC Staff's Response to the ACRS Report on the AP1000 Design (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding their response to ACRS comments and recommendations included in the March 17, 2004 ACRS report on the AP1000 design.

10:45 a.m.–12 Noon: Proposed Revisions to Standard Review Plan (SRP) Sections and Process and Schedule for Revising the SRP (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the proposed revisions to SRP Sections: 5.2.3, "Reactor Coolant Pressure Boundary Materials;" 5.3.1, "Reactor Vessel Materials;" and 5.3.3, "Reactor Vessel Integrity;" as well as the process and schedule for revising various SRP Sections, including milestones for ACRS review of the proposed revisions.

1:30 p.m.–2:30 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.

2:30 p.m.–2:45 p.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations (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.

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

Friday, June 4, 2004, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

8:30 a.m.–11 a.m.: Metrics for Evaluating the Quality of the NRC Research Programs (Open)—The Committee will discuss the quantitative metrics for use by the ACRS in evaluating the quality of the NRC research programs.

11 a.m.–4 p.m.: Preparation of ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

4 p.m.–4:30 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.

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 therefore 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: May 13, 2004.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 04–11298 Filed 5–18–04; 8:45 am]

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

May 12, 2004

SCHEDULE AND OUTLINE FOR DISCUSSION
513th ACRS MEETING
JUNE 2-4, 2004

WEDNESDAY, JUNE 2, 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)
   1.2) Opening Statement
   1.2) Items of current interest

2) 8:35 - ~~10:30~~ 10:25 A.M. Draft Final 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors" (Open) (GEA/MRS)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and Nuclear Energy Institute (NEI) regarding the draft final 10 CFR 50.69, and draft final Regulatory Guide DG-1121, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power plants according to their Safety Significances," which endorses NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline."

10:25- ~~10:30~~ - 10:45 A.M. ***BREAK***

3) 10:45 - 11:45 A.M. Revised License Renewal Review Process (Open) (MVB/MDS)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding the revised process for the staff's review of the license renewal applications.

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

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

4) 12:45 - ~~1:15~~ 1:05 P.M. Preparation for Meeting with the NRC Commissioners (Open) (MVB, et.al/JTL, et.al)
   Discussion of the following topics scheduled for the ACRS meeting with the NRC Commissioners:
   a) Overview (MVB)
   b) PWR Sump Performance (JDS)
   c) PRA Quality for Decisionmaking (GEA)
   d) Risk-Informing 10 CFR 50.46 (WJS)
   e) NRC Safety Research Program Report (DAP)
   f) ESBWR Pre-Application Review (TSK)
   g) Interim Review of the AP1000 Design (TSK)

1:15 - 1:30 P.M. *****BREAK*****

5) 1:30 - 3:30 P.M. Meeting with the NRC Commissioners, Commissioners' Conference Room, One White Flint North, Rockville, MD (Open) (MVB, et.al/ JTL, et.al)
Meeting with the NRC Commissioners to discuss the topics listed under Item 4.

3:30 - 4:00 P.M. *****BREAK*****

6) 4:00 - 5:30 P.M. Digital Instrumentation and Control System Research Activities (Open) (JDS/GEA/MDS)
- 6.1) Remarks by the Subcommittee Chairman
- 6.2) Briefing by and discussions with representatives of the NRC staff and their contractors regarding NRC research activities in the area of digital instrumentation and control (I&C) systems and related matters.

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

5:30 - 5:45 P.M. *****BREAK*****

7) 5:45 - ~~6:45~~ 6:40 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
- 7.1) Draft Final 10 CFR 50.69 and Regulatory Guide DG-1121 (GEA/MRS)
- 7.2) Digital I&C Research Activities (JDS/GEA/MDS)

6:45-7:30 pm Safeguards & Security Meeting, T8E-8 (CLOSED)

THURSDAY, JUNE 3, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

9) 8:35 - ~~10:30~~ 10:35 A.M. NRC Staff's Response to the ACRS Report on the AP1000 Design (Open) (TSK/MME)
- 9.1) Remarks by the Subcommittee Chairman
- 9.2) Briefing by and discussions with representatives of the NRC staff regarding their response to ACRS comments and recommendations included in the March 17, 2004 ACRS report on the AP1000 design.

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

10:35-10:55
~~10:30 - 10:45~~ A.M. *****BREAK*****

10:55 - 11:55 am
10) ~~10:45 - 12:00 Noon~~ Proposed Revisions to Standard Review Plan (SRP) Sections and Process and Schedule for Revising the SRP (Open) (FPF/SD)
- 10.1) Remarks by the Subcommittee Chairman
- 10.2) Briefing by and discussions with representatives of the NRC staff regarding proposed revisions to SRP Sections: 5.2.3; "Reactor Coolant Pressure Boundary Materials;" 5.3.1, "Reactor Vessel Materials;" and 5.3.3, "Reactor Vessel Integrity;" as well as the process and schedule for revising various SRP Sections, including milestones for ACRS review of the proposed revisions.

11:55 -
~~12:00~~ - 1:30 P.M. *****LUNCH*****

11) 1:30 - ~~2:30~~ 2:45 P.M Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (MVB/JTL/SD)
- 11.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.
- 11.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

2:45 - 2:55
12) ~~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.

2:55 - 3:15
~~2:45 - 3:00~~ P.M. *****BREAK*****

3:15 - 4 45
13) ~~3:00~~ - 6:30 P.M. Preparation of ACRS Reports (Open)
Discussion of the proposed ACRS reports on:
- 13.1) Draft Final 10 CFR 50.69 and Regulatory Guide DG-1121 (GEA/MRS)
- 13.2) Digital I&C Research Activities (JDS/GEA/MDS)
- 13.3) Proposed Revisions to SRP Sections (Tentative) (FPF/SD)

5:00-6:45 pm Safeguards & Security letter

FRIDAY, JUNE 4, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

8:30-10:00 am Safeguards & Security letter
14) ~~8:30 - 11:00 A.M.~~ Metrics for Evaluating the Quality of the NRC Research Programs (Open) (GEA/HPN)
10:00-10:15 A.M. BREAK)
10:15 - 11:50
Discussion of the quantitative metrics for use by the ACRS in evaluating the quality of the NRC research programs.

1:30 - 3:15 pm
15) ~~11:00 - 12:00 Noon~~ Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports listed under Item 13.

3:15 Adjourn

12:00 - 1:30 P.M. * LUNCH*****

16) 1:30 - 4:00 P.M. Preparation of ACRS Reports (Open)
Continue discussion of the proposed ACRS reports listed under Item 13.

17) 4:00 - 4:30 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.

NOTE:

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

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

APPENDIX III: MEETING ATTENDEES

513TH ACRS MEETING
June 2, 2004

NRC STAFF

P. T. Kuo	NRR	R. Sullivan	NRR	S.Bailey	NRR
J. Dozier	NRR	T. Liu	NRR	D. Tuner	NRR
S. West	NRR	S. Lee	NRR	P. Burke	NRR
J. Yerokum	NRR	J. Rowley	NRR	D. Guha	NRR
M. Heath	NRR	K. Chary	NRR	K.R. Hsu	NRR
R. Auluck	NRR	L. Lund	NRR	D. Dube	NRR
S. Arndt	NRR	M. Waterman	NRR	R. Shaffer	NRR
D. Harrison	NRR	M. Mitchell	NRR	T. Scarbrough	NRR
T. Reed	NRR	E. McKenna	NRR	D. Terao	NRR
M. Evans	RES	N. Dudley	NRR	J. Calvo	NRR
M. Chiramal	NRR	D. Duvigneaud	RES	D. Overland	RES
D. Tifft	RES	T. Guvan	RES	B. Kemper	OIG
J. Flack	RES	G. Parry	NRR	M. Young	NRR
D. Matthews	NRR	J. Wilson	NRR	S. Dinsmore	NRR
M. Lintz	NRR	K. Cuzens	NRR	G. Cranston	NRR
C. Lauron	NRR	M. Murphy	NRR	J. Honcharik	NRR
J. Eads	NRR	S. Ng	NRR	O. Yee	NRR
B. Wong	NRR	D. Chen	NRR	T. Terry	NRR
D. Fischer	NRR	P. Schemanski	NRR		

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

G. Schinzel	STPNOC
N. Chapman	SERCH/Bechtel
S. Levinson	AREVA
B. Bradley	NEI
Y. Guan	ASTM
F. Polaski	Exelon
E. Blocher	Parson Power
C. Pierce	Southern Nuclear
D. Flyte	PPL Susquehanna LLC
R. Grumbir	AEP - D.C. Cook
T. Chu	BNL
G. Martinez	BNL
T. Pietrangelo	NEI
J. Brown	Westinghouse

APPENDIX III: MEETING ATTENDEES

513TH ACRS MEETING
June 3, 2004

NRC STAFF

R. Kuntz	NRR
D. McCain	NRR
L. Gerke	NRR
M. Mitchell	NRR
S. Arndt	NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

J. Scoger	Westinghouse
T. Schulz	Westinghouse
E. Cummins	Westinghouse
H. Esmaih	Energy Research
R. Orr	Westinghouse
J. Grover	Westinghouse
B. Hammersley	Westinghouse
B. Corroran	NSRC Corp.

513TH ACRS MEETING
June 4, 2004

NRC STAFF

A. Levin	RES
F. Ramirez	OE

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

None

APPENDIX IV

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

June 17, 2004 (REVISED)

SCHEDULE AND OUTLINE FOR DISCUSSION
514th ACRS MEETING
JULY 7-9, 2004

WEDNESDAY, JULY 7, 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
   - 1.2) Items of current interest

2) 8:35 - 10:30 A.M. Final Safety Evaluation Report (SER) Associated with the AP1000 Design Certification (Open) (TSK/MME)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff and Westinghouse Electric Company regarding the final SER associated with the certification of the AP1000 design, resolution of any unresolved issues previously raised by the ACRS, and related matters.

10:30 - 10:45 A.M. ***BREAK***

3) 10:45 - 12:15 P.M. Draft Final Generic Letter on Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at PWRs (Open) (GBW/RC)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final Generic Letter on PWR sump blockage and the staff's resolution of public comments on the proposed version of this Generic Letter.

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

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

4) 1:15 - 3:45 P.M. Risk-Informing 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors" (Open) (WJS/MRS)
   - 4.1) Remarks by the Subcommittee Chairman
   - 4.2) Briefing by and discussions with representatives of the office of Nuclear Reactor Regulation regarding the proposed rule language for risk-informing 10 CFR 50.46.

4.3) Briefing by and discussions with representatives of the Office of Nuclear Regulatory Research regarding sensitivity studies on large-break loss-of-coolant accident frequency reevaluation performed in support of risk-informing 10 CFR 50.46.

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

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

5) 4:00 - 5:00 P.M. Differences in Regulatory Approaches and Requirements Between U.S. and Other Countries (Open) (DAP/HPN/SD)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with Dr. Nourbakhsh, ACRS Senior Staff Engineer, regarding his draft White Paper on differences in regulatory approaches and requirements between U.S. and other countries.

5:00 - 5:15 P.M. ***BREAK***

6) 5:15 - 6:45 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

6.1) AP1000 Design Certification (TSK/MME)

6.2) Proposed Rule Language for Risk-Informing 10 CFR 50.46 (WJS/MRS) (Tentative)

6.3) Draft Final Generic Letter on PWR Sump Blockage (GBW/RC)

THURSDAY, JULY 8, 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 - 10:30 A.M. Proposed Generic Communication on the Use of Ultrasonic Flow Measurement Devices for Measuring Feedwater Flow Rates in Nuclear Plants (Open) (JDS/MWW/CS)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed Generic Communication on the Use of Ultrasonic Flow Measurement Devices for Measuring Feedwater Flow Rates in Nuclear Plants.

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

10:30 - 10:45 A.M. ***BREAK***

9) 10:45 - 11:45 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (MVB/JTL/SD)

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

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

9.3) Subcommittee Reports (Open)

a) Plant Operations (JDS/MWW)
Report by and discussions with the Chairman of the Plant Operations Subcommittee regarding the visit to the D.C. Cook Nuclear Plant and meeting with the NRC Region III personnel on June 9-10, 2004.

b) Thermal-Hydraulic Phenomena (GBW/RC)
Report by and discussions with the Chairman of the Thermal-Hydraulic Phenomena Subcommittee regarding the ongoing staff activities associated with the resolution of GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance," that were discussed during the June 22-23, 2004 meeting.

c) Future Plant Designs (TSK/MME)
Report by and discussions with the Chairman of the Future Plant Designs Subcommittee regarding the NRC staff's proposed technology-neutral framework document for future plant licensing that was discussed at the June 24, 2004 meeting.

10) 11:45 - 12:00 Noon 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.

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

11) 1:00 - 2:00 P.M. Status of the ACRS Members' Assessment of the Quality of Selected NRC Research Projects (Open) (SLR/TSK/RC/HPN)
Discussion of the status of the activities of the cognizant ACRS members associated with the assessment of the quality of the research projects on Sump Blockage and on MACCS Code.

2:00 - 2:15 P.M. ***BREAK***

12) 2:15 - 6:30 P.M. Preparation of ACRS Reports (Open)
Discussion of the proposed ACRS reports on:

12.1) AP1000 Design Certification (TSK/MME)

12.2) Proposed Rule Language for Risk-Informing 10 CFR 50.46 (WJS/MRS) (Tentative)

12.3) Draft Final Generic Letter on PWR Sump Blockage (GBW/RC)

12.4) Proposed Generic Communication on the Use of Ultrasonic Flow Measurement Devices (JDS/MWW/CS)

FRIDAY, JULY 9, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

13)	8:30 - 3:00 P.M. **(12:00-1:00 P.M. LUNCH)**	Preparation of ACRS Reports (Open) Continue discussion of proposed ACRS reports listed under Item 12.
14)	3:00 - 3:30 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.

NOTE:

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

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
513th ACRS MEETING
June 2, 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

1 Opening Remarks by the ACRS Chairman

1. Items of Interest, dated June 2-4, 2004

2 Draft Final 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

2. Draft Final 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components" [Tab-2 Handout]
3. Risk-Informed Part 50 Special Treatment Requirements 10 CFR 50.69 [Viewgraphs]
4. Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systms, and Components in Nuclear Power Plants According to Their Safety Sigificance" [Viewgraphs]

3. Revised License Renewal Process

5. Improved License Renewal Application Review Process [Viewgraphs]
6. Updating License Renewal Guidance Documents [Viewgraphs]

4. Preparation for Meeting with the NRC Commissioners

5. Meeting with the NRC Commissioners and Control System Research Activities

7. ACRS Meeting With the U.S. Nuclear Regulatory Commission [Viewgraphs]

6. Digital Instrumental and Control System Research Activities

8. Overview of NRC Digital I&C Research Program in Digital Systems Reliability [Viewgraphs/electronic]
9. Context in the Risk Assessment of Digital Systems [Article written by C. Garrett and G. Apostolakis, MIT]
10. Draft Regulatory Guide -1130 (Proposed Revision 2 of Regulatory Guide 1.152), "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants"

Appendix V
513th ACRS Meeting

9. NRC's Staff's Response to the ACRS Report on the AP1000

    9. AP1000 Design Certification Review [Viewgraphs]
    10. AP1000 Design Certification Review , June 3, 2004 ACRS Full Committee Meeting [Viewgraphs]
    11. FCI Modeling Using PM-Alpha/Esprose.m [Viewgraphs]
    12. St Analysis: Evaluation of Aerosol Removal Rates (1) [Viewgraphs]

10. Proposed Revisions to Standard Review Plan (SRP) Sections and Process and Schedule for Revising the SRP

    13. Standard Review Plan Update Process [Viewgraphs]
    14. Schedule of SRP Sections to be Updated in FY05 [Viewgraph]

11. Future ACRS Activities/Report of the Planning and Procedures Subcommittee

    15. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - June 3, 2004 [Handout 1]

12. Reconciliation of ACRS Comments and Recommendations

    16. Reconciliation of ACRS Comments and Recommendations [Handout #2]

14. Metrics for Evaluating the Quality of the NRC Research Programs

    17. The Value Tree for Starting Projects [Viewgraph]

Appendix V
513th ACRS Meeting

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
2	Draft Final 10 CFR 50.69, “Risk-Informed categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors” [Handout]
	1. Table of Contents
	2. Proposed Agenda/Schedule
	3. Project Status Report, dated June 2, 2004 [Internal Committee Use Only: Predecisional Material Attached]
	4. Memorandum dated May 17, 2004 to John T. Larkins from Catherine Haney, Director, Reactor Policy Rulemaking Program, DRIP/NRR, Subject: Final Rule - Section 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants” (with attachments)
3	1. Table of Contents
	2. Proposed Agenda/Schedule
	3. Project Status Report, dated June 2, 2004 [Internal Committee Use Only: Predecisional Material Attached]
6	1. Table of Contents
	2. Proposed Agenda/Schedule
	3. Project Status Report, dated June 3, 2004 [Internal Committee Use Only: Predecisional Material Attached]
9	1. Table of Contents
	2. Proposed Agenda/Schedule
	3. Project Status Report, dated June 3, 2004 [Internal Committee Use Only: Predecisional Material Attached] (with attachments)
10	1. Table of Contents
	2. Proposed Agenda/Schedule
	3. Project Status Report, dated June 3, 2004 [Internal Committee Use Only: Predecisional Material Attached]

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

513th FULL COMMITTEE MEETING

JUNE 2-4, 2004

June 2, 2004
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
PT Kuo	NRR/DRIP/RLEP
Jerry Dozier	NRR/DRIP/RLEP
Steve West	NRR/DRIP/RLEP
Ram Subbaratnam	NRR/DRIP/RLEP-B
TILDA LIU	" A
Stewart Bailey	NRR/DE/EMEB
SAM LEE	NRR/DRIP/RLEP
DARREL TURNER	NMC
PATRICK BURKE	NMC
Jim Yerokun	NRR/DSSA
Jonathan Rowley	NRR/RLEP
Debbie Guha	NRR/RLEP
MAURICE HEATH	NRR/DRIP/RLEP
Ken Chang	"
K. R. HSU	"
R. Auluck	" "
L. Lund	NRR/DE/EMCB
D. DUBE	RES/DRAA/OERAB
Steven Arndt	RES/DET/ERAB
Mike Waterman	RES/DET/ERAB
Christina Antonescu	RES/DET/ERAB
Roman Shaffer	RES/DET/ERAB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

513th FULL COMMITTEE MEETING

JUNE 2-4, 2004

June 2, 2004
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
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NAME	AFFILIATION
Donnie Harrison	NRC/NRR/DSSA/SPSB
Matthew A. Mitchell	NRR/DE/EMCB
John Fair	NRR/DE/EMEB
Thomas Scarbrough	NRR/DE/EMEB
JASON BROWN	WESTINGHOUSE
PAUL SHEMANSKI	NRC/NRR/DE/EEIB
DAVID FISCHER	NRC/NRR/DE/EMEB
TIM REED	NRC/NRR/DRIP/RPRP
Eileen McKenna	NRC/NRR/DRIP
Glen Schinzel	STPNOC
Nancy Chapman	SERCH/Bechtel
STANLEY LEVINSON	AREVA
Biff Bradley	NEI
David Terao	NRC/NRR/EMEB
Yu Guan	ASTM
FRED POLASKI	EXELON
FRED EMERSON	NEI
Eric Blocher	Parsons Power
Charles Pierce	Southern Nuclear
Michele Evans	RES/DET/ERAB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

513th FULL COMMITTEE MEETING
JUNE 2-4, 2004

June 2, 2004
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
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NAME	AFFILIATION
NOEL DUDLEY	NRC/NRR/DRIP
Dave Flyte	PPL Susquehanna LLC
R. GRUMBIR	AEP - D.C. Cook
TSONG-LUN CHU	BNL
Gerardo Martinez	BNL
MATT CHIRAMAL	NRC/NRR/DE
DyLanne Duvigneaud	RES/DET/ERAB
Dean Overland	RES/DRAA/PRAB
JOSE CALVO	NRR/DE/EEIB
Doug Tifft	RES/DET/ERAB
Tekia Govan	RES/DET/ERAB
BILL KEMPER	NRC/OIG

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

513th FULL COMMITTEE MEETING
JUNE 2-4, 2004

June 2, 2004
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
JOHN FLACK	NRC/RES/DSARE
GARETH PARRY	NRC/NRR/DSSA
Matthew Young	NRC/NRR/EMEB
DAVID MATTHEWS	NRC/NRR/DRIP
JERRY Wilson	NRC/NRR/DRIP
Stephen Dinsmore	NRC/NRR/DSSA
Mark Lintz	NRC/NRR/DRIP
KURT COZENS	NRC/NRR/DRIP
Tony Pietrangelo	NEI
GREG CRANSTON	NRC/NRR/DRIP
Carolyn Lauron	NRC/NRR/DE
Martin Murphy	NRC/NRR/DE
John Honcharik	NRC/NRR/DE
Johnny Eads	NRC/NRR/DRIP
Saiwah Ng	NRC/NRR/DRIP
On Yee	NRC/NRR/DRIP
Ben Wong	NRC/NRR/DRIP
David Chen	NRC/NRR/DRIP
TOMEKA TERRY	NRC/NRR/DRIP

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

513th FULL COMMITTEE MEETING

JUNE 2-4, 2004

June 3, 2004
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
Rob Kuntz	NRR
Debbie McCain	NRR
BARRY ELLIOT	NRR
Laura Gerke	NRR
Matthew A. Mitchell	NRR/DE/EMCB
STEVEN ARNDT	RES/DET/ERAB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

513th FULL COMMITTEE MEETING
JUNE 2-4, 2004

June 3, 2004
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
Jim Scobel	Westinghouse
Terry Schulz	Westinghouse
Ed Cummins	Westinghouse
Hossein Esmaili	Energy Research
Richard Orr	Westinghouse
Jim Grover	Westinghouse
Bob Hammersley	Westinghouse
Ron Vijuk	Westinghouse
Bill Corcoran	NSRC Corp.
[illegible]	

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

513th FULL COMMITTEE MEETING

JUNE 2-4, 2004

June 4, 2004
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
Alan Levin	NRS/OD
Francesc Ramirez	OE

1

ITEMS OF INTEREST

513th ACRS MEETING

JUNE 2-4, 2004

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
513th MEETING
June 2-4, 2004

NRC ANNOUNCEMENTS

NUCLEAR NEWS FLASHES

INSIDE NRC ARTICLES

IN RESPONSE, PLEASE
REFER TO: M040504B

May 27, 2004

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON RESULTS OF THE AGENCY ACTION REVIEW MEETING, 9:30 A.M., TUESDAY, MAY 4, 2004, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff on the results of the Agency Action Review Meeting (AARM). The Commission identified the following items for staff follow-up:

1. In developing improved, risk-informed, performance indicators (PIs), the staff should try to recover the Mitigating Systems Performance Index (MSPI) efforts before initiating new efforts. The staff should work with stakeholders to develop clear requirements for PIs to enable them to be indicative of performance within the related cornerstone of safety.

2. The staff should continue efforts to better define thresholds for identifying and responding to substantive cross-cutting issues.

3. The Commission supports the flexibility provided in the Reactor Oversight Process (ROP) to allow deviations from the action matrix when senior managers find such deviations appropriate. When deviations occur, the staff should evaluate the causes for the deviations and identify changes to ROP, as appropriate, that may obviate the need for them in the future. This should be done as part of the staff's ROP self-assessment to ensure that the ROP meets the Agency's performance goals. Substantive changes should be provided to the Commission for approval prior to incorporation into the ROP.

4. The staff should inform the Commission when deviations from the action matrix are granted and highlight nuclear power plants (NPPs) for which such deviations are granted at the annual AARM Commission meeting.

5. As part of the normal self-assessment process, the staff should improve the standardization and transparency of the process for NPPs to exit from increased oversight columns in the action matrix. Additionally, the staff should standardize the process for requesting and documenting deviations from the action matrix.

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

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs Telephone: 301/415-8200

Washington, DC 20555-001 E-mail: opa@nrc.gov

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

No. S-04-007

BUILDING ON SUCCESS: THE REGULATORY CHALLENGE

"Between a Rock and a Nice Place"

Nuclear Energy Assembly, New Orleans

Remarks of Chairman Nils J. Diaz

U.S. Nuclear Regulatory Commission

May 13, 2004

Introduction

Good morning, ladies and gentlemen. It is a great pleasure and honor to address this milestone gathering today. As the aptly chosen title of this conference reminds us, this is a year of fiftieth anniversaries: fifty years since President Dwight Eisenhower's visionary and eloquent speech to the United Nations; fifty years also since a far-sighted Congress enacted the Atomic Energy Act, and opened the way to an era in which the atom would become an agent of human betterment rather than of destruction, bringing health to the sick; and safe, clean, affordable electrical power to an energy-deficient world.

Many things have changed in 50 years, but the NRC's job has not changed. Our job is to enable the use and management of radioactive materials and nuclear fuels for beneficial civilian purposes in a manner that

1. protects public health and safety and the environment,
2. promotes the security of our nation, and
3. provides for regulatory actions that are open, effective, efficient, realistic, and timely.

In the context of this conference's focus on "the next 50 years," I have been asked today to address the "regulatory perspective." I'd like to do so by reviewing where the NRC stands today in its regulation of nuclear power; the directions in which our regulatory program is moving; and the relationship between regulatory programs and industry's own role in the further growth and development of a safe nuclear option. I have subtitled this presentation "Between a Rock and a Nice

Place." It should be an interesting exercise for us to find the rocks, the hard places, and the nice places, looking at the past, the present, and the future of nuclear power. We can see occasionally difficult times, the improved safety and reliability of the industry's performance, and, yes, the improved predictability and consistency of the regulator's performance, as well as many challenges.

Nuclear Power Today: A Regulator's Viewpoint

From a regulator's standpoint, there are grounds for cautious optimism with the state of the nuclear power industry today. The level of reactor safety, and with it of plant reliability, has increased steadily; from the standpoint of American public protection, the record is admirable, with not a single member of the public ever exposed to a harmful level of radiation from a U. S. nuclear power plant; and nuclear power continues to be an essential element of our energy supply and of the nation's energy security. Indeed, we have reached a point at which, paradoxically, success itself could become a basis for concern. Too many years of excellent performance can all too easily bring on the kind of complacency that diminishes vigilance and puts that fine record in jeopardy. That this is a real issue, not a hypothetical one, was shown not long ago at Davis-Besse, where both the licensee and the NRC failed. This was a case where people failed to see a problem because they did not expect it and therefore did not look for it. It should be an object lesson to all of us, in industry and government, not to rest on our laurels -- indeed, not to rest, period. Success in the nuclear field has been achieved by maintaining vigilance, by being pro-active in inspection and surveillance, and by doing things rigorously and right.

Let me interject at this point that for the future viability and growth of nuclear power, a strong and credible regulator is a sine qua non (for us engineers, it's an unyielding boundary condition), and there are no two ways about it. In a democracy, activities are either regulated or not. Nuclear power needs to be regulated. In fact, I believe that the nuclear power industry needs a strong, fair, predictable, and credible regulator. This is, after all, an industry where many in the public regard any lapse by any utility as an indictment of the entire industry and of nuclear technology in general. That may not be fair, it may not be reasonable, but realistically, that is the world we live in.

Nuclear Safety Regulation: Evolving Approaches

In discussing the NRC's evolving approach to nuclear safety regulation, it may be appropriate to begin by mentioning another anniversary: the Three Mile Island accident, which took place 25 years ago this spring. There is no doubt that the industry and the NRC were at the time between a rock and a hard place. That event, as we all know, was a milestone in the history of commercial nuclear power, and not a happy one. Notwithstanding that the reactor was ultimately brought to safe shutdown without physical harm to any member of the public, it dealt a major setback to the development of the nuclear option in this country, largely because of the impact on public opinion. Public opinion can become a "rock" if the public is not given the factual information it needs. Even this setback was not without its benefits, however, for it was a wake-up call, shaking the industry and the NRC out of some complacency about the safety of nuclear plants, and it resulted in important advances in the way that both the industry and its regulators did business. The creation of the Institute for Nuclear Power Operations (INPO), providing internal industry discipline to identify weaker performers and bring them up to the mark, was a major step forward. For the NRC, the post-accident reviews confirmed the soundness of the approach employed several years earlier in WASH-1400, the "Reactor Safety Study": namely, the use of risk analysis and risk insights to ensure that resources are targeted optimally, to identify the types of accidents which are most important and the best ways to reduce their probability and consequences. In the years since 1995, we have made major strides in integrating the use of PRA into our regulatory structures, though much remains to be done.

Today, we have both the technical information and the analytical tools that we need to make substantial additional progress toward a risk-informed and performance-based regulatory framework. Furthermore, the will is there to move forward. Both in the NRC and in the nuclear industry, risk-informed decision-making has become an everyday tool. There is therefore no reason why these approaches cannot be taken to the next level, and incorporated in the basic design requirements as well.

In the near future, we will see two important steps in this area: 10 CFR 50.69, addressing special treatment requirements, and 50.46, addressing emergency core cooling system requirements. The proposed rule on the emergency core cooling system and loss of coolant accident (LOCA), now in preparation, will provide that the very low probability large break LOCA should be treated not as the design basis accident, but in the context of a required severe accident management program, as one of the severe accident scenarios to be addressed. This means that the really important accident scenarios, the ones with risk significance, would remain within the design basis; this approach would mean a new focus on them and should result in an increase in safety.

Integrating Safety, Security, and Emergency Preparedness

The events of 9/11 brought to this country a new recognition of the importance of physical security and emergency preparedness in the world of 21st century America. In the case of the NRC and the nuclear industry, this awareness had come already decades ago, and to that extent, we were, so to speak, ahead of the curve: for a generation, our regulations had postulated the existence of a terrorist threat, as part of defense-in-depth. Thus the kind of drastic changes in security seen in the airline industry, for example, were not required for nuclear plants, because we had put those structures in place long ago. To be sure, significant enhancements were made, and security orders were issued that tightened existing policies and procedures in the light of the most current information, but it was not a wholesale revamping of our entire regulatory structure.

What the post-9/11 review of security issues highlighted is how tightly interconnected reactor safety, security and emergency preparedness are. Many of the same issues are involved in avoiding and mitigating reactor accidents as in preventing and mitigating acts of terrorism. Though the initiating events may differ, defense-in-depth applies in very similar ways to both. The same principles are applicable -- high quality of design, fabrication, and testing; multiple barriers to fission product release; redundancy and diversity in safety equipment; and procedures and strategies in place for addressing expected as well as unexpected events. The essence of the problem is the same in both cases: to shut down the reactor, cool the core, and maintain the integrity of protective barriers. This industry as a whole has literally centuries of operating experience in doing just that.

Allow me to frame for you, and the American public, where we are with respect to our assessment of safety, security, and emergency preparedness. As you may know, the NRC has conducted an extensive analysis of the potential vulnerability of nuclear power plants to aircraft attacks. While this analysis is classified, the NRC remains convinced that nuclear power plants are among the most heavily protected civilian facilities in the United States. Our vulnerability studies confirm that the likelihood of damaging the reactor core and releasing radioactivity that could affect public health and safety is low. The reasons for this are clear. Nuclear reactor design requirements for structures to withstand severe external events (hurricanes, tornadoes and floods), and for safety systems to include redundant emergency core cooling, redundant and diverse heat removal, fire protection features, and station blackout capabilities, provide built-in means of dealing with attempted terrorist attacks. Existing emergency operating procedures and enhanced severe accident management guidelines are well suited for mitigating the effects of accidents or intentional attacks on nuclear power plants. In addition, all nuclear power plants have been required to enhance their safety, security, and emergency preparedness. Given these enhancements made to safety, security, and emergency preparedness, the potential radiological consequences to the public of an aircraft attack are low.

Further, the studies confirm that even in the unlikely event of a radiological release due to terrorist use of a large aircraft, NRC's emergency planning basis remains valid. Thus, we believe that nuclear power plant safety, security, and emergency planning programs continue to provide reasonable assurance of adequate protection of the public health and safety.

The analyses, conclusions and insights that I just presented for nuclear power plants also apply to spent fuel pools, since they are also well engineered and protected structures, and are amenable to simple and effective mitigative actions, if needed. For a dry spent fuel storage cask, it is highly unlikely that aircraft impacts on a cask would cause a significant release of radioactive material. In addition, results to date show that a large commercial aircraft crashing into a transportation cask would not result in a release of radioactive material.

In summary, I believe that the NRC and the industry have done their jobs well. The NRC has assessed what needed to be done and the industry has done it well. The NRC, other government organizations, and the licensees have taken action to provide adequate protection of the people of our nation.

Looking to the Present

I believe that this gathering has a strong interest in the NRC priorities today, as outlined in my recent address to the senior managers of the agency. The list is not all-inclusive, and these are key items:

- - Preparing a plan to enhance internal communication and ensuring that everyone in the agency has the information to do their job effectively and efficiently;

- - Completing the Davis Besse Lessons Learned Task Force recommendations;

- - Enhancing oversight of reactor engineering issues, including increased use of risk insights, operating experience and increased staff training;

- - Developing and implementing a pro-active approach to prevention, detection, mitigation, passivation, and repair of reactor materials degradation;

- - Preparing a plan to enhance the effectiveness of emergency preparedness and incident response, including implementing an improved training program for NRC responders;

- - Conducting a review of the scope, schedules with milestones, and deliverables of all research projects, and assuring alignment with Commission policies;

- - Completing a final action plan and milestone schedule to match tasks and resources for the potential review of an application for the licensing of the high-level waste repository;

- - Preparing a plan to enhance the training and execution of the staff's critical thinking skills. This should include the use of risk information, realistic conservatism, and other insights to help identify what issues are truly important, how they relate to other issues, and how they can be approached in an integrated and holistic manner; and

- - Completing an initial national inventory of high-risk sources, and proposing a plan for continuing its management.

Another high priority agency activity concerns rulemaking for export/import of high-risk radioactive sources, implementing the IAEA Code of Conduct.

Looking to the Future

The question may be asked, what do we as regulators expect from the nuclear industry in the years ahead. I would answer by saying that we must demand of the industry the same thing we demand of ourselves: an unconditional commitment to safety, security, and preparedness.

Both industry and the NRC must have a comprehensive safety management approach, focused on safety engineering, operations, and maintenance, and driven by a safety management program. For these purposes, a safety management program implies three central elements: a functional, executable commitment to operational, maintenance, and engineering safety; applying technical expertise where and when it should be; and, utilizing the people and resources that are needed to make it happen.

I personally believe that the nuclear option will have a growing part to play in this country's energy mix, as the nation -- the private sector, the public sector, and the American people -- soberly weighs the costs and benefits of different energy options. The NRC, as a regulator, is ready to do its part in ensuring that nuclear technology continues to be a safe and reliable source of power for the needs of our democracy and its citizens. For our part, we have been efficiently conducting, and will continue to conduct, the appropriate safety reviews in such areas of high priority agency activity as power uprates, license renewal, and new reactor design certifications. I believe these actions have been extremely important for maintaining an infrastructure for the safe use of nuclear energy. We must always bear in mind, however, that no technology is better, in the end, than the human beings who sit at the controls and oversee its operations, both in the sense of managerial direction and regulatory oversight.

The responsibility is an awesome one. Nuclear energy can continue to be an essential component of our energy supply, energy security, and environmental stewardship, and its use can expand, but only if a rigorous and dedicated commitment to safety is maintained. If we meet our various responsibilities -- as I know we can and feel sure that we will -- then 50 years from now, our successors will look back on the centennial of President Eisenhower's great vision, which ushered in the civilian nuclear era, in an America where nuclear power, safe and environmentally responsible, is ever more vitally a bulwark of our national security and prosperity.

Thank you.

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REMARKS BY
NILS J. DIAZ, CHAIRMAN

U.S. NUCLEAR REGULATORY COMMISSION

AT THE
ALL EMPLOYEES MEETING
MORNING SESSION

10:30 A.M. WEDNESDAY, May 26, 2004
PLAZA AREA, WHITE FLINT COMPLEX

Good morning, and welcome to the NRC's annual All Employees Meeting. Commissioner McGaffigan, Commissioner Merrifield, and I are pleased to join you in the tent "on the green" to answer your questions and address your concerns to the best of our ability. Our format this morning will be the same as in the past -- following my brief remarks, we will turn the meeting over to you. We encourage you to use this time to communicate with us.

I want to welcome those members of our staff who are located in the Regional Offices, at the Technical Training Center in Chattanooga, and at other sites throughout the country, all of whom are linked to our session this morning as well as to the second session that will take place this afternoon.

We have accomplished some very important objectives since our last All Employees Meeting, and several new challenges are about to begin or are on the horizon. I intend to be brief and very selective in what I cover this morning, so if I fail to mention an activity on which you personally are spending lots of time and attention, it is not a sign of Commission disinterest -- the Commission values the work that all of you are doing and your efforts to help us achieve the agency's mission.

Let me just briefly state at the very beginning for the benefit of our regional office employees that there is nothing before the Commission involving reorganizing the regions. I know this is the subject of ongoing concern and generates at least one question in each of these All Employees Meetings over the last few years, so I thought we should put that thought to rest this year early on.

As you know, the terrorist attacks of September 11, 2001, although not directed at NRC-licensed facilities or activities, have generated some profound changes at the NRC, in the nuclear industry, and in public perceptions about security. In fact, on several occasions this year, you have heard me say that the NRC of today is no longer the NRC with which you are familiar -- we are no longer just a safety agency, but rather a safety, security, and preparedness agency. Since 9/11, we have enhanced security requirements at nuclear facilities and for radioactive materials in many ways. This includes issuing a series of orders imposing new requirements on our licensees, revising the Design Basis Threat, working to improve coordination with Federal, State, and local officials, and organizing the NRC to put us in a better position to implement the necessary changes. It has been a very intense, exhausting, but very productive period. We have done our job well, we have addressed what needed to be done, and we have done it. My Commission colleagues and I are proud of what the NRC has ccomplished and grateful for all your hard work.

believe we are approaching a period of stability in the security arena and I am sure we are all eager to get there and to ıave stable and effective processes to deal with every aspect of security. The Commission, and, I hope, all of you recognize hat e Dorothy in The Wizard of Oz, we do not have the option of returning back home to Kansas. Security concerns will en th us for as long as any of us can foresee, and we will need to ensure that our new security requirements continue o be plemented effectively. Fundamentally, we must keep in mind that we do have a continuing role to play in "promoting he common defense and security," but that role needs to be seen in a balanced perspective with our other responsibilities ıow that we have taken the steps necessary to enhance security.

Nhat we need to do now is to continue to integrate security with other areas, like safety and preparedness, in a logical and, /es, natural way. This is natural because the concerns raised in the security arena involve many of the same issues involved n avoiding and mitigating accidents. The safety solution would be the same for both cases: to shut down the reactor, cool :he core, and maintain the integrity of protective barriers. Our approach to safety, security, and emergency preparedness is :herefore an integrated activity that will ensure protection of the public. When our defense-in-depth procedures to accomplish these ends are employed on site, we consider defense-in-depth to be in the realm of reactor safety; when we apply them off site, we consider defense-in-depth part of emergency preparedness.

In the reactor arena, we dealt with the Davis-Besse hole-in-the-head issue, and the plant is now operating at full power for the first time since February 2002. It is critical that we prevent a recurrence of such a challenge to reactor safety. For this reason, we must expeditiously implement the remainder of the Task Force's recommendations. We are also moving forward with risk-informed and performance-based regulation to ensure a more focused attention on what is truly important to safety.

Our materials program is also in the midst of a significant change in focus. We are, of course, only a few months away from the anticipated submission by the Department of Energy of an application to construct a high-level waste repository at Yucca Mountain, Nevada. The NMSS staff, the Atomic Safety and Licensing Board, OGC, and other offices are all engaged in activities to prepare the NRC for its central role in this one-of-a-kind licensing process. The Commission is confident that we are prepared to fulfill our role and equally sure that once the process has begun, we will find it one of the most closely watched and contentious activities in which we have ever been engaged. In addition to the high-level waste repository, the NM aff is also continuing to review a request to authorize construction of a mixed-oxide (MOX) fuel fabrication facility at th nnah River Site in South Carolina as part of DOE's program to dispose of excess weapons grade plutonium and a proposed new uranium enrichment facility to be located in New Mexico. An application for a second enrichment facility in Ohio is expected in August. All of these activities are breaking new ground for the NRC.

We have implemented most of the changes in our senior management assignments that we announced recently. These changes have already taken place or soon will take place. I have personally experienced the disciplined manner in which senior management changes are done at the NRC when I took over as Chairman after former Chairman Meserve's departure. I am very pleased and proud of the manner in which our senior managers have addressed and discharged their new responsibilities. In remarks I delivered to a meeting of all senior managers earlier this month, I stressed the need to bring a new sense of commitment and awareness to their new responsibilities, to retain what seems to be working and change what is not, and to manage issues and personnel to a new level of effectiveness and efficiency. I challenged the senior management and I challenge all of you to make the NRC work even better than before as an integrated safety, security, and preparedness agency, where enhanced internal communications are being used to manage issues better, and enhanced external communications are being used to keep the American people better informed.

We have a lot on our plate for the coming year -- reviews associated with new power reactor licensing, license renewals, power uprates, fuel enrichment facilities, high-level waste disposal, oversight of licensed facilities, security -- the list goes on and on. I have only mentioned a few in any detail, but I want to stop here and open the meeting up to questions from the floor. I do want to conclude by emphasizing once again that the Commission has the utmost confidence in the ability of the NRC staff to meet the challenges before us. I also want to thank all of you personally for the support you have given the Commission and for the service you are providing to the American people.

Do my fellow Commissioner have initial comments?

May we have the first question, please.

May 5, 2004

Mr. James Wells, Director
Natural Resources and Environment
United States General Accounting Office
441 G Street, NW
Washington, D.C. 20548

Dear Mr. Wells:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to your letter of April 2, 2004, requesting the NRC's review of the draft report entitled "Nuclear Regulation: NRC Needs to More Aggressively and Comprehensively Resolve Issues Related to the Davis-Besse Nuclear Power Plant's Shutdown" (GAO-04-415). I appreciate the opportunity to provide comments to the General Accounting Office (GAO) on this report.

I am concerned that the draft report does not appropriately characterize or provide a balanced perspective on the NRC's actions surrounding the discovery of the Davis-Besse reactor vessel head condition or NRC's actions to incorporate the lessons learned from that experience into our processes. The NRC also does not agree with two of the report's recommendations, as discussed in the following paragraphs.

The first sentence of the draft report states: "...oversight did not generate accurate, complete information on plant conditions." I agree that our oversight program should have identified certain evolving plant conditions for regulatory follow-up. This was also identified in the report of the Davis-Besse Lessons Learned Task Force (LLTF) that the NRC formed to ensure that lessons from the Davis-Besse experience are learned and appropriately captured in the NRC's formal processes. However, the draft report does not acknowledge that the NRC, in carrying out its safety responsibilities, must rely heavily on our licensees to provide us with complete and accurate information. In fact, Title 10 of the Code of Federal Regulations Section 50.9 requires that information provided to the NRC by a licensee be complete and accurate in all material respects. The report should clearly indicate that NRC's licensees are responsible for providing us with accurate and complete information. While the NRC's Davis-Besse LLTF concluded that the NRC, the Davis-Besse licensee (FirstEnergy), and the nuclear industry failed to adequately review, assess, and follow up on relevant operating experience, they also noted that the information that FirstEnergy provided in response to Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles" was inconsistent with information identified by the task force. Further, the LLTF report stated that had this information been known in the fall of 2001, "...the NRC may have identified the VHP [vessel head penetration] nozzle leaks and RPV [reactor pressure vessel] head degradation a few months sooner than the March 2002 discovery by the licensee." As you are aware, there is an ongoing investigation by the Department of Justice regarding the completeness and accuracy of information that FirstEnergy provided to the NRC on the condition of Davis-Besse.

The NRC is particularly concerned about the draft report's characterization of the NRC's use of risk estimates. The statement in the report that the NRC's "estimate of risk exceeded the risk levels generally accepted by the agency" is not factually correct. NRC officials pointed out to GAO and GAO's consultants, both in interviews and in written responses to GAO questions, that our estimate of delta core damage frequency was $5x10^{-6}$ per reactor year, not $5x10^{-5}$ per reactor year as indicated in the report. In fact, the NRC staff safety evaluation (attached to a December 3, 2002, letter to FirstEnergy) stated that the change in core damage frequency due to the potential for control rod drive mechanism nozzle ejection was consistent with the guidelines of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The enclosure to this letter provides detailed comments on issues of correctness and clarity in the report, many of which are related to the NRC's estimate of risk at Davis-Besse.

We disagree with the finding that the NRC does not have specific guidance for deciding on plant shutdowns and with the report's related recommendation identifying the need for NRC to develop specific guidance and a well-defined process for deciding when to shut down a nuclear power plant. We believe our regulations, guidance, and processes that cover whether and when to shut down a plant are robust and do, in fact, provide sufficient guidance in the vast majority of situations. Plant technical specifications, as well as many other NRC requirements and processes, provide a spectrum of conditions under which plant shutdown would be required. Plants have shut down numerous times in the past in accordance with NRC requirements. From time to time, however, a unique situation may present itself wherein sufficient information may not exist or the information available may not be sufficiently clear to apply existing rules and regulations definitively. In these unique instances, the NRC's most senior managers, after consultation with staff experts and given all of the information available at the time, will decide whether or not to require a plant shutdown. Risk information is used in accordance with Regulatory Guide 1.174. This process considers deterministic factors as well as probabilistic factors (i.e., risk information). We regard the combined use of deterministic and probabilistic factors to be a strength of our decision-making process.

Another issue identified in the draft report as a systemic weakness is that the NRC has not proposed specific actions to address a licensee's commitment to safety, also known as safety culture. We disagree with the report's recommendation that NRC should develop a methodology to assess licensees' safety culture that includes indicators of and/or information on patterns of licensee behavior, as well as on licensee organizational structures and processes. To date, the Commission has specifically decided not to conduct direct evaluations or inspections of safety culture as a routine part of assessing licensee performance due to the subjective nature of such evaluations. As regulators, we are not charged with managing our licensees' facilities. Direct involvement with safety culture, organizational structure, and processes crosses over to a management function. The NRC does conduct a number of assessments that adequately evaluate how effectively licensees are managing safety. These include an inspection procedure for assessing licensees' employee concerns programs, the NRC allegation program, enforcement of employee protection regulations, and safety-conscious work environment assessments during problem identification and resolution (PI&R) inspections. In addition, the NRC's LLTF made several recommendations (which are being addressed) to enhance the NRC's capability in this area. The NRC does not assess, nor does it plan to assess, licensee management competence, capability, or optimal organizational structure as part of safety culture.

While there are a number of factual errors in the draft report, as noted in the enclosure, we agree with many of the findings in the draft report. Most of GAO's findings are similar to the findings of the NRC's Davis-Besse LLTF. The NRC staff has made significant progress in implementing actions recommended by the LLTF and expects to complete implementation of more than 70 percent of them, on a prioritized basis, by the end of calendar year 2004. Reports tracking the status of these actions are provided to the Commission semiannually and will continue until all items are completed, at which time a final summary report will be issued.

I have enclosed the NRC's detailed comments on the draft report. If you have any questions, please contact Stacey L. Rosenberg, of my staff, at (301) 415-3868.

Sincerely,

(RA Luis A. Reyes for)

William D. Travers
Executive Director
for Operations

Enclosures:
1. NRC Comments on GAO Draft Report on Davis-Besse
2. Memorandum from EDO to OIG dated April 19, 2004

NRC Comments on Draft Report, GAO-04-415

1. The draft report does not speak to a key issue, the responsibility of licensees to provide complete and accurate information to the NRC. In carrying out its safety responsibilities, NRC must rely heavily on our licensees to provide us with complete and accurate information. Title 10 of the Code of Federal Regulations Section 50.9 requires that information provided to the NRC by a licensee be complete and accurate in all material respects. By not recognizing this explicitly and its role in this matter, the draft report conveys the expectation that the NRC staff should have known about the thick layer of boron on the reactor vessel head. The Davis-Besse Lessons Learned Task Force (LLTF), which NRC formed to ensure that lessons from the Davis-Besse experience are learned and appropriately captured in the NRC's formal processes, noted that the information that FirstEnergy provided in response to Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles" was inconsistent with information identified by the task force. Further, the LLTF report stated that had this information been known in the fall of 2001, the NRC may have identified the vessel head penetration (VHP) nozzle leaks and reactor pressure vessel (RPV) head degradation a few months sooner than the March 2002 discovery by the licensee. See also the related information in response #2.

2. Page 7, first sentence of the last paragraph states: ***"NRC should have but did not identify or prevent the vessel head corrosion at Davis-Besse because both its inspections at the plant and its assessments of the operator's performance yielded inaccurate and incomplete information on plant safety conditions."***

   Response: This statement is misleading. We agree that our oversight program should have identified certain evolving plant conditions for regulatory follow-up. This was also identified in the report of the Davis-Besse Lessons LLTF. It is the responsibility of licensees to provide the NRC with complete and accurate information. In fact, Title 10 of the Code of Federal Regulations Section 50.9 requires that information provided to the NRC by a licensee be complete and accurate in all material respects. The report should clearly indicate that NRC's licensees are responsible for providing us with accurate and complete information. While the NRC's Davis-Besse LLTF concluded that the NRC, the Davis-Besse licensee (FirstEnergy), and the nuclear industry failed to adequately review, assess, and follow up on relevant operating experience, the LLTF also noted that the information that FirstEnergy provided in response to Bulletin 2001-01 was inconsistent with information identified by the task force. Further, the LLTF report stated that had this information been known in the fall of 2001, the NRC may have identified the vessel head penetration nozzle leaks and the reactor vessel head degradation a few months sooner than the March 2002 discovery by the licensee. As you are aware, there is an ongoing investigation by the Department of Justice regarding the completeness and accuracy of information that FirstEnergy provided to the NRC on the condition of Davis-Besse.

3. Page 8, last sentence states: ***"Further, the risk estimate indicated that the likelihood of an accident occurring at Davis-Besse was greater than the level of risk generally accepted as being reasonable by NRC."***

Response: This is incorrect. NRC staff explained to the GAO consultants that NRC guidance produces an estimate for the change in core damage frequency of $5x10^{-6}$ per year, not $5x10^{-5}$ as indicated in the GAO report. According to Regulatory Guide (RG) 1.174, for Davis-Besse, this estimate is within acceptable bounds. NRC specifically documented the acceptability of the estimate in the December 2002 assessment. Thus, the December 3, 2002, safety evaluation concluded that the delta core damage frequency was consistent with the guidelines of RG 1.174.

4. Page 15 states that borax (i.e., sodium borate) is dissolved in the water. This is incorrect. Please replace the word "borax" with "boric acid crystals."

5. Page 18, first full paragraph states: ***"NRC, in deciding on when FirstEnergy had to shutdown Davis-Besse for the inspection,..."***

   Response: In addition, the staff relied upon information provided by the licensee regarding the condition of the vessel head (i.e., previous leakage and action taken to repair leaks and clean the vessel head).

6. Page 26, beginning on line 4, states: ***"According to the NRC regional branch chief—who supervised the staff responsible for overseeing FirstEnergy's vessel head inspection activities during the 2000 refueling outage—he was unaware of the boric acid leakage issues at Davis-Besse, including its effects on the containment air coolers and the radiation monitor filters."***

   Response: According to the individual to whom this statement is attributed, the statement would be correct if the phrase, "he was unaware...filters" is changed to "he was unaware that boric acid was found on the reactor vessel head during the outage."

7. Page 27, first sentence states: ***"Similarly, NRC officials said that NRC headquarters had no systematic process for communicating information in a timely manner to its regions or on-site inspectors."***

   Response: If the "information" in question refers to issues of potential safety significance into which inspectors should look, then this statement is inaccurate. The systematic process for temporarily focusing inspection activity in a coordinated program-wide manner on high-priority issues is the "Temporary Instruction" (TI) process, which is well established within the NRC Inspection Manual and frequently used. The legitimate point to be made is that until the Davis-Besse event, the NRC had not concluded that boric acid corrosion was a sufficient safety concern that reached the threshold for using the TI process.

8. Page 33, middle paragraph states: ***"For example, concern over alloy 600 cracking led France, as a preventive measure, to develop plans for replacing all of its reactor vessel heads and installing removable insulation to better inspect for cracking."***
Response: French regulators instituted requirements for an extensive, non-visual nondestructive examination inspection program for vessel head penetration nozzles that resulted in plant operators deciding, on the basis of economic considerations, to replace vessel heads in lieu of conducting such examinations.

9. Page 34, last paragraph states: ***"If such small leakage can result in such extensive corrosion..."***
Response: Small leakage alone was not the cause of the corrosion. It was a combination of prolonged leakage in conjunction with allowing caked-on boron to remain on the vessel head.

10. Page 36, middle paragraph states: ***"However, NRC decided that it could not order Davis-Besse to shut down on the basis of other plants' cracked nozzles and identified leakage or the manager's acknowledgment of a probable leak. Instead, it believed it needed more direct, or absolute, proof of a leak to order a shutdown."***
Response: As discussed at the NRC-GAO exit conference, plant Technical Specifications, as well as many other NRC requirements and processes, provide a number of circumstances in which a plant shutdown would or could be required, including the existence of reactor coolant pressure boundary leakage while operating at power.

Please note that there was no legal objections to the draft order and the stated basis for deciding to not issue the order was not an insufficient legal basis.

11. Page 36, last paragraph states: ***"...NRC does not have specific guidance for shutting down a plant when the plant may pose a risk to public health and safety even though it may be complying with NRC requirements."***

Response: We disagree with this finding and with the report's related recommendation on Page 63 identifying the need for NRC to develop specific guidance and a well-defined process for deciding when to shut down a nuclear power plant. We believe our regulations, guidance, and processes that cover whether and when to shut down a plant are robust and do, in fact, provide sufficient guidance in the vast majority of situations. Plant technical specifications, as well as many other NRC requirements and processes, provide a spectrum of conditions under which plant shutdown would be required. Plants have shut down numerous times in the past in accordance with NRC requirements. From time to time, however, a unique situation may present itself wherein sufficient information may not exist or the information available may not be sufficiently clear to apply existing rules and regulations definitively. In these unique instances, the NRC's most senior managers, after consultation with staff experts and given all of the information available at the time, will decide whether or not to require a plant shutdown. Risk information is used in accordance with RG 1.174. This process considers deterministic factors as well as probabilistic factors (i.e., risk information). We regard the combined use of deterministic and probabilistic factors to be a strength of our decisionmaking process.

12. Page 38, third paragraph states: ***"At some point during this time, NRC staff also concluded that the first safety principle was probably not being met, although the basis for this conclusion is not known."***

Response: The report should clarify GAO's basis for this statement. NRC staff believed that the regulations were met.

13. Page 40, last paragraph states: ***"However, NRC did not provide the assessment until a full year later—in December 2002. In addition, the December 2002 assessment, which includes a 4-page evaluation, does not fully explain how the safety principles were used or met—other than by stating that if the likelihood of nozzle failure were judged to be small, then adequate protection would be ensured."***

    Response: The attachment to the December 3, 2002, letter is an 8-page evaluation, not 4 pages. We note this to make sure GAO is referring to the same document. The assessment addresses four of the five safety principles. In the NRC's December 2002 safety evaluation, the staff stated that the criterion related to compliance with the regulations was being met because the inspections performed by the licensee were in conformance with the ASME Code. In addition, the safety evaluation stated that Davis-Besse met the criterion related to defense-in-depth because all three barriers against release of radiation were intact and reliable; they met the margin criterion because even the largest circumferential cracks found in pressurized-water reactors had considerable margin to structural failure, and they met the low-risk impact criterion based on a comparison of delta core damage frequency estimates with the guidelines of RG 1.174. The fifth safety principle, requiring a monitoring program, was not relevant to a decision that lasted only 6 weeks.

14. Page 42, first paragraph states: ***"Multiplying these two numbers, NRC estimated that the potential for a nozzle to crack and cause a loss-of-coolant accident would increase the frequency of core damage at Davis-Besse by about $5.4x10^{-5}$ per year, or about 1 in 18,500 per year. Converting this frequency to a probability, NRC***

calculated that the increase in probability of core damage was approximately $5.0x10^{-6}$, or 1 chance in 200,000. While NRC officials currently disagree that this was the number it used, this is the number that it included in its December 2002 assessment provided to FirstEnergy. Further, we found no evidence in the agency's records to support NRC's current assertion."

Response: These statements mischaracterize the facts. NRC estimated that the probability of nozzle cracking leading to a loss-of-coolant accident during the first 6 weeks in 2002 would increase the annual core damage frequency (CDF) by about $5.4x10^{-6}$ per year, or about 1 in 185,000 per year. The estimate of $5x10^{-5}$ was an intermediate step in our calculation. The estimate of $5x10^{-5}$ represents the change in CDF if Davis-Besse were allowed to operate for one year without shutting down for inspection of the vessel head. Allowing Davis-Besse to continue to operate for one year was never a consideration. Thus, multiplying by the fraction of time in one year under consideration (in this case 7 weeks) was the final step in the calculation of delta CDF. The confusion about the estimate NRC used in the decisionmaking process may be due to NRC's method of calculating delta CDF for plant conditions which do not persist for the entire year. If this final step (the fraction of the year the plant is allowed to operate) were not part of the calculation, then the risk estimate of allowing the licensee to continue to operate for 7 weeks, as compared to one year, would be the same. Logically, this does not make sense. Therefore, the estimate of $5x10^{-5}$ does not automatically convert to a probability, as GAO's statement implies. Because the period of operation under consideration was approximately 0.13 years, the annual average change in CDF was about $5x10^{-6}$ per year, and the increase in the probability of core damage was about $5x10^{-6}$ as well. NRC officials **agree** that $5x10^{-6}$ was the estimate used in the decisionmaking process and is the estimate provided in the December 2002 assessment.

15. Page 42, second paragraph states: ***"For example, the consultants concluded that NRC's estimate of risk was incorrectly too small, primarily because the calculation did not consider corrosion of the vessel head."***

Response: An underlying assumption in any risk assessment is that you have complete and accurate information from the licensee. NRC staff was of the understanding that efforts had been made to remove boric acid accumulation from the vessel head during previous outages. For all six B&W plants that found signs of penetration leakage, the leakage manifested itself in the form of small amounts of dry boron crystals on the vessel head, which are not corrosive, and did not produce any corrosion on the vessel heads of these six B&W plants. Boron leaking onto a clean vessel head does not cause corrosion. Therefore, corrosion this extensive was not anticipated at the time. Also, it is important to note that had Davis-Besse shut down on December 31, 2001, the same corrosion would have been found.

16. Page 43, first full paragraph discusses the experience at French nuclear power plants.

Response: The NRC staff was aware of the issue as illustrated in an internal memorandum dated December 15, 1994, from Brian Grimes to Charles Rossi.

17. Page 44, first full paragraph states: ***"Third, NRC's analysis was inadequate because the risk estimates were higher than generally considered acceptable under NRC guidance. Despite PRA's [probabilistic risk assessment's] important role in the decision, our consultants found that NRC did not follow its guidance for ensuring that the estimated risk was within levels acceptable to the agency.*** Page 45, first paragraph states: ***"...NRC's PRA estimate for Davis-Besse resulted in an increase in the frequency of core damage of $5.4x10^{-5}$ or 1 chance in about 18,500 per year was higher than the acceptable level."***

Response: This conclusion is not supported by the facts and it is misleading. The estimate referenced by GAO is an intermediate calculation in our process, and was not used, and should not be used, in the decisionmaking process. NRC staff explained to the GAO consultants that NRC guidance produces an estimate for the change in core damage frequency of $5x10^{-6}$ per year, not $5x10^{-5}$ as indicated in the GAO report. According to RG 1.174, for Davis-Besse, this estimate is within acceptable bounds. NRC specifically documented the acceptability of the estimate in the December 2002 assessment. Thus, the December 3, 2002, safety evaluation concluded that the delta CDF was consistent with the guidelines of RG 1.174.

18. Page 45, first paragraph states: ***"NRC's guidance for evaluating requests to relax NRC technical specifications suggests that a probability increase higher than $5x10^{-7}$ or 1 chance in 2 million is considered unacceptable for relaxing the specifications. Thus, NRC's estimate would not be considered acceptable under this guidance."***

Response: This criterion in RG 1.177 is not relevant to the Davis-Besse decision. It is confined to decisions on allowed outage times (AOT) for equipment, and is defined to avoid very high instantaneous risks (CDF > 10^{-3}) for very short periods (5 hours).

19. Page 46, first full paragraph states: ***"Lastly, NRC's analysis was inadequate because the agency does not have clear guidance for how PRA estimates are to be used in the decision-making process."***

Response: The NRC's process for risk-informed decision-making is considerably more robust than characterized in this section. Regulatory Guide 1.174 comprises 40 pages of guidance on how to use risk in decisions of this type, and it is backed up by equally detailed guidance for specific types of decisions such as technical specifications, in-service inspection programs, in-service testing, and quality assurance. The NRC has amassed a great deal of experience in application of the guidance. Risk assessment is a tool to help better inform decisions that are based on engineering judgements.

20. Page 46, last paragraph states: ***"It is not clear how NRC staff used the PRA risk estimate in the Davis-Besse decision-making process."***

    Response: The December 3, 2002, safety evaluation clearly states how the PRA estimate was used in the decisionmaking process; the estimate was compared with the guidelines of RG 1.174. The safety evaluation also points out that NRC staff who are expert in non-PRA disciplines such as probabilistic fracture mechanics, gave more weight to deterministic factors, such as the structural margin that remains in the nozzles with circumferential cracks. The NRC considers the combined use of deterministic and probabilistic factors to be a strength of our decisionmaking process.

21. Page 48, last paragraph states: ***"...NRC had made progress in implementing the recommendations, although some completion dates have slipped."***

    Response: The schedules for implementation of all high priority recommendations have not slipped. The implementation schedule for certain low or medium priority recommendations slip only in accordance with NRC's Planning, Budgeting and Performance Management (PBPM) process, which explicitly considers safety significance when making budget priority decisions.

22. Page 51, top of page, first full bullet states: ***"One recommendation is directed at improving NRC's generic communications program. NRC is..."***

    Response: We recommend re-wording this as follows: "One recommendation is directed at improving follow up of licensee actions taken in response to NRC generic communications. A Temporary Instruction (Inspection Procedure) is currently being developed to assess the effectiveness of licensee actions taken in response to generic communications. Additionally, improvements in the verification of effectiveness of generic communications are planned as a long-term change in the operating experience program."

23. Page 51, last paragraph states: ***"...NRC's revised inspection guidance for more thorough examinations of reactor vessel heads and nozzles, as well as new requirements for NRC oversight of licensees' corrective action programs, will require at least an additional 200 hours of inspection per reactor per year."***
Response: It is unclear where this number comes from, but the changes to the corrective action program procedure require only about 16 hours per reactor year for the trend review.

24. Page 53, first paragraph discusses the NRC's Office of the Inspector General's (OIG's) findings on communications.
Response: The NRC's actions are not limited primarily to improving communication about boric acid corrosion and cracking. There are multiple task force recommendations, and other NRC initiatives, that are aimed at addressing the broader implications stemming from communication lapses noted by the task force and the OIG. For example, actions have been implemented to more effectively disseminate operating experience to end users, reenforce a questioning attitude in the inspection staff, and discuss Davis-Besse lessons learned at various forums.

NRC's initial response to the OIG did not directly address the broader actions we are taking to improve communications. Our response to the OIG only indirectly addressed this by discussing the operating experience program enhancements. Part of the enhancements to the operating experience program is the expectations for improved communications. In addition, communication improvement initiatives with internal and external stakeholders are in progress to address shortcomings in this critical area. Our revised response to the OIG on this issue, dated April 19, 2004, is provided as Enclosure 2.

25. Page 53, second paragraph states: ***"NRC's Davis-Besse task force did not make any recommendations to address two systemic problems: evaluating licensees' commitment to safety and improving the agency's process for deciding on a shutdown."***

Response: The LLTF did not make a recommendation for improving the agency's process for deciding on a shutdown. This area was not reviewed in detail by the task force because of coordination with the OIG. Moreover, the task force review efforts were focused on why the degradation cavity was not prevented. While related, the shutdown issue had little to do with the degradation cavity.

The task force made multiple recommendations aimed at enhancing NRC's capability to evaluate the licensees' commitment to safety, by indirect means. Refer to task force recommendations: 3.2.5(1), 3.2.5(2), 3.3.2(2), 3.3.4(5), and Appendix F.

26. Page 54, last paragraph states: ***"This problem identification and resolution inspection procedure is intended to assess the end-results of management's safety commitment rather than the commitment itself."***

Response: This statement is inaccurate. Regarding its accuracy, the PI&R inspection procedure (IP 71152) actually has six stated inspection objectives (refer to section 71152-01) including: (1) provide for early warning of potential performance issues that could result in crossing threshold in the action matrix and (2) to provide insights into whether licensees have established a safety-conscious work environment. Using this IP, inspectors seek factual evidence of the licensee's assumed commitment to safety (by reviewing their identification and correction of actual problems). Inspection issues routinely are raised with regard to a licensee's weakness in correcting recurrent problems or in adequately addressing issues that could become a future significant safety concern. The statement on Page 55 of the report, ***"Furthermore, because NRC directs its inspections at problems that it recognizes as being more important to safety, NRC may overlook other problems until they develop into significant and immediate safety problems"*** does not accurately reflect the stated objectives and demonstrable implementation of IP 71152.

27. Pages 55-56, discuss safety culture.

Response: To a significant degree, the areas referenced in this draft report are addressed either by NRC requirements or inspection activities. For example, the NRC has requirements limiting work hours for critical plant staff members such as security officers and plant operators. The NRC has requirements governing operator training. Inspectors routinely monitor various licensee meetings and job briefings to evaluate the licensee's emphasis on safety.

Moreover, the NRC has a number of other means to indirectly assess safety culture. Other NRC tools that provide indirect insights into licensee safety culture include:

- inspection procedure for assessing the licensee's employee concerns program,
- NRC's allegation program,
- enforcement of employee protection regulations,

- Safety-Conscious Work Environment (SCWE) assessments during problem identification and resolution inspections,
- lessons-learned reviews such as the one conducted for the Davis-Besse reactor pressure vessel head degradation; and
- Reactor Oversight Process cross-cutting issues of human performance, problem identification and resolution, and SCWE.

28. Page 58, paragraph under the first header states: ***"It recognized that NRC's written rationale for accepting FirstEnergy's justification for continued plant operation was not prepared until 1 year after its decision..."***

    Response: For clarification, the documentation of the decision about one year later was corrective action from a task force finding.

29. Page 58, paragraph under second header states: ***"The NRC task force did not address NRC's failure to learn from previous incidents at power plants and prevent their recurrence."***

    Response: This sentence is factually inaccurate. The task force performed a limited review of past lessons-learned reports and actually identified many more potentially recurring programmatic issues as a result of that review than the three examples cited by the GAO in this section of the draft report. As discussed during the NRC-GAO exit conference, the task force made a recommendation to perform a more detailed effectiveness review of the actions stemming from other past NRC lessons learned reviews (Appendix F). This review is currently in progress.

April 19, 2004

MEMORANDUM TO: Hubert T. Bell
Inspector General

FROM: William D. Travers ***/RA Carl J. Paperiello Acting For/***
Executive Director for Operations

SUBJECT: FEBRUARY 2, 2004, OFFICE OF INSPECTOR GENERAL (OIG) MEMORANDUM CONCERNING AGENCY RESPONSE TO OIG EVENT INQUIRY CASE NO. 03-02S (NRC'S OVERSIGHT OF DAVIS-BESSE BORIC ACID LEAKAGE AND CORROSION DURING THE APRIL 2000 REFUELING OUTAGE)

This memorandum responds to your memorandum to Chairman Diaz, dated February 2, 2004, concerning the Nuclear Regulatory Commission (NRC) staff's response of January 12, 2004, to OIG Event Inquiry 03-02S. The referenced OIG event inquiry was initiated in response to a Congressional request that OIG determine how the NRC staff handled Davis-Besse Condition Report (CR) 2000-0782 at the time of discovery in refueling outage (RFO) 12 (2000) and whether the CR was considered in the November 2001 decision to allow Davis-Besse to continue to operate to February 16, 2002. The NRC staff's previous response to OIG (January 12, 2004) regarding this issue provided a matrix of those recommendations from the Davis-Besse Lessons Learned Task Force (DBLLTF) report that specifically addressed the event inquiry findings and referenced the report for a complete picture of the staff's efforts. The OIG response of February 2, 2004, stated that the NRC staff had not addressed the problem of communications as an underlying cause of the findings of the OIG event inquiry and that the agency should include an expectation of improved communication between and among NRC Headquarters and regional staff and should outline specific guidance to achieve this goal. In addition, OIG specifically concluded that "had the [Davis-Besse Nuclear Power Station] DBNPS inspectors been better informed of ongoing NRC industry-wide efforts to address coolant pressure boundary leakage and the effects of boric acid corrosion, they would have recognized the significance of Condition Report 2000-0782 and highlighted the information to regional management."

The DBLLTF report discusses the NRC's and industry's failure to understand the significance of boric acid corrosion of the reactor vessel head. The NRC staff believes that this failure caused the underlying communications lapses. Although the potential for this type of degradation existed previously, the significance of boric acid deposits was not understood by the staff. The assumption throughout NRC was that the boric acid deposits would be in a dry, powder-like form that could easily be removed and would not accumulate in a condition that would be corrosive to the reactor vessel head. As identified in the event inquiry, the inspectors did communicate a substantial amount of information to the region and the NRR Project Manager, particularly regarding the fouling of the containment air coolers and radiation monitor filter

Contact: Edwin M. Hackett, NRR/DLPM/PDII
415-1485

elements; however, the significance of this information was also not appreciated at the time. This same failure to understand the significance of the situation was the cause of the lack of communication from Headquarters to the regions. Several elements of the matrixed DBLLTF Action Plans address this underlying issue of lack of recognition of the significance of the evidence. The desired outcome for these actions is for all NRC staff to maintain a questioning attitude and lower thresholds for communications concerning materials degradation corrosion.

More broadly, the NRC staff agrees that communications are of critical importance in all aspects of NRC activities and particularly important as an underlying cause for issues discovered at DBNPS. The corrective actions outlined in the DBLLTF Action Plans address communications beyond the topic of boric acid corrosion control. For example, corrective actions in the area of operating experience development and use are focused on enhancing communications. The recommendations to strengthen inspection guidance, institute training to reinforce a questioning attitude on the part of management and staff, and change the Inspection Manual to provide guidance for the staff to pursue issues identified during plant status reviews are intended to establish more definitive expectations for improved communications of operating experience. As discussed in the February 23, 2004, semiannual update report and at the February 26, 2004, Commission meeting, implementation plans for this area are still under development and may significantly influence the way the agency does business in the future. Developing the most effective and efficient communications channels will be key to the successful implementation of an effective operating experience program.

Beyond the DBLLTF Action Plan, the agency has several ongoing initiatives that provide examples of efforts to more broadly improve intra-agency communications. These examples include establishment of a Communication Council reporting to the Executive Director for Operations and the creation of a communications specialist position reporting to the Office of Nuclear Reactor Regulation (NRR) Associate Director for Inspections and Programs. NRR also continues to improve and enhance its Web site as a focused means of communicating with both internal and external stakeholders. From a regional perspective, examples of communication enhancements include lowering the threshold for communication of plant issues on morning status calls, devoting additional time to discussing lessons learned from plant events and inspection findings during counterpart meetings, and developing enhanced guidance for documenting significant operational event followup decisions. Collectively, these examples provide a strong indication that NRC Headquarters and regional staff have begun to internalize two of the most important lessons from the Davis-Besse event. These are that on occasion, information initially considered to have low significance by the first NRC recipient is later found to be of greater significance once the information is shared and evaluated more collegially; and with regard to the complex nature of commercial nuclear power operations, no one person can be aware of all aspects of an issue. As a result, the more information that is shared, the more likely significant problems will be identified and appropriate action(s) taken.

In summary, the NRC staff recognizes that communication failures were an underlying cause of the agency's problems concerning the delayed discovery of the boric acid corrosion at DBNPS. Our January 12, 2004, response to the event inquiry specifically addressed what we considered to be the root cause of the event-specific communication failures, namely that the entire staff did not recognize the potential significance of boric acid corrosion. Expectations for improved communications will be developed as an integral part of our operating experience program enhancements. More broadly, communication improvement initiatives with internal and external stakeholders are in progress to enhance agency performance in this critical area of our responsibilities. We regret that our initial response did not clearly address the broader actions we are taking to improve communications and appreciate the opportunity to clarify our response.

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
SECY
LReyes

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

CHAIRMAN

May 4, 2004

Mr. Michael H. Dworkin, Chairman
Vermont Public Service Board
112 State Street, Drawer 20
Montpelier, Vermont 05620-2701

Dear Mr. Dworkin:

I am responding on behalf of the U.S. Nuclear Regulatory Commission (NRC) to your letters dated March 15 and 31, 2004, regarding the request by Entergy Nuclear Vermont Yankee, LLC, and Entergy Nuclear Operations, Inc. (Entergy), to amend the Vermont Yankee Nuclear Power Station license to increase the power level of the facility. In those letters, the Vermont Public Service Board requested that the NRC conduct its review of the proposed power uprate in a way that would provide Vermont a level of assurance about plant reliability equivalent to an independent engineering assessment. The NRC has decided to conduct a detailed engineering inspection that we believe will be appropriate for addressing our oversight responsibilities and is also responsive to the Board's concerns. This inspection will be performed as part of a new engineering inspection program that the NRC has been developing to enhance the Reactor Oversight Process.

NRC regulations and its oversight process focus on ensuring nuclear safety, whether the facility is operating at power or shut down. The NRC's statutory authority does not extend to regulating the reliability of electrical generation. The NRC recognizes, however, that there is some overlap between attributes that result in safe operation and those that contribute to overall plant reliability.

The Commission understands that the Board is concerned about the reliability of Vermont Yankee following an increase in power level, especially in light of operational issues that have occurred at some other plants that have recently implemented extended power uprates. The NRC recognizes the importance of these issues and is taking steps to ensure that they are satisfactorily addressed to maintain safety. For example, in response to instances of steam dryer cracking at some boiling water reactors, outside technical experts are assisting NRC staff in performing an audit of General Electric's analyses related to steam dryer performance and specific issues related to Vermont Yankee. We continue to engage the industry to ensure resolution of these issues and will consider additional regulatory action, if needed.

The NRC's established review process for power uprate applications is independent, thorough, and comprehensive. A description of the review process is enclosed. Engineering assessments have always been an integral part of the NRC's safety activities. Under our current Reactor Oversight Process, NRC resident inspectors and regional specialists routinely evaluate the work performed by the licensee's engineering organization to determine whether engineering analyses adequately support safe operation. Over the past several months, the NRC has been developing a new engineering inspection program which we intend to pilot at selected plants. The NRC staff considered a number of factors, including the Board's request for an independent engineering assessment, and concluded it is appropriate to conduct this engineering inspection at Vermont Yankee. This new engineering assessment inspection incorporates the best practices of the existing and past engineering inspections. The NRC will use this inspection to verify that design bases have been correctly implemented for a sampling of components across multiple systems and to identify latent design issues. The inspection process uses operating experience, risk assessment, and engineering analysis to select risk-significant components and operator actions, and will ensure that adequate safety margins exist. Although the specific sampling of components is still being developed, it will include components from multiple systems that are potentially affected by a power uprate such as the emergency core cooling systems, the containment system, power conversion systems, and auxiliary systems. The inspection will be performed by a team of approximately six inspectors, including some NRC inspectors who do not have recent oversight experience with Vermont Yankee and at least two contractors with design experience. Three weeks of on-site inspection and over 700 hours of direct inspection time will be conducted. This level of effort exceeds that of the biennial safety system design inspection. The Commission believes it is appropriate for addressing the NRC's oversight responsibilities and is also responsive to the Board's concerns. The NRC staff will inform the State of Vermont of the schedule for this inspection to facilitate participation by State representatives, consistent with NRC policy.

The NRC Advisory Committee on Reactor Safeguards (ACRS) will also review the Vermont Yankee power uprate request. The ACRS is a statutory committee that reports directly to the Commission and is structured to provide a forum where experts representing many technical perspectives can provide advice that is factored into the NRC's decision-making process. The NRC staff will provide the results of its review efforts, including relevant inspection findings, to the ACRS for review. After the ACRS completes its review, it will make an independent recommendation regarding whether the proposed power uprate amendment should be approved.

The NRC will not approve the Vermont Yankee uprate, or any proposed change to a plant license, unless the NRC staff can conclude that the proposed change will be executed in a manner that assures the public's health and safety. In response to your request, the NRC staff has taken a close look at proposed inspections and technical reviews to ensure that they will identify and address potential safety concerns for operating at uprated power conditions. The staff has concluded that the detailed technical review, prescribed in the Extended Power Uprate Review Standard, coupled with the normal associated program of power uprate and engineering inspections, will provide the information necessary for the NRC staff to make a decision on the safety of operation of Vermont Yankee under uprated power conditions. The Commission believes that the results of NRC reviews and inspections, particularly the new engineering inspection, will assist in addressing the Board's concerns regarding the future reliability of Vermont Yankee. The NRC staff is prepared to meet with the Board to explain further our review process and scope, including the engineering assessment inspection.

Sincerely,

/RA/

Nils J. Diaz

Enclosure:
Established NRC Power Uprate Review Process

Established NRC Power Uprate Review Process

The NRC's established review process for power uprate applications is independent, thorough, and comprehensive. A team of engineers with specialties in a minimum of 17 different technical areas will review the Vermont Yankee power uprate application. The NRC plans to expend about 4000 hours to perform a comprehensive assessment of the engineering, design, and safety analyses related to the uprate. The NRC's "Review Standard for Extended Power Uprates" guides the staff in its review of the application. The Review Standard also provides guidance for determining when and what type of audits should be performed at the plant or vendor sites, as well as for performing our own confirmatory analyses and independent calculations to supplement the review.

The NRC's review of the power uprate application also includes on-site inspections. NRC inspections will review selected activities and modifications made to allow operation at higher power levels to verify that changes to plant systems will support safe plant operation and are in accordance with Vermont Yankee's licensing and design bases. The NRC will use Inspection Procedure 71004, "Power Uprates," as well as a number of our baseline inspection procedures to inspect issues specifically related to power uprate. These inspections will assess changes that could impact the integrity of barriers (e.g., higher flow rates which could increase vibration at specific support points), safety evaluations, plant modifications, post maintenance and surveillance testing, heat exchanger performance, and integrated plant operation. Additionally, our other baseline inspection activities, while not specifically directed at power uprate activities, will provide additional information about Vermont Yankee's ability to operate safely at a higher power level.

The NRC will adjust, as necessary, our technical review, audit plans, confirmatory analyses, or inspection activities if any issues are identified which may have a bearing on our decision on the Vermont Yankee power uprate application. For example, a recent examination of the steam dryer at Vermont Yankee identified cracks on both interior and exterior structures of the steam dryer. The steam dryer is an important component in the process for converting steam to electrical energy, but is not used to mitigate any accidents. The NRC is interested in steam dryer cracking because of the potential for parts to break loose and impact the performance of safety-related equipment. Entergy has indicated that the cracks are in low-stress, low-steam flow areas of the dryer and not in the areas where cracks were observed at other plants that implemented extended power uprates. NRC inspectors monitored Entergy's steam dryer inspection activities, and we will thoroughly review Entergy's follow-up actions as part of our evaluation of Vermont Yankee's request to operate at a higher power level.

Assessment of engineering has always been an integral part of the NRC's safety mission. In the 1990s, the NRC performed extensive reviews at plants across the country to determine if licensees were operating plants in accordance with their design bases. As part of this review, two team inspections were conducted at Vermont Yankee in 1997. One of these inspections was led by staff from NRC headquarters and included six contractors. In 1998, the NRC conducted an engineering inspection, as well as a team inspection to address operability issues resulting from Vermont Yankee's configuration improvement program. Under our current Reactor Oversight Process, NRC resident inspectors and regional specialists routinely evaluate the work performed by the licensee's engineering organization to determine whether the engineering analyses adequately supports safe operation. Our inspectors conduct both routine engineering inspections, as well as an in-depth team inspection every two years. Since the Reactor Oversight Process was implemented in 2000, the NRC has conducted two such safety system design team inspections.

Enclosure

STATEMENT SUBMITTED

BY THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

TO THE

SUBCOMMITTEE ON CLEAN AIR, CLIMATE CHANGE,

AND NUCLEAR SAFETY

COMMITTEE ON ENVIRONMENT AND PUBLIC WORKS

UNITED STATES SENATE

FOR THE

OVERSIGHT HEARING

SUBMITTED BY

DR. NILS J. DIAZ

CHAIRMAN

SUBMITTED: MAY 20, 2004

Introduction

Mr. Chairman and members of the Subcommittee, it is a pleasure to appear before you today with my fellow Commissioners to discuss the Nuclear Regulatory Commission's programs. We appreciate the past support that we have received from the Subcommittee and the Committee as a whole, and we look forward to continue working with you.

As you know, the NRC's mission is to license and regulate the Nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of public health and safety, promote the common defense and security, and protect the environment. The Commission does not have a promotional role -- rather, the agency seeks to ensure the safe application of nuclear technology if society elects to pursue the nuclear energy option. The Commission recognizes, however, that its regulatory system should not establish inappropriate impediments to the application of nuclear technology. Many of the Commission's initiatives over the past several years have focused on maintaining or enhancing safety and security while simultaneously improving the effectiveness and efficiency of our regulatory system.

With your permission Mr. Chairman, I will highlight a few of our ongoing initiatives and achievements.

Reactor Safety Programs

The past three years have seen the maturing of the reactor oversight process. We believe that this program is a significant improvement over the former inspection, enforcement, and assessment processes. We received external recognition of the effectiveness of our

Reactor Inspection and Performance Assessment program when the Office of Management and Budget evaluated it using its Performance Assessment Rating Tool (PART) and awarded the top rating, "effective," a rating achieved by only 11% of the Federal programs assessed. One of its strongest attributes is its transparency and accessibility to members of the public. You will find performance indicators and inspection findings for every power reactor on NRC's public web site page, as well as our current assessment of each reactor's overall performance. The transition to the reactor oversight process has gone well, and we will strive to make further improvements.

Overall, the industry has performed well. As of the end of CY 2003, there were two plants designated for the highest level of scrutiny under the reactor oversight process, the Cooper plant in Nebraska and the Point Beach plant in Wisconsin. In addition, the Davis-Besse plant in Ohio has been treated under our Manual Chapter 0350 Startup Oversight Process. The Cooper and Point Beach plants have received significant attention from our regional and headquarters offices, and we are confident that these plants are on a path to resolving long-standing problems.

Over the past two years, the NRC staff has devoted significant resources for enhanced regulatory oversight of the Davis-Besse plant following the discovery of extensive degradation of the reactor vessel head. After an extgensive plant recovery program and comprehensive corrective actions by the licensee, FirstEnergy, and considerable NRC inspection and assessment, the staff determined that there was reasonable assurance that the plant could be safely restarted and operated. This decision was made in a deliberate manner, based on sound regulatory and technical findings, and in accordance with the requirements of Federal statutes and NRC regulations. On March 8, 2004, the NRC staff gave approval for the restart of Davis-Besse. In addition, the staff issued a confirmatory Order requiring independent assessments and inspections at Davis-Besse to assure that long-term corrective actions remain effective. The NRC's oversight panel will continue to coordinate the inspection and regulatory activities for Davis-Besse until plant performance warrants resumption of the normal reactor oversight process.

We acknowledge the extensive interest in, and concerns about, the restart of Davis-Besse by area residents; public interest groups; Federal, State, and local officials; and others. We have conducted our regulatory responsibilities in an open and candid manner, keeping the public informed to the maximum extent possible at each step of the process. We have not been able to share the results of our Office of Investigations' reports because those have been referred to the Department of Justice for its consideration. Those reports have, however, been fully considered by NRC staff prior to restart. We have had extensive communication with our stakeholders, including establishing a web site and issuing monthly newsletters. Also during the past two years, the NRC staff conducted 75 public meetings on Davis-Besse – most of these meetings were held in the vicinity of the plant – and held 50 briefings for Federal, State, and local government officials. The oversight panel will continue to hold periodic public meetings near Davis-Besse with FirstEnergy officials to review the status of ongoing activities at the plant.

Concurrently, we have undertaken a significant and critical review of our programmatic and oversight activities to evaluate our own actions associated with the reactor vessel head degradation at Davis-Besse. These actions have considered the Davis-Besse Lessons

Learned Task Force Report. The Task Force completed its review in September 2002 and issued a report that contained a number of recommendations for improvements to the reactor research, oversight, and licensing programs. These recommendations are being implemented as part of four action plans, encompassing: (1) stress corrosion cracking, (2) operating experience program effectiveness, (3) inspection, assessment and project management guidance, and (4) barrier integrity requirements. Of the 49 recommendations, 16 were completed in 2003, including all seven high priority items scheduled to be completed that year. Inspection program guidance was revised to address the high-priority recommendations regarding follow-up to long-standing equipment issues and oversight of plants in extended shutdowns. Enhancements to inspector training programs were initiated. Guidance was issued regarding the adequate documentation of certain decisions. We continue to work on addressing the remaining recommendations and are making significant progress. Except for three items, all other high-priority recommendations will be completed by the end of 2004. The remaining high priority items will be completed during 2005.

In April 2004, we completed an examination of reactor vessel cladding and structural analyses. Based on these efforts, the staff concluded that near-term vessel failure was unlikely and that it was highly likely the vessel could have operated safely for at least several more months following the February 2002 Davis-Besse shutdown. As you are aware, the plant restarted with a new reactor vessel head; thus, the degraded condition no longer exists.

The NRC's Office of the Inspector General conducted an inquiry into our oversight of the Davis-Besse reactor vessel head degradation. The issues identified in the IG's report are similar to a subset of those identified by the Lessons Learned Task Force; and as such, corrective actions have either been completed or are in progress for each of the IG's findings. The IG was particularly concerned with the flow of information within the agency -- communication between headquarters, the regional offices, and the resident inspector staff. We are committed to improving this communication and have already witnessed a lowering threshold for raising issues. For example, there has been a significant increase in the scope and level of detail discussed during daily status meetings among NRC regional, headquarters, and site offices, as well as improvements in internal communications. We have also placed renewed emphasis on improving communication with the international nuclear community to ensure that new issues are promptly communicated as they arise. Going forward, we are dedicated to improving our inspection and assessment programs to prevent recurrence of this or similar significant challenges to safety.

Reactor Licensing Programs

Let me now turn to significant achievements in our reactor licensing programs. The reactor licensing program ensures that operating nuclear power plants maintain adequate protection of public health and safety throughout the plant's operating life. NRC licensing activities include reviewing license applications and changes to existing licenses, reviewing reactor events for safety significance, and improving safety regulations and guidance. In FY 2003, the NRC met or exceeded all established measures for the timeliness and quantity of completed nuclear power plant licensing-related actions.

The reactor licensing program's timeliness in responding to licensee requests has improved dramatically since 1997. At the end of FY 2003, 96 percent of licensing actions in the working inventory were less than one year old and 100 percent of licensing actions in the working inventory were less than two years old. We also completed 500 other licensing activities, most of which were associated with identification and resolution of emerging technical issues. For example, we issued generic communications to the industry alerting them to emerging issues such as leakage from reactor pressure vessel lower head penetrations, the potential impact of debris blockage on emergency sump recirculation at pressurized-water reactors, and control room habitability. We will not be able to sustain this level of timeliness in FY 2004 because of a very large volume of security licensing actions which we are giving the highest priority. We are managing our licensing action inventory to ensure that appropriate timeliness goals are being established for each action, and that no safety-significant issue is left untreated.

A significant type of reactor licensing action, called a power uprate, is a request to raise the maximum power level at which a plant may be operated. Improvement of instrument accuracy and plant hardware modifications have allowed licensees to submit power uprate applications for NRC review and approval. The focus of our review of these applications has been and will continue to be on safety. In addition, we continue to monitor operating experience closely to identify issues that may affect power uprate implementation.

Power uprates range from requests for small increases of less than two percent based on the recapture of power measurement uncertainty, to large increases in the range of 15 to 20 percent that require substantial hardware modifications to the plants. In all instances, the NRC must be satisfied that appropriate safety margins remain. To date, the NRC has approved 101 power uprates which have safely added approximately 4175 megawatts electric to the nation's electric generating capacity and is the equivalent of about four large nuclear power plants.

Currently, the NRC has four power uprate applications under review and expects to receive an additional 25 applications through calendar year 2005. This would add approximately 1760 megawatts electric to the nation's electric generating capacity. The NRC recently issued a Review Standard for Extended Power Uprates (i.e., uprates that increase the current power by 7 percent or more), which is available publicly, that enhances the NRC's focus on safety and improves consistency, predictability, and efficiency of these reviews.

As stated earlier, the NRC monitors operating experience at plants that have implemented power uprates. Cases of steam dryer cracking and flow-induced vibration damage affecting components and supports for the main steam and feedwater lines have been observed at some of these plants. We conducted inspections to identify the causes of several of these issues and evaluated many of the repairs performed by the licensees. We continue to monitor the industry's generic response to these issues and will consider additional regulatory action, as appropriate.

License renewals are another significant type of licensing action. In 2003, thirteen units -- North Anna Units 1 and 2 and Surry Units 1 and 2 in Virginia, Peach Bottom Units 2 and 3 in Pennsylvania, Saint Lucie Units 1 and 2 in Florida, Fort Calhoun in Nebraska, McGuire Units 1 and 2 in North Carolina, and Catawba Units 1 and 2 in South Carolina -- had their licenses extended for an additional 20 years. Thus far in 2004, 2 units -- H.B. Robinson, Unit 2 and V.C. Summer, Unit 1 in South Carolina -- have had their licenses renewed. That brings the total of renewed reactor licenses to twenty-five. The staff currently has license renewal applications under review for seventeen additional units. In every instance, the staff has met its timeliness goals in carrying out the safety and environmental reviews required by our regulations. If all of the applications currently under review are approved, approximately 40 percent of the nuclear power plants in the U.S. will have extended their operating licenses. We expect that almost all of the 104 reactors licensed to operate will apply for renewal of their licenses. The staff will continue to face a significant workload in this area with the sustained strong interest in license renewal by nuclear power plant operators due to many benefits of license renewal.

While improved performance of operating nuclear power plants has resulted in significant increases in their electrical output, it is expected that continuing increased demands for electricity will need to be addressed by construction of new generating capacity. As a result, industry interest in new construction of nuclear power plants in the U.S. has recently emerged. The NRC is ready to accept applications for new power plants. New nuclear power plants will likely utilize 10 CFR Part 52, which provides a stable and predictable licensing process. This process ensures that all safety and environmental issues, including emergency preparedness and security, are resolved prior to the construction of a new nuclear power plant. The design certification part of the process resolves the safety issues related to the plant design, while the early site permit process resolves safety and environmental issues related to a potential site. The issues resolved in these two parts can then be referenced in an application which would lead to a combined construction permit and operating license, referred to as a combined license. This license contains inspections, tests, analyses, and acceptance criteria that must be attained before the facility can commence operation.

As you know, the NRC has already certified three new reactor designs. These designs include General Electric's Advanced Boiling Water Reactor and Westinghouse's AP600 and System 80+ designs. In addition to the three advanced reactor designs already certified, there are new nuclear power plant technologies which some believe can provide enhanced safety, improved efficiency, and lower costs. The NRC staff is currently reviewing the Westinghouse AP1000 design certification application. The staff has met all scheduled milestones for the AP1000 design review and is on track to issue its recommendations to the Commission this fall on whether the final design should be certified. This recommendation would be followed by the design certification rule in 2005. The NRC staff is also actively reviewing pre-application issues on two additional designs and has four other designs in various stages of pre-application review.

In September and October of last year, we received three early site permit applications for sites in Virginia, Illinois, and Mississippi where operating reactors already exist. The staff has established schedules to complete the safety reviews and environmental impact statements in approximately two years. The mandatory adjudicatory hearings associated with the early site permits will be concluded after completion of the NRC staff's technical review. As with design certification rulemaking, issues resolved in the early site permit proceedings will not be revisited during a combined license proceeding absent new and compelling information.

Security

During the past year, the Commission has continued to enhance security of licensed nuclear facilities and materials through close communication and coordination with other agencies in the intelligence and law enforcement communities and with the Department of Homeland Security. For commercial nuclear power reactors, we issued Orders in April 2003 to impose a revised design basis threat (DBT) and enhanced requirements for security officer work hour limits (to ensure officers remained fit for duty) and standards for their training and qualification. With these requirements, we have established an enhanced set of security requirements for power reactors that is appropriate in the post-9/11 threat environment. The work-hour limits and the previously imposed access authorization enhancements have been fully implemented. Revisions to site security plans (including training and qualification) and site modifications to provide protection against the revised DBT have been submitted to the NRC for review and implementation. The review is in progress with full implementation scheduled for October 2004. We have redefined our baseline inspection program for security and are phasing in the new inspection program consistent with the new requirements. As a complement to licensee security measures, NRC is working with the Department of Homeland Security and the Homeland Security Council, and other partners to enhance the integrated Federal, State, and local response to threats.

We continue to conduct force-on-force exercises to evaluate licensees' defensive capabilities and identify areas for improvement. During 2003, we implemented a pilot force-on-force exercise program and conducted exercises at 15 power plants to evaluate the significance and impact of enhanced adversary characteristics and associated compensatory measures and to develop program improvements to enhance the realism and effectiveness of the exercises. In 2004, we are conducting exercises roughly twice a month to evaluate the effectiveness of program enhancements including the use of Multiple Integrated Laser Enhancement System (MILES) equipment, adversary force standards, improved controller training, and other enhancements to improve the realism of the exercises while maintaining safety of both the plant and personnel. In November of this year, we will begin full implementation of the triennial force-on-force exercise program for power reactors.

In the area of materials security, we have coordinated closely with State agencies and affected licensee groups to develop additional security requirements for two classes of materials licensees who possess high-risk radioactive materials (irradiator licensees and manufacturers and distributors of radioactive materials). We are preparing proposed Orders for other materials users. We are developing enhanced import and export controls for high-risk sources. In addition, we have developed an interim database for high-risk sources and, with the assistance of other Federal agencies as well as the States, we are laying the foundations for the national source tracking system. We are also engaged with other Federal agencies to increase security involving transportation of large quantities of radioactive materials and are conducting a comprehensive review of material control and accounting requirements and practices.

The NRC has completed most of its work on vulnerability assessments and identification of mitigation strategies for a broad range of threats to NRC-licensed activities involving radioactive materials and nuclear facilities. Thus far, the results of these studies have validated the actions NRC has taken to enhance security. These efforts have continued to affirm the robustness of these facilities, the effectiveness of redundant systems and defense-in-depth design principles, and the value of effective programs for operator training and emergency preparedness. Our vulnerability studies confirm that the likelihood of damaging the reactor core and releasing radioactivity that could affect public health and safety is low. Further, the studies confirm that even in the unlikely event of a radiological release due to terrorist use of a large aircraft, NRC's emergency planning basis remains valid. The aircraft vulnerability studies also indicate that significant damage to a spent fuel pool is improbable, that it is highly unlikely that the impact on a dry spent fuel storage cask would cause a significant release of radioactivity, and that the impact of a large aircraft on a transportation cask would not result in a release of radioactive material. Thus, we believe that nuclear power plant safety, security, and emergency planning programs continue to provide reasonable assurance of adequate protection of the public health and safety.

In summary, NRC licensees had robust private sector security programs long before the attacks of September 11, 2001, and those programs have been further enhanced over the past 30 months. We continue to ensure that our licensees implement effective security programs for the current threat environment. In addition, we continue to work closely with our Federal, State, and local partners and with the private sector to ensure an appropriate integrated response to threats to licensed nuclear facilities and materials.

Emergency Preparedness Program

The events of September 11, 2001, highlighted the need to examine the way the NRC is organized to carry out its safeguards, security, and incident response functions. Consequently, the NRC has taken several actions in response to the new environment, including the issuance of compensatory measures and Orders to licensees, re-examination of the emergency planning basis, creation of the Office of Nuclear Security and Incident Response, and evaluation of reactor integrity to new threats. In addition, the NRC as well as our stakeholders have become increasingly aware of the importance of emergency preparedness to mitigating the effects of potential security threats. Along with this increased awareness, the NRC recognizes the need for increased communication of our emergency preparedness activities with internal and external stakeholders, including the public; industry; the international nuclear community; and Federal, state, and local government agencies. As a result, the NRC established the Nuclear Emergency Preparedness Project Office. The Project Office is responsible for the continuing development and refinement of emergency preparedness policies, regulations, programs, and guidelines for both currently licensed nuclear reactors and potential new nuclear reactors. The Project Office provides technical expertise regarding emergency preparedness issues to other NRC offices and also coordinates and manages emergency preparedness communications with internal and external stakeholders including the public, industry, the international nuclear community, and Federal, State, and local government agencies.

Materials Program

The NRC, in partnership with the 33 Agreement States, conducts a comprehensive program to ensure the safe use of radiological materials in a variety of medical and industrial settings. As some of NRC's responsibilities, including inspection and licensing actions, have been assumed by Agreement States, our success depends in part on their success, and we closely coordinate our activities with the States.

Recently, the Commission has completed a complex rulemaking on the medical uses of byproduct material -- a rulemaking in which there was significant interaction with Congress. We are now implementing that rule and assuring that compatible regulations are adopted by the Agreement States.

The NRC is developing a web-based materials licensing system. The system is expected to provide a secure method for licensees to request licensing actions and to view the status of licensing actions on the Web. In addition, the NRC, with assistance from other Federal agencies and the States, is creating a National Source Tracking System that will be used to monitor radioactive sources in quantities of concern with respect to a radiological dispersal device (RDD) threat. The development of the National Source Tracking System will remain a high priority effort.

The Commission has also implemented a major rule change related to large fuel cycle facilities. This rule requires licensees and applicants to perform an integrated safety analysis that applies risk-based insights to the regulation of their facilities. Major licensing reviews currently underway, or soon to be submitted, will test the new rule. These licensing reviews include two new gas centrifuge enrichment facilities.

The first proposed enrichment facility would be located in New Mexico and the second in Ohio. Louisiana Energy Services submitted an application for its facility in Eunice, New Mexico, to the NRC in December 2003. U.S. Enrichment Corporation is expected to submit its application to the NRC for its site in Piketon, Ohio, in August 2004. The Commission has directed its staff to conduct reviews of the applications for the two proposed enrichment facilities in a timely manner. The Commission will endeavor to identify efficiencies and provide the necessary resources to reduce the time the agency needs to complete these reviews.

The staff is currently reviewing a request to authorize construction of a mixed oxide (MOX) fuel fabrication facility at the Savannah River site in South Carolina as part of the Department of Energy's program to dispose of excess weapons grade plutonium. The staff is also providing support to its Russian counterparts regarding the licensing of a Russian MOX facility that will have a design similar to the U.S. facility.

In addition to the new facilities discussed above, the NRC regulates several other existing fuel facilities. NRC's oversight of these facilities includes licensing actions, inspection, enforcement, and assessment of licensee performance. Our Fuel Facilities Licensing and Inspection program was the second of our regulatory programs assessed under the Office of Management and Budget's Performance Assessment Rating Tool (PART) and awarded the top rating, "effective," a rating achieved by only 11% of the Federal programs evaluated.

Nuclear Waste Program

The NRC staff has made progress on a wide array of programs relating to the safe disposal of nuclear waste. A central focus of these programs is to ensure that the agency is prepared to review an application by the Department of Energy to construct a high-level radioactive waste repository at Yucca Mountain, Nevada. Progress has been made in our pre-application interactions with DOE in addressing technical issues that are significant to repository performance. The application is expected to be submitted to NRC in December 2004. The NRC would make a docketing decision on the license application, and, if docketed, review the license application and make a determination regarding to what extent the Yucca Mountain Final Environmental Impact Statement can be adopted.

We are also preparing to conduct a related licensing proceeding. Our preparations include the creation of an information technology system to handle the large number of complex documents that will be involved and the leasing of a hearing facility near Las Vegas, Nevada. This licensing proceeding will present the NRC with a formidable challenge and the technical issues involved will be substantial. Moreover, no single NRC decision or set of decisions, since the Three Mile Island accident, is likely to be scrutinized as closely as those concerning this one-of-a-kind facility.

In our waste program, the NRC staff also has a substantial effort underway in the area of dry cask storage of spent reactor fuel. Storage and transport cask designs continue to be reviewed and certified. Independent Spent Fuel Storage Installations (ISFSIs) continue to be licensed and inspected. The Atomic Safety and Licensing Board currently is expected to issue its final decision on the proposed Private Fuel Storage ISFSI in Utah early in 2005. The Surry ISFSI in Virginia is the lead facility for license renewal. Indeed, our workload related to ISFSIs and dry cask storage in general will increase substantially in the years ahead. This projection is based on licensees' plans to adopt dry cask storage at their sites. We are currently formulating a major research program, the Package Performance Study, which will include a demonstration test of the robustness of NRC-certified spent fuel transportation casks.

The NRC staff is also continuing to make significant progress in ensuring the decommissioning of contaminated sites. The staff identified several policy issues requiring Commission direction that will help expedite decommissioning under NRC's License Termination Rule, and the Commission has provided the necessary guidance. Complicated decommissioning sites that pose technical challenges include the Safety Light site near

Bloomsburg, Pennsylvania. We are currently working with the Environmental Protection Agency to have this site included on the National Priority List to make other Federal resources available for the cleanup of this site.

Human Capital

The NRC is very dependent on a highly skilled and experienced work force for the effective execution of its activities. The Commission's human capital planning integrates strategies for finding and attracting new staff, and for promoting employee development, succession planning, and retention. The Commission has developed and implemented a strategic workforce planning system to identify and monitor its human capital assets and needs and to address critical skills shortages. This includes the use of an agency-wide online skills and competency system to identify gaps in needed skills; the ongoing review of NRC's organizational structure to align with its mission and goals; and the development of a web-based staffing system that includes online application, rating, ranking, and referral features. The agency has also implemented two leadership competency development programs to select high-performing individuals and train them for future mid-level and senior-level leadership positions. In addition, the agency has continued to support its fellowship and scholarship programs and identified a significant number of diverse, highly qualified entry-level candidates through participation in recruitment events and career fairs.

NRC is utilizing a variety of recruitment and retention incentives to remain competitive with the private sector. So far we have been successful in attracting and retaining new staff, particularly at entry levels. Nonetheless, it is likely to become more difficult for NRC to hire and retain personnel with the knowledge, skills, and abilities to conduct the safety reviews, licensing, research, and oversight actions that are essential to our safety mission. Moreover, the number of individuals with the technical skills critical to the achievement of the Commission's safety mission is rapidly declining in the Nation, and the educational system is not replacing them. The maintenance of technically competent staff will continue to challenge governmental, academic, and industry entities associated with nuclear technology for some time to come.

Budget

The NRC has proposed a Fiscal Year 2005 budget of $670.3 million. In developing the budget, the Commission has ensured that we continue only those programs that are effective in meeting our mission and goals. Even with our efforts to be more efficient in our utilization of resources, we must still request a Fiscal Year 2005 budget increase of approximately 7 percent ($44 million) over the Fiscal Year 2004 budget for essential activities. This budget proposal will allow the NRC to continue to protect the public health and safety, promote the common defense and security, and protect the environment, while providing sufficient resources to address increasing personnel costs and new work. Approximately 32 percent ($14 million) of the budget growth is for personnel costs, primarily the pay raise that the President has authorized for Federal employees. The remaining increase supports our High-Level Waste and Nuclear Reactor Safety programs. We are requesting an increase of approximately $30 million for our High-Level Waste program to initiate the review of the anticipated DOE application to construct a high-level waste repository at Yucca Mountain and to conduct a Package Performance Study, which will confirm that our regulations provide for the safe transportation of spent nuclear fuel even under accident scenarios. We are also requesting an increase of approximately $10 million for our Nuclear Reactor Safety programs primarily to keep pace with industry interest in new reactor initiatives and to strengthen our reactor inspection and performance assessment activities. These increases are offset by a decrease of approximately $10 million in our Homeland Security programs for completed homeland security activities.

Legislative Needs

Over the years, the NRC has repeatedly expressed its support of enactment of legislation needed to strengthen the security of facilities regulated by the Commission. Although we did not support all the provisions contained in bills that addressed nuclear security in the first session of this Congress, we were encouraged by Congressional action on the subject. Although, the Commission has used existing authority to ensure robust security for nuclear power plants and high risk radioactive materials, provisions that the Commission supports would provide the statutory authority for steps that we believe should be taken to further enhance the protection of the country's nuclear infrastructure and prevent malevolent use of radioactive material. In particular, the Commission supports enactment of the nuclear security-related provisions contained in H.R. 6, as approved by the conferees on that bill in the last session of this Congress, and S. 2095, which has been introduced in this session.

The proposals that the Commission believes to be most important are: (1) authorization of security officers at NRC-regulated facilities and activities to receive, possess, and, in appropriate circumstances, use more powerful weapons against terrorist attacks, (2) enlargement of the classes of NRC-regulated entities and activities whose employees are subject to fingerprinting and criminal history background checks, (3) Federal criminalization of unauthorized introduction of dangerous weapons into nuclear facilities, (4) Federal criminalization of sabotage of additional classes of nuclear facilities, fuel, and material,

(5) authorization for NRC to carry out a training and fellowship program to address shortages of individuals with critical nuclear regulatory skills, and (6) extension of NRC's regulatory oversight to discrete sources of accelerator-produced radioactive material and radium-226. All but the last of these are included in H.R. 6 and S. 2095.

In addition, enactment of the following proposals would enhance the NRC's ability to protect the public health and safety:

(1) long-term extension of the Price-Anderson Act;

(2) authorization to charge Federal agencies fees for licensing and inspections, rather than recouping the costs of these activities through charges to other licensees;

(3) authorization for costs of security-related activities to be covered from the general fund (except for fingerprinting, criminal background checks, and security inspections);

(4) elimination of NRC's antitrust review authority over new power reactor license applications;

(5) clarification of the length of combined construction permits and operating licenses for new reactors;

(6) allowing rehired annuitants to receive full pay from the NRC for their services without reduction in pension payments;

(7) authorization to compensate individuals with critical skills at rates competitive with rates paid to persons with similar skills in the private sector;

(8) modification of the organizational conflict of interest provisions in the Atomic Energy Act to allow the agency to engage valuable expertise at a national laboratory that also performs work for the nuclear industry; and

(9) authorization to establish and participate in science, engineering, and law partnership outreach programs to increase the participation of Historically Black Colleges and Universities, Hispanic Serving Institutions, and Tribes.

All but the last three proposals are included in H.R. 6 and S. 2095. We look forward to working with you on the enactment of these proposals by this Congress.

Conclusion

Mr. Chairman, I can assure you that the Commission will continue to be very active in managing the staff's efforts on ensuring the adequate protection of public health and safety, promoting common defense and security, and protecting the environment in the application of nuclear technology for civilian use.

We appreciate the opportunity to appear before you today. My colleagues and I welcome the opportunity to respond to your questions.

EA-04-063- Browns Ferry (Tennessee Valley Authority)

May 12, 2004

EA-04-063

Tennessee Valley Authority
ATTN: Mr. J. A. Scalice
Chief Nuclear Officer and
Executive Vice President6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: NOTICE OF VIOLATION (BROWNS FERRY NUCLEAR PLANT UNIT 1 RECOVERY - NRC INSPECTION REPORT NO. 05000259/2004011)

Dear Mr. Scalice:

This refers to the inspection completed on February 13, 2004, involving recovery activities at Tennessee Valley Authority's (TVA) Browns Ferry 1 (BF1) reactor facility. The results of the inspection, including the identification of an apparent violation of 10 CFR 50, Appendix B, Criterion V, were forwarded to you by NRC letter dated April 6, 2004. Based on the results of the inspection, a pre-decisional enforcement conference was held on April 28, 2004, in the NRC's Region II Office in Atlanta, Georgia, with members of your staff to discuss the apparent violation, its significance, root causes, and your corrective actions. A listing of conference attendees, material presented by the NRC, and material presented by TVA are included as Enclosures 2, 3, and 4, respectively.

Based on the information developed during the inspection, and the information presented at the conference, the NRC has determined that a violation of NRC requirements occurred. The violation is cited in the enclosed Notice of Violation (Notice), and the circumstances surrounding it are described in detail in the subject inspection report. The violation involves four examples of a failure to adhere to the requirements of 10 CFR 50, Appendix B, Criterion V. All four examples were associated with the BF1 Long-Term Torus Integrity Program, and involved: failure to evaluate or incorporate numerous deficient welds into Deficiency Fix Requests sketches; failure to perform numerous repairs on the correct welds; omission of numerous welds requiring repair from Work Orders, and failure of Quality Control (QC) to independently verify the correct location of numerous weld repairs. At the conference, TVA acknowledged the errors, discussed its root cause and extent of condition reviews, and corrective actions.

As described in NRC Manual Chapter 2509, "Browns Ferry Unit 1 Restart Project Inspection Program", and explained during the conference, BF1 is not considered to fall within the scope of the Commission's current "General Statement of Policy and Procedures for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600, for commercially operating nuclear power plants. As such, traditional enforcement is in effect for the restart of BF1 for violations in those cornerstones which cannot be monitored under the Reactor Oversight Program. The significance of violations will be evaluated in accordance with 10 CFR Part 2 and other applicable enforcement guidance, including Supplement II of the Enforcement Policy. In this case, the violation identified above involves TVA's Quality Assurance program for construction related to a single work activity (BF1 Long-Term Torus Integrity Program), and involves a failure to conduct adequate audits/reviews and take prompt corrective actions on the basis of such audits/reviews. In addition, the errors were associated with multiple examples of deficient construction due to inadequate program implementation. As such, the NRC has concluded that the violation is appropriately characterized at Severity Level III.

In accordance with the Enforcement Policy, a base civil penalty in the amount of $60,000 is considered for a Severity Level III violation. Because your facility has not been the subject of escalated enforcement action within the last 2 years, the NRC considered whether credit was warranted for *Corrective Action* in accordance with the civil penalty assessment process in Section VI.C.2 of the Enforcement Policy. TVA's immediate corrective actions included the development and implementation of a plan to systematically verify the scope of torus weld problems. The plan consisted of training personnel on torus orientation and the proper use of sketches, independent review of the welds that were to be repaired to ensure they were identified in work documents, a walk-down of the torus welds that did not require repair to verify acceptability, and a determination of the cause of each example of the violation. Other corrective actions included the verification and revision of torus sketches, the placement of placards inside the torus to aid in orientation, revision of weld data sheets and weld maps, establishment of a single point of contact to control sketches, meetings with QC inspectors to stress the critical importance of independence, additional training for QC inspectors, increased Nuclear Assurance oversight of field activities, the assignment of dedicated resources for focused oversight of QC and other disciplines, and the conduct of a self-assessment of BF1 Nuclear Assurance oversight effectiveness. Based on these and other corrective actions discussed at the conference, the NRC concluded that credit was warranted for the factor of *Corrective Action*.

Therefore, to encourage prompt and comprehensive correction of violations and in recognition of the absence of previous escalated enforcement action, I have been authorized to propose that no civil penalty be assessed in this case. However, similar violations in the future could result in further escalated enforcement action. Issuance of this Notice constitutes escalated enforcement action, that may subject you to increased inspection effort.

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

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

Sincerely,

/RA/ LAR

Luis A. Reyes
Regional Administrator

Docket No. 50-259
License No. DPR-33

Enclosures:

1. Notice of Violation
2. List of Attendees
3. Information Presented by NRC
4. Information Presented by TVAcc w/encls:

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Tennessee Valley Authority
Electronic Mail Distribution

James E. Maddox, Vice President
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Browns Ferry Nuclear Plant
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NOTICE OF VIOLATION
AND
PROPOSED IMPOSITION OF CIVIL PENALTY

Tennessee Valley Authority
Browns Ferry Unit 1

Docket No. 50-259
License No. DPR-33
EA-04-063

During an NRC inspection completed on February 13, 2004, 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.

10 CFR 50, Appendix B, Criterion V, requires activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances, and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Contrary to the above, as of February 13, 2004, instructions, procedures, or drawings were inadequate or were not implemented for weld repairs to ECN P-0093 torus modifications as described below:

1. TVA procedure NEDP-5, Design Documents Review, Section 3.1.1 requires the preparer of design documents to provide an adequate and accurate solution for the problem, provide a quality product, and ensure that the design documents are complete. Section 3.1.2 requires the Checker (design verifier) to ensure that the design documents are adequate, complete and accurate.

   Deficiency Fix Request Sketches for the Long Term Torus Integrity Program were inadequate, in that approximately 50 examples of deficiencies requiring repairs were not identified on the sketches. In this regard, the preparer and design verifier failed to ensure that discrepancies identified during the torus walkdowns were adequately and accurately evaluated, failed to ensure that the discrepancies requiring repair were included in engineering output documents (Deficiency Fix Request Sketches), and failed to ensure the sketches were accurate and that required repairs were shown at the correct locations.

2. The drawings titled Deficiency Fix Requests, Sketches 4 through 38, detailing corrective actions for Problem Evaluation Report (PER) 03-017339, Unit 1 Torus, Differences Between As-Built and As-Designed Configurations, show locations for repairs to welds.

   Welds designated as weld numbers MS-1-WO 03017394016-008 in work order 03-017394-016, weld numbers PCI-1-WO 03017394002-029 and -30 in work order 03-017394-002, and weld numbers MS-1-WO 03017394006-047, -048, PCI-1-002-004, -005, and -006 in work order 03-017394-006, were repaired (welded) at the incorrect location. However, review of the work order documentation, specifically weld maps and data sheets, indicated the welds had been repaired. The deficient welds at these locations shown on Deficiency Fix Requests, Sketches 31 and 36 were not repaired. Approximately 20 additional welds were identified by the licensee which were repaired in the incorrect location.

3. TVA Procedure VT-6, Visual Examination of Structural Welds Using the Criteria of NCIG-01, requires quality control inspectors to perform an independent inspection of completed work activities important to safety. A requirement of the inspection procedure is independent verification that the work was performed at the correct location.

   Quality Control (QC) inspection personnel failed to independently verify that welds designated as weld numbers MS-1-WO 03017394016-008 in work order 03-017394-016, weld numbers PCI-1-WO 03017394002-029 and -30 in work order 03-017394-002, and weld numbers MS-1-WO 03017394006-047, -048, PCI-1-002-004, -005, and -006 in work order 03-017394-006 were repaired at the correct location. However, review of the QC inspection documentation in the work orders indicated the welds had been repaired, inspected, and accepted by quality control inspectors. The deficient welds at these locations shown on Deficiency Fix Requests, Sketches 31 and 36 were not repaired.

4. TVA Procedure MMDP-1, Maintenance Management System, Paragraph 3.2, requires work orders to be developed to a level of detail appropriate for the circumstances which address the aspects of the work, including the scope of the work and work instructions. MMDP-1 requires that the work order specify that work is to be performed in

accordance with approved procedures, when approved procedures are available. Paragraph 3.8.1 of TVA procedure MMDP-1 requires independent/technical review of the work order to insure the work order contains detailed work steps to perform the required work prior to approval and implementation of the work order.

TVA procedure MMDP-10, Controlling Welding, Brazing, and Soldering Processes, Section 3.3, requires work implementing documents and weld data sheets be prepared and included in the work order for all welding activities.

Work implementing documents and weld data sheets for six welds, which required restoration to the sizes shown on Deficiency Fix Request, Sketch Number 30, referenced in PER 03-017394, were omitted from Work Order 030017394-006. The independent/technical review of the work order did not identify the omission when performing the independent technical quality review. As followup, the licensee identified approximately 30 additional welds which were shown on the drawings as requiring repair but were not included in the work order instructions.

This is a Severity Level III Violation (Supplement II).

Pursuant to the provisions of 10 CFR 2.201, Tennessee Valley Authority is hereby required to submit a written statement or explanation to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, within 30 days of the date of this Notice of Violation and Proposed Imposition of Civil Penalty (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-04-063" and should include: (1) admission or denial of the alleged violation, (2) the reasons for the violation if admitted, and if denied, the reasons why, (3) the corrective steps that have been taken and the results achieved, (4) the corrective steps that will be taken to avoid further violations, and (5) the date when full compliance will be achieved. Your response may reference or include previously docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked or why such other action as may be proper should not be taken. Consideration may be given to extending the response time for good cause shown.

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

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

Dated this 12th day of May 2004

EA-04-086 - Hope Creek 1 (PSEG Nuclear LLC)

May 10, 2004

EA-04-086

Mr. Roy A. Anderson
Chief Nuclear Officer and President
PSEG Nuclear LLC - N09
P. O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (NRC Inspection Report 05000354/2003006) Hope Creek Nuclear Generating Station

Dear Mr. Anderson:

The purpose of this letter is to provide you with the final results of our significance determination for the preliminary White finding identified at the Hope Creek Nuclear Generating Station during an inspection completed on December 31, 2003. The results of the inspection were discussed with Mr. J. Hutton and other members of your staff on January 21, 2004. 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 resulted from a self-revealing event and involved maintenance procedures that you found had failed to contain adequate instructions and were not followed, which contributed to the "A" station service water system (SSWS) traveling screen failure that occurred on July 1, 2003. The unavailability of the SSWS traveling screen increased the likelihood of the loss of service water initiating event and affected the ability of a service water pump train to mitigate the effects of initiating events.

Regarding the finding, a maintenance procedure did not include appropriate quantitative acceptance criteria to ensure that the SSWS traveling screen head-shaft key was installed correctly, and as a result, the key was cut too short by maintenance workers. Also, the traveling screen basket chains had not been tensioned adequately in accordance with another maintenance procedure, and they failed to document that this procedure had been completed.

In a letter dated April 20, 2004, the NRC transmitted the referenced inspection report and informed you that the staff had sufficient information to make an enforcement decision. However, you were given an opportunity to request a regulatory conference or to provide a written response. In a telephone conversation on May 3, 2004, Mr. S. Mannon, PSEG Nuclear LLC, informed Mr. W. Lanning, NRC, Region I, that the licensee will not request a Regulatory Conference nor provide 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 Hope Creek is appropriately characterized as White, an issue with low to moderate importance to safety, which may require additional NRC inspection. Although you have not indicated a desire to do so, our process allows 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 examples of a violation of 10 CFR 50, Appendix B, Criterion V, as described in the enclosed Notice of Violation (Notice). The circumstances surrounding this violation are described in detail in the subject inspection report. In accordance with the NRC Enforcement Policy, NUREG-1600, the Notice is considered escalated enforcement action because it is associated with a White finding. You are required to respond within 30 days of the date of this letter and should follow the instructions specified in the enclosed Notice when preparing your response.

This issue causes the Hope Creek facility to be in the regulatory response band of the NRC Action Matrix, and we will notify you, by separate correspondence, of any further action we plan to take.

In accordance with 10 CFR 2.390 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 http://www.nrc.gov/reading-rm/doc-collections/ enforcement/actions.

Sincerely,

/RA/

Hubert J. Miller
Regional Administrator

Docket No. 50-354
License No. NPF-57

Enclosure: Notice of Violation

cc w/encl:
C. Bakken, Senior Vice President Nuclear Operations
M. [illegible]ers, Vice President - Site Operations
J. [illegible]lin, Vice President Nuclear Assurance
D. F. Garchow, Vice President, Engineering and Technical Support
W. F. Sperry, Director Business Support
S. Mannon, Manager - Licensing (Acting)
J. A. Hutton, Hope Creek Plant Manager
R. Kankus, Joint Owner Affairs
J. J. Keenan, Esquire
Consumer Advocate, Office of Consumer Advocate
F. Pompper, Chief of Police and Emergency Management Coordinator
M. Wetterhahn, Esquire
J. Lipoti Ph.D., Assistant Director of Radiation Programs, State of New Jersey
H. Otto, Ph.D., DNREC Division of Water Resources, State of Delaware
N. Cohen, Coordinator - Unplug Salem Campaign
W. Costanzo, Technical Advisor - Jersey Shore Nuclear Watch
E. Zobian, Coordinator - Jersey Shore Anti Nuclear Alliance

NOTICE OF VIOLATION

PSEG Nuclear LLC
Hope Creek Nuclear Generating Station

Docket No. 50-354
License No. NPF-57
EA-04-086

During an NRC inspection conducted between September 28, 2003 and December 31, 2003, for which our exit meeting was held on January 21, 2004, 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:

10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures and Drawings" requires that activities affecting the quality of safety-related equipment functions be accomplished in accordance with documented instructions, procedures or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Contrary to the above, between June 20 and June 26, 2003, the licensee replaced the head-shaft on the "A" service water system traveling screen under Work Order 60037345. Procedures directed to be used by the work order failed to contain adequate instructions to perform the maintenance and were not followed, resulting in the subsequent failure of the traveling screen on July 1, 2003. Specifically,

1. Procedure HC.MD-CM.EP-0003(Q), "Service Water Traveling Screens Overhaul and Repair," Revision 11, did not include appropriate quantitative acceptance criteria to ensure that the vendor-supplied service water system traveling screen head-shaft key was installed correctly. As a result, the key was cut too short during installation.

2. Procedure HC.MD-PM.EP-0001(Q), "Service Water Traveling Screen 12 Month Preventative Maintenance," Section 5.4.1, provided acceptance criteria to level the traveling water screen head-shaft while applying tension on the basket chains. The licensee determined that the traveling screen basket chains had not been tensioned adequately during the work, and the licensee failed to document in Work Order 60037345 that the procedure had been completed.

This violation is associated with a WHITE significance determination process finding.

Pursuant to the provisions of 10 CFR 2.201, PSEG Nuclear LLC is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D.C. 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 of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-04-086" and should include for the 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, United States Nuclear Regulatory Commission, Washington, D.C. 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. 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. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

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

Dated this 10th day of May 2004.

EA-03-209 - Davis-Besse (FirstEnergy Nuclear Operating Company)

May 7, 2004

EA-03-209

Mr. Lew Myers
Chief Operating Officer
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION - NOTICE OF VIOLATION NRC SPECIAL INSPECTION - COMPLETENESS AND ACCURACY OF REQUIRED RECORDS AND SUBMITTALS TO THE NRC - REPORT NO. 50-346/03-19(DRP)

Dear Mr. Myers:

This refers to the inspection conducted from October 20 through 24, 2003, at the Davis-Besse Nuclear Power Station. The purpose of the inspection was for the NRC to determine whether reasonable confidence exists that important docketed information is complete and accurate in all material respects and that Davis-Besse personnel took appropriate corrective actions to ensure that future regulatory submittals are complete and accurate. During the exit meeting on November 12, 2003, the NRC informed FirstEnergy Nuclear Operating Company (FENOC) of an apparent violation for the failure to provide complete and accurate information in the November 11, 1998, response to NRC Generic Letter (GL) 98-04 regarding protective coating deficiencies and foreign material in containment. FENOC staff had previously identified this issue.

The NRC letter dated January 28, 2004, transmitting Inspection Report 50-346/03-19, provided FENOC the opportunity to address the apparent violation identified in the report before the NRC made its final enforcement decision by either attending a predecisional enforcement conference or by providing a written response. In a letter dated February 27, 2004, FENOC provided a response to the apparent violation.

In its February 27, 2004, letter, FENOC admitted that the violation occurred and described the reasons for the violation, the actions taken to correct the violation and underlying hardware deficiencies, and actions taken to prevent recurrence of the violation. These actions included revising the procedure for regulatory submittals to ensure they are properly validated before submission to the NRC, training employees on the revised procedures and the requirements of 10 CFR 50.9, and submitting a revised response to Generic Letter 98-04. FENOC also stated that it was in full compliance.

The letter also requested that the NRC not issue a civil penalty because FENOC believed credit is warranted for licensee identification and corrective action; the time for issuing civil penalties has exceeded the statute of limitations; and the criteria are met for exercise of discretion in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600, Section VII.B.2, "Violations Identified During Extended Shutdowns or Work Stoppages."

Based on the information developed during the inspection and the information that you provided in your February 27, 2004, letter, the NRC has determined that a violation of NRC requirements occurred. The violation is cited in the enclosed Notice of Violation and the circumstances surrounding it are described in detail in Inspection Report 50-346/03-19. Specifically, the coatings and debris had the potential to clog the emergency sump screen and affect post accident long-term reactor core cooling and containment atmosphere cooling by the high pressure injection, low pressure injection, and containment spray systems.

This violation is significant because inaccurate information was provided to the NRC regarding significant safety issues at Davis-Besse. Had the issues related to deficient coatings and debris been disclosed in the November 11, 1998, response to GL 98-04, the NRC would have initiated substantial further inquiry. Therefore, this violation has been categorized in accordance with the Enforcement Policy at Severity Level III.

In accordance with the Enforcement Policy, a base civil penalty in the amount of $55,000 was considered for a Severity Level III violation that occurred in 1998. To encourage prompt identification and comprehensive correction of violations, Section VI.C.2 of the Enforcement Policy permits mitigation of the base civil penalty if certain criteria are met. Davis-Besse met these criteria as described in FENOC's February 27, 2004, letter. Further, the time frame permitted in the statute of limitations for applying civil sanctions has been exceeded. Therefore, no civil penalty will be proposed.

Section VII.B.2 of the Enforcement Policy permits discretion for the NRC to not issue a Notice of Violation or impose enforcement sanctions if specific criteria are met regarding violations identified during extended shutdowns. As described in FENOC's February 27, 2004, letter, this violation met these criteria. Notwithstanding the provisions of Section VII.B.2, after consultation with the Director, Office of Enforcement, I have decided to issue the enclosed Notice of Violation to emphasize the importance of providing complete and accurate information to the NRC. Significant violations in the future could result in civil penalties.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved, is already adequately addressed on the docket in NRC Inspection Report 50-346/03-19, LERs 2003-002 and 2002-005, and licensee letters dated February 27, 2004 (Serial Number 1-1349), November 26, 2003 (Serial Number 2994), and October 24, 2003 (Serial Number 1-1330). Therefore, you are not required to respond to this letter unless the descriptions therein do 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 of Violation.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), 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/

James L. Caldwell
Regional Administrator

Docket No. 50-346
License No. NPF-3

Enclosure: Notice of Violation

cc w/encl:
The Honorable Dennis Kucinich
G. Leidich, President - FENOC
J. Hagan, Senior Vice President
Engineering and Services, FENOC
Plant Manager
Manager - Regulatory Affairs
M. [illegible]lly, Attorney, FirstEnergy
Ohio State Liaison Officer
R. Owen, Administrator, Ohio Department of Health

Public Utilities Commission of Ohio
President, Board of County Commissioners
of Lucas County
C. [illegible]el, President, Ottawa County Board of Commissioners
D. [illegible]baum, Union Of Concerned Scientists
J. Riccio, Greenpeace
P. Gunter, N.I.R.S.

NOTICE OF VIOLATION

First Energy Nuclear Operating Company
Davis-Besse Nuclear Power Plant, Unit 1

Docket No. 50-346
License No. NPF-3
EA-03-209

During an NRC inspection conducted from October 20 through October 24, 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:

> 10 CFR 50.9 requires, in part, that information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects. On July 14, 1998, the NRC issued 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." The licensee's November 11, 1998, letter responding to NRC Generic Letter 98-04 stated:
>
> a. The Service Level 1 protective coatings used inside containment at the DBNPS are qualified with the exceptions noted in the response to Item 1."
>
> b. ". . . large amounts of paint are not likely to be carried to the emergency sump screen and clog over 50 percent of the screen area preventing long-term or containment atmosphere cooling by HPI, LPI, CS or the CACs."
>
> c. "Any paint debris fragments that are small enough to pass through the 1/4-inch emergency sump intake screen openings would not clog spray nozzles or damage pumps."
>
> Contrary to the above, the licensee failed to provide to the Commission complete and accurate information in its November 11, 1998, response. Specifically:
>
> a. Unqualified coatings applied to structures, systems, and components located in the containment building were applied to surfaces other than those listed in the exceptions in the response to Item 1. Locations where unqualified coatings not listed as exceptions existed included the reactor vessel, steam generators pressurizer, reactor coolant system piping, and core flood tanks.
>
> b. Large amounts of paint were likely to be carried to the emergency sump screen and clog over 50 percent of the screen area preventing long-term or containment atmosphere cooling.
>
> c. Paint debris fragments small enough to pass through the emergency sump screen openings could have damaged the high pressure injection pumps.
>
> This information was material because it affected whether the emergency core cooling system and the containment spray system would perform their safety functions after a postulated loss-of-coolant accident.
>
> This is a Severity Level III violation (Supplement VII).

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence and the date when full compliance was achieved is already adequately addressed on the docket in NRC Inspection Report 50-346/03-19, LERs 2003-002 and 2002-005, and licensee letters dated February 27, 2004 (Serial Number 1-1349), November 26, 2003 (Serial Number 2994), and October 24, 2003 (Serial Number 1-1330). 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," include the EA number, 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 III, and a copy to the NRC Resident Inspector at the Davis-Besse facility within 30 days of the date of this letter.

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, United States 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 two working days.

Dated this 7th day of May 2004.

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs
Washington, DC 20555-0001

Telephone: 301/415-8200
E-mail: opa@nrc.gov

www.nrc.gov

No. 04-064 May 27, 2004

NRC TO CONDUCT PILOT INSPECTION PROGRAM FOCUSED ON NUCLEAR PLANT ENGINEERING AND DESIGN ISSUES

The Nuclear Regulatory Commission is preparing a new inspection program that could eventually be applied to the nation's 104 commercial nuclear power plants.

"The program is intended to provide a more in-depth inspection of engineering activities, thereby improving the ability of the agency's current Reactor Oversight Process to identify significant engineering issues before they could impact plant sa[illegible] said NRC Chairman Nils Diaz.

The new program will focus on verifying that a plant's design basis has been correctly implemented for selected components that play a significant role in either reducing the risk of an accident or mitigating one. A pilot inspection will be carried out at four sites -- Vermont Yankee and three others yet to be determined. The pilot program incorporates aspects of existing and past programs, and includes:

-- Devoting significant effort to assessing industry operating experience relevant to the components being inspected;
-- Enlarging the inspection sample, which could now include components that could contribute to the initiation of an accident;
-- Creating a more detailed inspection report that integrates assessment of any design/engineering weaknesses, and;
-- Conducting approximately 700 hours of direct inspection.

An important aspect of the new inspection is that it will more intently focus resources on areas of risk significance and components operating close to design margins.

The NRC expects the pilot inspections will be completed in six to nine months. The agency will then review the inspection results and determine whether permanent changes to the Reactor Oversight Process are warranted. Additional information on the pilot inspection program is available electronically through the NRC's Agencywide Documents Access and Management System on the NRC web site at: http://www.nrc.gov/reading-rm/adams/web-based.html, by entering accession number ML040970328.

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-04-023 May 26, 2004

CONTACT: Victor Dricks E-mail: opa4@nrc.gov
Phone: 817-860-8128

NRC TO DISCUSS PERFORMANCE
OF SOUTH TEXAS PROJECT NUCLEAR PLANT JUNE 3

Printable Version

The U.S. Nuclear Regulatory Commission staff will meet with representatives of the South Texas Project Nuclear Operating Company on June 3 to discuss the results of the agency's assessment of safety performance at the South Texas Project nuclear power plant during 2003. The plant is located in Bay City, Texas.

The meeting will be held at 6 p.m. at the Bay City Civic Center, 201 Seventh Street, Bay City. The public is invited to observe the meeting and NRC officials will be available before the conclusion of the meeting to answer questions from the public.

The performance period to be discussed is January 1 to December 31, 2003. In addition, the NRC staff will provide an overview of how the agency Reactor Oversight Process works.

A letter from the NRC to South Texas Project officials addresses the performance of the plant during this period and will serve as the basis for the meeting discussion. It is available on the NRC web site at: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/LETTERS/stp_2003q4.pdf.

The NRC concluded that the plant operated safely last year. "However because the number of automatic shutdowns crossed a critical threshold during the first quarter of 2003, the NRC conducted a supplemental inspection, which concluded that appropriate corrective actions had been taken," said Arthur T. Howell III, Director of Region IV's Division of Reactor Projects. Routine inspections will continue in 2004.

With regard to security issues, the letter points out that the NRC has issued several orders and threat advisories to enhance security capabilities at all nuclear power plants and improve guard force readiness since the terrorist attacks on September 11, 2001. The agency has also conducted inspections to review the implementation of these requirements and has monitored the action of plant operators in response to changing plant conditions. The NRC will continue security inspections during 2004.

Current performance indicators for Unit 1 are available on the NRC web site at: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/STP1/stp1_chart.html.

Current performance indicators for Unit 2 are available on the NRC web site at:

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs, Region III
801 Warrenville Road, Lisle IL 60532
www.nrc.gov

No. III-04-035 May 25, 2004
CONTACT: Jan Strasma (630) 829-9663 E-mail: opa3@nrc.gov
Viktoria Mitlyng (630) 829-9662

NRC TO MEET WITH NUCLEAR MANAGEMENT COMPANY TO DISCUSS PERFORMANCE OF KEWAUNEE NUCLEAR PLANT

Printable Version

The U.S. Nuclear Regulatory Commission staff will meet with representatives of Nuclear Management Company on June 2 to dis[illegible] the results of the agency's assessment of safety performance at the Kewaunee Nuclear Power Plant during 2003. The fa[illegible]s located in Kewaunee, Wisconsin.

The meeting will be held at 7 p.m. at the Kewaunee Municipal Building Council Chamber Conference Room, 401 5th Street, in Kewaunee. The public is invited to observe the meeting, and NRC officials will be available before the conclusion of the meeting to answer questions from the public. In addition, the NRC staff will provide an overview of how the agency's Reactor Oversight Process works.

The NRC has concluded that the plant operated safely last year. All NRC inspection findings during the year were of very low safety significance and safety performance data indicated no issues requiring NRC follow-up.

Routine inspections are performed by the two NRC resident inspectors assigned to the plant and by inspection specialists from the NRC's Region III office in Lisle, Illinois.

In its assessment, the NRC noted two concerns: recent inspections findings indicated deficiencies within the engineering program and longstanding issues in the area of emergency preparedness had not been resolved effectively. The NRC will continue to monitor the utility's response to these two issues.

A March 4 letter from the NRC to Nuclear Management Company officials addresses the performance of the plant during 2003 and will serve as the basis for the meeting discussion. It is available at: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/LETTERS/kewa_2003q4.pdf.

With regard to security issues, the NRC has issued several orders and threat advisories to enhance security capabilities and improve guard force readiness since the terrorist attacks on September 11, 2001. The agency has also conducted inspections to review the implementation of these requirements and has monitored the action of plant operators in response to changing th[illegible]nditions. The NRC will continue security inspections during 2004.

Current performance indicators and inspection findings for the Kewaunee plant are available on the NRC web site at: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/KEWA/kewa_chart.html.

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

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NRC BEGINS SPECIAL INSPECTION OF PUMP FAILURE AT PERRY NUCLEAR PLANT

Printable Version

The Nuclear Regulatory Commission has begun a special inspection to review the circumstances surrounding a failure of a pump which provides cooling water to various safety systems at the Perry Nuclear Power Plant. The plant, located in Perry, Ohio, is operated by FirstEnergy Nuclear Operating Company.

The pump in the emergency service water system failed during testing on Friday, May 21. Plant personnel shut the plant down on Saturday to investigate the cause of the pump failure and to make necessary repairs.

The pump is one of three in the emergency service water system. At the time of the failure, the other pumps were available to provide cooling to plant equipment if needed.

A similar failure of this pump occurred during testing on September 1 of last year. That problem was attributed to improper reassembly of the pump following maintenance in 1997.

The NRC had started a broad team inspection on May 17 to review that September 1 pump failure along with two other equipment problems which have occurred at Perry over the past 18 months. These equipment problems were determined to be of low to moderate safety significance -- "white" inspection findings in the NRC classification of problems which ranges from "green," for findings of minor safety significance, through "white," "yellow," and "red," indicating increasing safety significance.

The May 17 inspection team has concluded the first week of its inspection and plans to return June 7 for a second week of inspection.

The special inspection looking at the May 21 failure is separate from the broader team inspection, although both involve the NRC's resident inspector at Perry.

The reports of both the special inspection and the broader equipment inspection will be publicly available about a month following the completion of the inspections. They may be obtained from the Region III Office of Public Affairs or from the NRC's online document library: http://www.nrc.gov/reading-rm/adams/web-based.html - use docket number 05000440 to locate Perry documents. Assistance in using the online document library is available by calling the NRC Public Document Room at 800-397-4209.

SAFETY CULTURE STANDARDS SHOULD BE SET TO AVOID A DAVIS-BESSE REPEAT Sen. George Voinovich (R-Ohio) insisted today at an oversight hearing on NRC. Voinovich pounded the NRC commissioners with questions about the lack of regulations on assessing a company's attitude toward safety. NRC Chairman Nils Diaz told the Senate Environment & Public Works Subcommittee on Clean Air, Climate Change & Nuclear Safety that NRC was "not in the business" of directly managing utilities but does oversee companies' safety management. Voinovich, who chairs the subcommittee, said he didn't believe that was good enough and promised to have more talks with the commission about establishing some requirements. Testifying on a separate witness panel, Union of Concerned Scientists nuclear safety engineer David Lochbaum told the subcommittee that NRC "has a safety culture issue of its own" that needs an internal fix. (*NUCLEAR NEWS FLASHES - Thursday, May 20, 2004*)

LAWMAKERS SHOULD REVIEW NRC'S BUDGET AND STAFFING LEVELS to see where cost savings could be achieved, Nuclear Energy Institute Chief Nuclear Officer Marvin Fertel urged a Senate panel. Speaking at an NRC oversight hearing, Fertel suggested more could be done to build a "stable" regulatory environment to sustain the industry. He said in written testimony that NRC should "codify [its] safety-focused principles as part of the rules themselves." Sen. George Voinovich (R-Ohio), chair of the panel holding the hearing, said he was concerned whether NRC had the resources and personnel to do its job and asked the NRC commissioners to send him a letter explaining how the agency would be impacted if Congress didn't approve a fiscal 2005 spending bill this year. In that case, NRC's spending would remain flat until a new funding measure was enacted. (*NUCLEAR NEWS FLASHES - Thursday, May 20, 2004*)

--NRC IS THE "VERY LAST" AGENCY THAT SHOULD BE OPERATING WITH VACANCIES because of its role overseeing the safety of nuclear power plants, Sen. Harry Reid (D-Nev.) said today in an explanation of his decision to hold up certain Senate business. Reid has vowed to block all votes on action pending before the Senate Environment & Public Works Committee until the committee holds a confirmation hearing for NRC nominee Gregory Jaczko, one of his staffers. Reid asserted the committee should not wait until President Bush names a candidate to fill a second opening on the commission. Committee Chairman James Inhofe (R-Okla.) indicated that the NRC, which has been without a full five-member complement since last July, may end up continuing to have only three commissioners for some time. (*NUCLEAR NEWS FLASHES - Thursday, May 20, 2004*)

NRC SHOULD HAVE IDENTIFIED OR PREVENTED THE DAVIS-BESSE REACTOR HEAD corrosion but failed to do so because the agency's oversight "did not generate accurate information on plant conditions," the General Accounting Office (GAO) said in a report publicly released today. The GAO, which is the investigative arm of Congress, said "NRC's process for deciding to allow Davis-Besse to delay its shutdown lacks credibility," in part because NRC does not have a specific guidance for that procedure. The GAO recommended that NRC develop "specific guidance and a well-defined process for deciding on when to shut down a nuclear power plant." NRC spokesman Scott Burnell said the agency "very strongly disagreed" with that approach; the necessary shutdown criteria already exist in regulations and plant technical specifications, he said. ***The GAO also recommended that NRC develop a methodology to assess a plant's safety culture, an idea that NRC previously considered and rejected.*** The report is available at (www.gao.gov), under "Reports." (*NUCLEAR NEWS FLASHES - Tuesday, May 18, 2004*)

--REP. EDWARD MARKEY QUESTIONED WHETHER NRC IS DOING ENOUGH TO MONITOR THE INTEGRITY OF STEAM GENERATOR TUBES. In a letter sent to NRC, Markey (D- Mass.) said only 31 of the 69 operational PWRs had replaced their steam generators as of July 2002, with nine of those replacements not using the more crack-resistant alloy 690. He asked NRC to respond by June 30 with a list of all reactors that have had deferred steam generator tube inspections over the past 10 years and the names of those not in compliance with the tube cracking limits and plugging requirements. Markey also wants to know how many hours are dedicated to steam generation inspections in the baseline program and how many reactors had more than the baseline effort. N (*NUCLEAR NEWS FLASHES - May 27, 2004*)

--DOE WILL HELP FUND A STUDY ON THE COSTS OF BUILDING A TWO-UNIT ABWR ON TENNESSEE VALLEY AUTHORITY'S (TVA) BELLEFONTE SITE IN ALABAMA, the department announced today. The study is expected to cost $4.25 million--to be split roughly 50-50 between DOE and a TVA-led team--and take 10 months to complete. The site hosts two unfinished reactors. DOE spokeswoman Hope Williams could not give the status of requests from two other consortia seeking to cost-share the expenses of developing a combined construction- operating license application for NRC. Energy Secretary Spencer Abraham called the TVA project an "important step" in considering new nuclear construction. (*NUCLEAR NEWS FLASHES - May 24, 2004*)

Catawba steam generator request to be forerunner of industry changes

Inside NRC
Volume 26 / Number 10 / May 17, 2004

Agreement may be near on license amendments for Duke Power Co.'s Catawba-1 and -2 that would pave the way for fundamental reform of how NRC regulates steam generators, NRC and industry officials said last week.

NRC staff met on May 14 with Nuclear Energy Institute (NEI) representatives to continue discussions of outstanding technical issues related to NRC's review of license amendments that would incorporate NEI steam generator program guidelines (NEI 97-06) into technical specifications for Catawba-1 and -2. If approved, these amendments would for the first time codify industry's performance-based criteria for steam generators into regulations, and allow longer intervals between inspections of some of Catawba's steam generators. Duke submitted its license amendment request in February 2003 for Catawba, which was selected by industry as the lead plant for it generic license change package initiative because it "provided a good sample, with examples of different kinds of steam generator tubing" and "a good cross section of materials," NEI senior project manager Jim Riley said.

The May 14 meeting, which took place at press time, was convened to address two unresolved technical issues related to the amendment request. These issues have been detailed by staff in several requests for additional information, the most recent sent to Duke on March 24. An NRC staffer said last week that the meeting should help Duke address NRC's questions and "move toward resolution of the remaining issues."

Riley also expressed optimism that the meeting would lead to expeditious closure of those issues. That would clear the way for Duke to revise and resubmit its license amendment to NRC in July for final approval.

After the Catawba license amendment is finalized and approved by NRC, other PWR licensees would be able to use a revised generic license change package incorporating the Catawba results as a template for their plants' license amendment applications. NEI submitted a Technical Specification Task Force (TSTF) "traveler" in March 2003. Travelers are applications to amend generically NRC technical specification requirements for designs used in a number of different plants. If approved, the revised technical specs can then be incorporated into a specific unit's license with an amendment based on the generic review.

Staff consideration of the TSTF traveler is on hold pending conclusion of the Catawba license amendment review, an NRC staffer said May 7.

Steam generator outages declining

The Electric Power Research Institute (EPRI) maintains a comprehensive steam generator database for the industry that shows that forced outages for steam generator repairs have declined significantly from about 10 per year in the 1980s to about one annually between 1995 and 2002. Nonetheless, efforts to assure steam generator tube integrity remain a significant priority of both industry and regulators. "These tubes have an important safety role because they constitute one of the primary barriers between the radioactive and non-radioactive sides of the plant," NRC said in a January 2004 fact sheet.

In 1997, PWR owners approved the guideline NEI 97-06, which defines three sets of steam generator performance criteria related to structural integrity, accident-induced leakage, and structural leakage. The industry guidelines have been implemented at all 69 U.S. PWRs, NEI's Riley said.
In October 1998, NRC and industry launched a joint initiative to incorporate NEI 97-06 and associated EPRI guidelines into NRC technical specification requirements for PWRs. Regulation based on the NEI guidelines, Riley said, would take advantage of more modern inspection technologies and approaches. Steam generator tubes made of "older materials," particularly mill-annealed alloy 600, "have been causing almost all the problems," Riley said.

NRC data indicate that as of December 2003, 25 U.S. PWRs have steam generators using this alloy. Steam generators in 17 PWRs are made from thermally-treated alloy 600 and 27 PWR generators use thermally-treated alloy 690. Both of these alloys offer improved corrosion resistance, NEI's Riley said.

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Regulation based on NEI guidelines would, among other revisions, "allow for different amounts of time between inspections" based on "the tube's material and condition," Riley said. Under a regulatory framework implementing the NEI guidelines, plants using thermally treated alloy 600 and alloy 690 "will not need to request approval for extending the operating interval between inspections since the new framework reflects the improved performance of these materials. As a result, the staff will no longer be required to review many of these requests," NRC said in its January 2004 fact sheet.

The NEI guidelines also require "monitoring and maintaining the tubes to provide reasonable assurance that the performance criteria are met at all times between scheduled inspections of the tubes," Gary Peterson of Duke said in a Feb. 25, 2003 letter to NRC.—*Steven Dolley, Washington*

Operators asked to monitor parts that get loose in steam generators

Inside NRC
Volume 26 / Number 10 / May 17, 2004

NRC's Office of Nuclear Reactor Regulation (NRR) issued an information notice (IN 2004-10) May 4 to alert plant operators of recent instances of loose parts found in steam generators at four of the nation's 69 PWRs.

Steam generators contain thousands of tubes to convert water into steam and are closed systems while the plant is in operation. However, items as welding slag, metal bars, and probes have on occasion become loose and cause generator tube wear or damage.

Jim Riley, senior project manager at the Nuclear Energy Institute, said May 12 that there are various reasons why parts get loose in steam generators. Sometimes during maintenance, "someone loses track of a tool or part which falls within the secondary system and is swept into the steam generator," Riley said.

NRC and industry have been aware of the loose parts issue for many years. An NRC information notice issued in April 1983 described loose parts associated with tube rupture events at Ginna and Prairie Island-1 and loose parts incidents at Wolf Creek, Watts Bar-1, Zion-1, San Onofre-1, Turkey Point-4, Cook-1, and Point Beach. The 1983 notice said that "loose parts have been implicated in two of the four tube rupture events in operating plants" and had "resulted in tube damage" in "at least two plants."

The May 4 information notice discussed instances in 2002 and 2003 where loose parts were detected in steam generators at Braidwood-2, Byron-2, Prairie Island-2, and Wolf Creek. Loose parts caused tube wear at Braidwood-2 and a tube leak at Byron-2. No damage attributable to the loose parts was found at Prairie Island-2 or Wolf Creek, but Wolf Creek had to shut down temporarily to retrieve the parts. "Recent operating experience of most plants indicates that loose parts have not significantly affected tube integrity; however, they have resulted in tube degradation," NRR said in the notice.

Riley emphasized that "it is important to distinguish a small amount of tube leakage from a wear scar caused by loose part from a tube integrity issue."

Though not yet required by NRC regulations, NEI's steam generator program guidelines, most recently updated in 2001, have been voluntarily implemented at all U.S. PWRs (see story, page 3).

The guidelines specify that licensees must "have procedures to monitor for loose parts and control of foreign objects to inhibit fretting and wear degradation of the tubing," including inspections and "procedures to preclude the introduction of foreign objects."

An NRC staffer said May 10 that "for the most part, plant programs have been effective in maintaining tube integrity from loose parts." There have been some tube leaks, the staffer said, but in general the loose parts issue "has not been safety significant."

Riley said that "operators find these things before they get to the point of becoming a threat to tube integrity." The information notice pointed out that "many licensees are beginning to extend the operating interval between tube inspections, especially at plants with advanced tube materials." As a result, "it is important to ensure that programs continue to effectively limit the introduction of loose parts, promptly detect loose parts that do enter, and implement appropriate corrective action upon identification of loose parts in steam generators," the notice said.

The notice does not require licensees to take any formal action.—*Steven Dolley, Washington*

NRC staff makes change in PRA plan to aid, not punish, proactive licensees

Inside NRC
Volume 26 / Number 10 / May 17, 2004

The NRC staff has accepted criticism from industry and the Advisory Committee on Reactor Safeguards (ACRS) and has revised a draft plan on a phased approach to the quality of probabilistic risk assessments (PRA) to eliminate the possibility that licensees with proactive PRA policies might be penalized during some license application reviews.

An earlier version of the plan, which implemented commission direction on a phased approach to PRA quality, would have assigned to a low priority review status to license amendment requests that relied on PRAs with a scope that went beyond existing NRC guidance endorsing industry PRA consensus standards. Currently, NRC has only endorsed standards dealing with how PRAs handle internal events. It has begun reviewing American Nuclear Society's standard for handling external events.

In an April 27 letter, the ACRS said that while it understood the staff's concerns that such review would be resource intensive (in the absence of standards), "proactive licensees should not be discouraged from pushing the boundaries of the state of the practice. In fact, the development of guidance documents at a later time will rely heavily on the work of these licensees."

At a meeting May 13, the staff told industry representatives that it would now be developing a review prioritization process as one of the early tasks in implementing the PRA plan after it is approved by the commission. This process, which would set the schedule for staff reviews of licensee risk-informed submittals, would take into account such things as staff resources, the safety benefit of the application, any economic benefit, whether the application is furthering the state of the practice, and whether the application is a pilot for an application seen as a net safety benefit. In its letter, the ACRS was generally positive about the staff's plan for a phased approach to PRA quality, calling it a "practical strategy." The ACRS, however, also said that development of guidance on how to perform sensitivity and uncertainty analyses "should receive a higher priority in the draft plan."

The ACRS also said that NRC, in its guidance documents, should be providing "sufficient guidance" so that those conducting peer reviews of licensee PRAs "will be aware of what the agency expects in the area of technical adequacy." A final plan on the phased approach to PRA quality is due to the commission by June 30.

At the meeting, the NRC staff, followed by industry representatives, discussed their respective efforts to come up with guidelines for how model uncertainties should be handled by those using PRAs in decision-making. Mary Drouin of the Office of Nuclear Regulatory Research (RES) said that the agency is working to develop by the end of the year a draft version of a Nureg-type document on model uncertainties. She said that much of the early effort has been devoted to modifying work done for a future reactor framework to use with current generation LWRs. She said she hopes to schedule a public meeting in June to provide the public with a first look at the effort.

Industry, through the efforts of the Westinghouse Owners Group and the Electric Power Research Institute, are also working on PRA-related uncertainty issues. Out of these projects, industry hopes to develop a document on handling uncertainty that would be endorsed by NRC. Drouin said she saw the NRC and industry efforts as complementary. Where the NRC document stopped, the industry document could pick up, she said.—*Michael Knapik, Washington*

United States
Nuclear Regulatory Commission

RISK-INFORMED PART 50 SPECIAL TREATMENT REQUIREMENTS 10 CFR 50.69

ACRS FULL COMMITTEE 513th MEETING JUNE 2, 2004

Timothy Reed
Office of Nuclear Reactor Regulation
US Nuclear Regulatory Commission

(W)

BRIEFING OBJECTIVE

- To brief the Committee on 10 CFR 50.69 final rulemaking and gain the Committee's endorsement on the final rule (requesting a letter)

- Focus of the discussion will be:
  - Changes from February 2004 subcommittee briefing
  - Changes to the package provided to ACRS on May 17, 2004
  - RG 1.201 (remaining issues)

INTRODUCTION/STATUS

- Briefed ACRS Subcommittee on Reliability and Probabilistic Risk Assessment on February 19, 2004

- February 2004 briefing focused on public comment resolution + NEI 00-04 review status

- Clarifications to rule in response to public comments implemented into final rulemaking package as discussed at February 2004 briefing

NOTEWORTHY CHANGE TO RM PACKAGE

- Applicants for Part 52 Design Certifications removed from 50.69:
  - Difficult issues identified with Design Certification applications - (e.g., how to allow SSCs to change RISC categories over time per 50.69 monitoring/feedback vs Part 52 restrictions on changes)
  - Part 52 COL applicants can still reference certified designs and apply 50.69

ONGOING TASKS TO ISSUE 50.69

- **Review/concurrence process for final rulemaking process**
  - **meet with CRGR (mid June)**
  - **deliver final rulemaking package to EDO - 6/23**
  - **deliver final rulemaking package to Commission - 6/30**

- **RG 1.201 issue for trial use - update/revise with lessons learned from pilots**

Donnie Harrison, NRR

Regulatory Guide 1.201

Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance

RG 1.201

Endorsement of NEI 00-04

- Received NEI 00-04 Final Draft in April 2004
  - Incorporated a Number of Changes Due to Staff Comments on Revision C (DG-1121) and Revision D
  - Many Staff Positions in DG-1121 Addressed Directly or Staff Position has Changed with Better Understanding of the NEI 00-04 Approach
- Staff Provided Comments on NEI 00-04 in May 2004
  - Comments are captured in RG 1.201 Positions
- NEI Addressing Staff Comments

RG 1.201

Key General Comments

- Implementation Limitations Depend on Types of Analyses Used in Categorization
- PRA Quality Attributes
- Uncertainty Considerations
- Common Cause Failure/Degradation Mechanism Considerations

RG 1.201

Key Specific Comment Topics

- Interpretations/Clarifications
  - Many carry-over from DG-1121
- Regulatory/Legal Clarifications
- Technical Clarifications
  - SSCs Screened out of Seismic PRA Due to Seismic Robustness
  - Addressing Self-Assessment Findings
  - IDP considerations
- Technical Objections
  - RISC-3 SSC Reliability Reduction Factor Assumed in Risk Sensitivity Study (Implementation Issue)

RG 1.201

IDP Considerations

- Rule Requires IDP be Staffed with Expert, Plant-Knowledgeable Members
- NEI 00-04 Section 9.1 Provides Additional Information on Panel Makeup and Training
  - Licensees Establish Specific Requirements to Ensure and Maintain Adequate Expertise of IDP Memebers
  - Key Areas of Emphasis are Experience at Specific Plant and Experience with Plant-Specific Risk Information
  - IDP described in Formal Plant Procedure, including Training and Qualifications of Members
- At ASME/ANS Joint Meeting IDP Qualifications Identified as an Action Item for Future Consideration

RG 1.201

Rationale for Issuing For Trial Use

- Remains 1 Technical Issue
- Some Supporting Guidance Documents Still Not Finalized
  - N-660 Guidance
- Though Staff Does NOT Expect to Need to Make Major Changes, Will Learn from Early Implementation/Pilots

RG 1.201

Summary

- NEI is Addressing Latest Set of Staff Comments
  - Only 1 Technical Objection Remains (which is more an Implementation Issue than a Categorization Issue)

- Staff will Continue to Work with NEI to Address Staff Comments and Develop Final Version of NEI 00-04 that is Endorseable with Few, if Any, Conditions/Exceptions

- Staff will Continue to Work with NEI During Early Implementation of Rule to Refine Guidance
  - Develop Submittal Template

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Improved License Renewal Application Review Process

Presented to
ACRS
June 2, 2004

Kurt Cozens, Senior Materials Engineer
License Renewal Section B
License Renewal and Environmental Impacts Program
U. S. Nuclear Regulatory Commission

Objective

- To provide an overview of the improved LRA review process
  - Why change
  - What's changed and what's not
  - Project team's audit and review process

Benefits of change

- Leverage resources to perform simultaneous reviews
- Take advantage of efficiencies inherent in the license renewal guidance documents

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What's Changed and What's Not Changed

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AMP & AMR assignments

- RLEP-B reviews AMPs and AMRs that are:
  - Consistent with GALL Report
  - Based on NRC approved precedents
- DE reviews
  - Remaining AMPs
  - Emerging issues

Typical work splits

Plant	AMPs Assigned (%)		AMRs Assigned (%)	
	DE	RLEP-B	DE	RLEP-B
ANO2, Unit 2	18	82	37	63
DC Cook, Units 1 & 2	23	77	28	72
Point Beach, Units 1 & 2	17	83	17	83

Team's LRA review process

- Assemble a team
- Prepare a plan
- Prepare for the reviews
- Perform reviews
  - GALL Report
  - SRP-LR
- Document the results of the reviews
  - Audit and review report
  - SER input
- Feedback lessons learned

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Use of questions versus RAIs

- Site visits permit use of questions to obtain clarification
  - Applicant may chose voluntary docketed submittal, if warranted
  - Use of questions during site visits integral part of front loaded schedule
- RAIs used when docketed responses necessary

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What's involved in the AMR reviews?

- AMR types
  - Consistent with GALL
  - Consistent, but require further evaluation
  - Based on NRC approved precedents
- Initial AMR reviews may be performed in office
  - Complete AMR reviews, where possible
  - Develop questions
- Site visits
  - Resolve questions

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Documentation

- Audit and review report
  - Document AMP and AMR audits and reviews
  - Majority of contents transferred to SER input
- SER input
  - Sections 3.0 – 3.6 of SER
  - Addresses RAIs

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Updating License Renewal Guidance Documents

Jerry Dozier

License Renewal and Environmental Impacts - B

License Renewal Guidance Documents

- NUREG-1800, *Standard Review Plan for License Renewal Applications for Nuclear Power Plants*
- NUREG-1801, *Generic Aging Lessons Learned* (GALL) *Report*
- RG 1.188, *Standard Format and Content for Applications to Renew Nuclear power Plant Operating Licenses*

Objectives

- Incorporate lessons learned to make GALL a better document
- Increase applicant and review efficiency

Activity

- The GALL update will involve component consolidations, reformatting, correction of errors, and incorporation of approved staff positions (Precedents, Interim staff guidance)
- Corresponding changes to the SRP-LR will also be developed including update to incorporate new review approach.

Schedule

- A preliminary draft GALL and SRP will be available on the Web by 9/30/04
- Final GALL, SRP, and Bases document expected to be ready for use in early 2006

ACRS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION

JUNE 2, 2004

2

OVERVIEW

Mario V. Bonaca
ACRS Chairman

LICENSE RENEWAL

- **Streamlined ACRS review of license renewal applications**
- **Reviewed three applications since October 2003 and plan to review another three during the remainder of CY 2004**

LICENSE RENEWAL (Cont'd)

- **Will review six applications in CY 2005**
- **Will review updates to Generic License Renewal Guidance documents (SRP, GALL, and Reg. Guide)**

10 CFR 50.69

- **Held a Subcommittee meeting in February 2004 to discuss:**
  - **Resolution of public and ACRS comments on the proposed 10 CFR 50.69**
  - **NEI implementation guidance document, Revision D**
- **Plan to review the draft final 10 CFR 50.69 in June 2004**

ACR-700 DESIGN

- **Held a Subcommittee meeting with AECL representatives and staff in January 2004 to discuss the ACR-700 design**
- **Plan to review the staff's Safety Assessment Report**
- **Plan to tour the Chalk River facility**

EARLY SITE PERMIT APPLICATIONS

- Plan to review staff's SERs on ESP applications
- Anticipate review of one SER in late CY 2004

FUTURE ACTIVITIES

- Risk-Informed and Performance-Based Regulation
- Materials and Metallurgy
- Advanced Reactor Designs
- Resolution of GSIs
- Revisions to SRP
- High-Burnup Fuel Issues

FUTURE ACTIVITIES (Cont'd)

- Use of MOX Fuel in Commercial Reactors
- Safeguards and Security Matters
- Assessment of Research Quality
- Core Power Uprates
- License Renewal Applications

FUTURE ACTIVITIES (Cont'd)

- **Fire Protection**
- **Human Factors and Human Reliability Assessment**
- **Operating Plant Issues**

PWR SUMP PERFORMANCE

John D. Sieber

PRECURSOR EVENTS

- TMI-2
- Perry-1 (two events)
- Limerick (two events)
- Barsebäck Event

TECHNICAL ISSUES

- **Debris Generation**
  - **Break size**
  - **Zone of influence**
  - **Materials**
- **Debris Transport**
  - **Analytical methods**
  - **Debris interception**

TECHNICAL ISSUES (Cont'd)

- Head Loss
  - Screen sizing
- Chemical Effects

ACRS ISSUES

- Limitations of the present knowledge base
- Maturity of the technical content of RG 1.82, Rev. 3
- Adequacy of industry guidance
- Use of risk information

ACRS ISSUES (Cont'd)

- Alternative solutions, if uncertainties are too large
- Need for additional research

PRA QUALITY FOR DECISIONMAKING

George E. Apostolakis

PRA QUALITY FOR DECISIONMAKING

- The NRC staff has developed a practical strategy that would encourage the development of guidance documents necessary to implement the Commission's phased approach to PRA quality

PRA QUALITY (Cont'd)

- **The phased approach is contingent on the availability of guidance documents (i.e., consensus standards and regulatory guides)**

PRA QUALITY (Cont'd)

- **The staff should be prepared to develop guidance documents independently, if consensus standards are not developed in a timely manner to meet the Commission's deadline for achieving Phase 3**

PRA QUALITY (Cont'd)

- **It is more appropriate to refer to the technical adequacy of PRA for a specific regulatory decision rather than its quality**

PRA QUALITY (Cont'd)

- **An application that uses a PRA scope greater than that for which guidance documents exist should not be given low-priority staff review**
- **Proactive licensees should not be discouraged from pushing the boundaries of the state of the practice**

PRA QUALITY (Cont'd)

- Licensees should be encouraged to address in their application the relevant technical issues, as discussed in the December 18, 2003 SRM

- The staff should give high priority to these reviews

PRA QUALITY (Cont'd)

- Development of guidance on how to perform sensitivity and uncertainty analyses should receive a higher priority in the action plan

RISK-INFORMING
10 CFR 50.46

William J. Shack

Risk-Informing 10 CFR 50.46

- The risk-informed revision to 10 CFR 50.46 should permit a wide range of applications

- RG 1.174 is appropriate for evaluating the acceptability of changes proposed under a revised rule

RISK-INFORMING 10 CFR 50.46 (Cont'd)

- **Explicit criteria for mitigative capability should be developed to ensure that sufficient defense-in-depth is maintained as plant changes are made**

RISK-INFORMING 10 CFR 50.46 (Cont'd)

- The appropriate metric for the design basis maximum break size is the LOCA initiating event frequency

- It is possible and desirable to make generic definitions of maximum break size applicable to categories of plants

RISK-INFORMING
10 CFR 50.46 (Cont'd)

- **The number and kind of plant changes allowable will depend on the scope and technical detail of the licensee's PRA**

RISK-INFORMING 10 CFR 50.46 (Cont'd)

- **If a limited scope PRA is used, contributions to the total risk and the change in risk from the omitted portions of the PRA must be estimated**
- **A convincing demonstration that the resulting changes in risk are small enough is needed**

RISK-INFORMING
10 CFR 50.46 (Cont'd)

- The results of the expert elicitation for the frequency of LOCA events are not yet final
- The results need to be peer reviewed
- The process is well structured and the expert panel has an appropriate range of expertise

RISK-INFORMING 10 CFR 50.46 (Cont'd)

- The results will help provide a technical basis for the selection of the maximum break size

- Will review the draft final NUREG report on LOCA frequencies

ACRS 2004 REPORT ON NRC SAFETY RESEARCH PROGRAM

Dana A. Powers

SCOPE

- **Research projects dealing with the safety of existing plants**

CONSIDERATIONS

- Programmatic justification
- Technical approach
- Progress of the work

GENERAL OBSERVATIONS

- **NRC has a well-focused, well-planned Safety Research Program**
- **Research effort may well be near the minimum needed to support regulatory activities**
- **Resources for exploratory research are minimal and may limit the agency's ability to anticipate future needs**

HIGHLIGHTS

- **High-burnup fuel research for reactivity-initiated accidents**
- **PRA research in support of ROP**
- **Rejuvenation of human factors research**
- **Realism in severe accident analysis**

SOME PROJECTS HAVE ACHIEVED SUCCESS

- **Realistic structural capacity of existing reactor containments**
- **Seismic engineering of existing reactors**

EXPERTISE BEING MAINTAINED

- Neutronic Analysis
- Criticality Safety
- Radiation Effects
- Reactor Fuels

ADDITIONAL EFFORTS NEEDED

- Independent analysis and evaluation of operational data
- Fire safety research
- PWR sump blockage issue
- Integration of TRACE code into regulatory process

ADDITIONAL PLANNING NEEDED

- **ACRS supports plans to examine the utility of a proactive materials degradation initiative at NRC**

- **RES should examine activities in pressure vessel embrittlement**

QUALITY OF RESEARCH PROGRAMS

- RES is required to have an independent evaluation of the quality of its research programs
- At the request of RES, the ACRS has agreed to this major undertaking

ESBWR PRE-APPLICATION REVIEW

Thomas S. Kress

BACKGROUND

- **Analytical methods and supporting experimental data for LBLOCA and containment scenarios**
- **Based on previous work done for SBWR**

CONCLUSION AND OBSERVATIONS

- TRACG computer code is acceptable for analyzing ESBWR response to a LOCA scenario
- Large design margins (core never uncovers)
- Many conservative assumptions

FUTURE USE OF TRACG

- Code application and input assumptions for licensing
- Degree of conservatism
- Sources of margin
- Scaling evaluation assessment
- Vacuum breaker performance

INTERIM REVIEW OF THE AP1000 DESIGN

Thomas S. Kress

THE AP1000 DESIGN

- **Completed the Phase-2 pre-application review -- ACRS report dated March 14, 2002 concluded that:**
  - **The staff has made a competent and thorough review of the Phase-2 issues**
  - **ACRS agrees that the proposal by Westinghouse to use Design Acceptance Criteria (DAC) for the piping should be approved**

AP1000 (Cont'd)

- On March 17, 2004, the ACRS issued an interim letter to the EDO commenting on several technical areas

- Will hold further discussions with the Staff and Westinghouse in June 2004

AP1000 (Cont'd)

- ACRS reviews have not addressed security matters related to the design

- ACRS will review FSER in July 2004

ACRONYMS

ACR	Advanced CANDU Reactor
ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic Depressurization System
AECL	Atomic Energy of Canada Limited
CFR	Code of Federal Regulations
CY	Calendar Year
DAC	Design Acceptance Criteria
DSER	Draft Safety Evaluation Report
EDO	Executive Director for Operations
ESBWR	Economic Simplified Boiling Water Reactor
ESP	Early Site Permit
FSER	Final Safety Evaluation Report
FCI	Fuel-Coolant Interaction
GALL	Generic Aging Lessons Learned Report
I&C	Instrumentation and Control

ACRONYMS (Cont'd)

LBLOCA	Large break Loss-of- Coolant Accident
MOX	Mixed Oxide
NRC	Nuclear Regulatory Commission
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
ROP	Reactor Oversight Process
SBWR	Simplified Boiling Water Reactor
SER	Safety Evaluation Report
SRM	Staff Requirements Memorandum
SRP	Standard Review Plan
TMI	Three Mile Island

Overview of NRC Digital I&C Research Program in Digital Systems Reliability

Advisory Committee on Reactor Safeguards

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Office of Nuclear Regulatory Research
June 2, 2004

(8)

- On March 26th RES briefed the ACRS Subcommittee on Plant Operations on the NRC digital systems reliability research program
- This presentation provides an overview of the information presented to the subcommittee

OVERVIEW

- Conclusions
- Review of digital I&C research program
- Drivers and boundary conditions
- Digital systems reliability program
- Research projects
- Interfaces
- Summary

CONCLUSIONS

- US nuclear industry is moving forward to retrofit digital systems into control, monitoring, and protection systems throughout the plants. The NRC research efforts will provide tools, methods, and guidance to support reviews in this area.
- A part of the RES instrumentation and control research program is devoted to supporting the research needed to develop digital systems risk and reliability information, models and guidance.
- Research at several universities, and national laboratories supplemented by in-house efforts are part of an integrated program to develop tools and methods to evaluate these digital systems and to support the development of regulatory guidance.

DIGITAL I&C RESEARCH PROGRAM

- SECY-01-0155, NRC Digital Instrumentation and Control Research Plan, published in August 2001
  - Established research objectives and program areas
  - Reviewed and endorsed by ACRS
  - Contained four main program areas including digital systems reliability
- Since SECY-01-0155, additional areas have been added including future reactors
- Revision to research plan is currently under development (available in forth quarter of FY 2004)

I&C RESEARCH PROGRAM GOAL

"Continually improving the staff's analytical capabilities, and fundamental knowledge of I&C technology as demonstrated by the development of analytical tools, technical reports, regulatory guidance, papers and articles, and interaction with licensees, vendors, research organizations, and the public."

PROGRAM EXTERNAL DRIVERS

- National Academy of Sciences and Engineering/National Research Council (NAS) report recommendations, 1997
- As new technology has become available, NRC licensees have been moving to more modern control and protection systems
- New systems have different failure modes and are more difficult to analyze.
- NRR User Need, provided in March, 2000 and Reaffirmed in July, 2002
- DOE I&C and HMI Working Group Recommendations, and Halden Workshop on Digital System Reliability, 2002
- EPRI D3 Draft Topical, 2004

I&C RESEARCH PROGRAM AREAS

- System aspects of digital technology
- Software quality assurance
- Digital systems reliability
- Emerging I&C technologies and applications
- Future Reactors

NAS REPORT RECOMMENDATIONS

- "Include the relative influence of software failures on system reliability in PRAs"
- "Develop methods for estimating digital system failure probabilities, including COTS. Include acceptance criteria, guidelines, limitations, rationale and justifications"
- "Develop advanced techniques for analyzing digital systems to increase confidence and reduce uncertainty in quantitative assessments"
- "NRC and industry should evaluate their capabilities and develop a sufficient level of expertise to understand the requirements for gaining confidence in digital implementations of system functions and the limitations of quantitative assessments"

WHAT IS NEEDED IN DIGITAL SYSTEMS PRAs

- Develop methods for reviewing digital system reliability models
  - Understanding the state of the data
  - Digital system failure mechanisms
  - Strengths and limitations of digital system models
  - Incorporating digital system models into PRAs
  - Acceptance criteria (quality of modeling, etc.)

DIGITAL SYSTEMS RELIABILITY MODELING

- Modeling of digital systems should be realistic to:
  - Account for the most important system characteristics
  - Model important failure modes
  - Be able to predict system behavior (including system reliability)
  - Be consistent with available data
- Modeling issues
  - Availability of reliability data
  - Level of detail of the models
  - Independence of hardware and software
  - Software diversity
  - Number of possible states and the ability to test

DIGITAL SYSTEMS RELIABILITY PROGRAM

- An integrated program plan is being developed for the digital systems reliability program within the instrumentation and control research program (will be available in fourth quarter of FY 2004)
- The program consists of three elements
  - Development of quantitative digital systems models that can be used to determine digital system failure modes and reliability estimates
  - Development of methods for integration of reliability models that are capable of modeling digital systems (Dynamic Fault trees, Markov, Dynamic flow graph, Petri nets, etc.) into current PRAs to support risk-informed regulatory applications
  - Establishment of regulatory guidance, including modeling quality and acceptance criteria

DIGITAL SYSTEMS RELIABILITY PROGRAM (CONT.)

- Development of quantitative digital systems models that can be used to determine digital system failure modes and reliability estimates
  - This part of the program includes the development of both models for software and integrated hardware-software systems
  - The models will be first evaluated on their ability to provide meaningful quantitative information on digital system performance
  - The models will then be modified or adapted to provide information needed for risk and reliability assessments

DIGITAL SYSTEMS RELIABILITY PROGRAM (CONT.)

- Development of methods for integration of reliability models that are capable of modeling digital systems (Dynamic Fault trees, Markov, Dynamic flow graph, Petri nets, etc.) into current generation PRAs to support risk-informed regulatory reviews
  - Both traditional methods (FMECA, etc) for developing fault tree models and dynamic modeling methods will be evaluated
  - Integration methods will be developed
  - All new models and methods will be piloted

DIGITAL SYSTEMS RELIABILITY PROGRAM (CONT.)

- Establishment of regulatory guidance, including modeling quality and acceptance criteria
  - NRC position on what is acceptable as the "default" level of analysis
  - Which of the various analysis methods are acceptable
  - How software and its hardware context needs to be modeled
  - How much and what kind of data is needed to support reliability models
  - The acceptable level at which component failures should be modeled
  - How the system operation profile is to be developed
  - Acceptance criteria for both the modeling fidelity and the system reliability

DIGITAL SYSTEMS RELIABILITY PROGRAM (CONT.)

- Implementation
  - All three parts of the program will proceed in parallel
  - Part one (digital system modeling) will be carried out by UVa, UMd, and Halden
  - Part two (PRA modeling and integration) will be carried out by BNL, Ohio State, and In-house efforts
  - Part three (Guidance Development) will be carried out primarily through in-house efforts

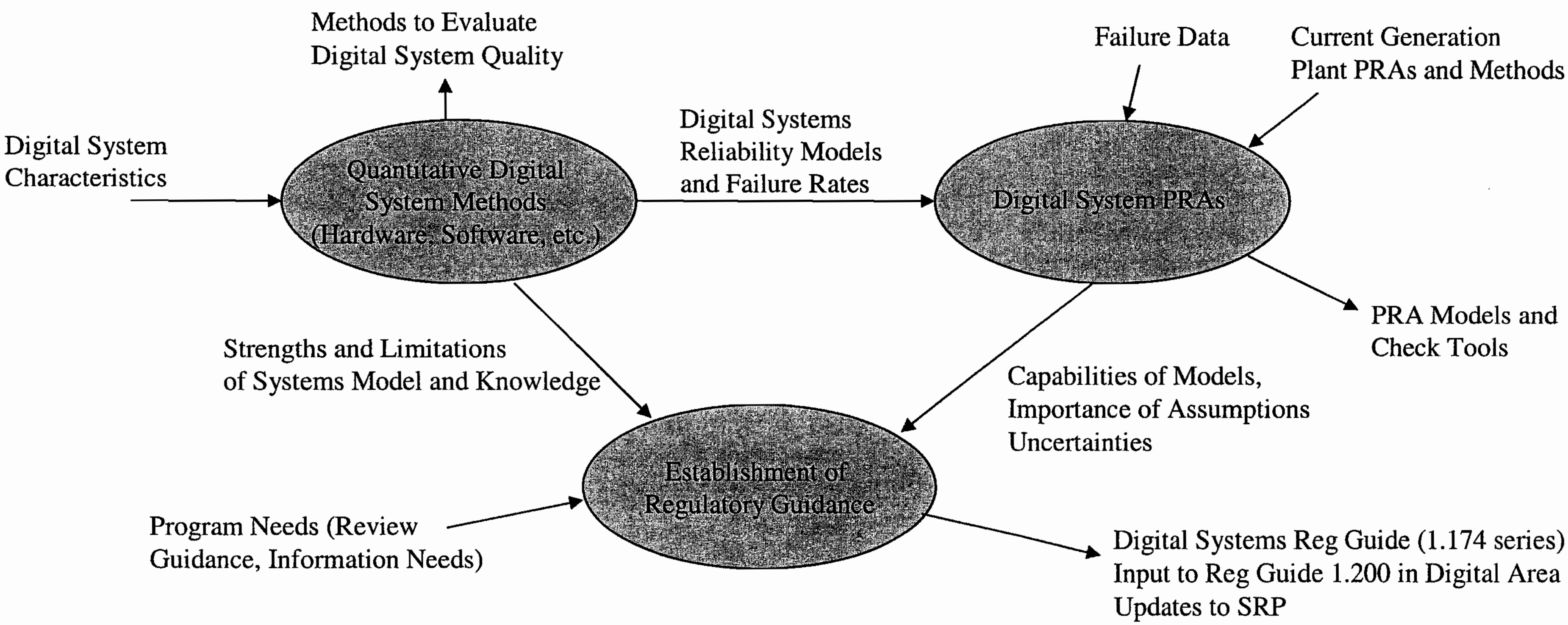

Diagram of Digital System Reliability Program

United States Nuclear Regulatory Commission

Primary issues	Approach	Resources
Digital systems have different failure modes, and are much more challenging to model. More quantitative methods are needed	Gain a better understanding of how these systems fail and how likely it is that they will fail in use. Develop models and methods for evaluating how digital system fail.	•University of Virginia (systems analysis) •Halden (requirements analysis) •University of Maryland (software)
Digital systems are being retrofitted into current generation of nuclear power plants and they need to be reviewed in a risk informed manner	Develop methods and tools for including digital system models into PRA a) Develop a better understanding of current data and generate application specific databases b) Develop new methods for integrating current methods of digital system modeling into current PRAs c) Pilot methods using plant-specific PRAs plants and validate the models using available data on digital system failures. d) Develop ways to estimate uncertainty in quantitative assessments	•Brookhaven National Lab (database review and evaluation, and pilot studies) •NRC staff (new methods, and pilot studies) •Ohio State (Dynamic PRA, uncertainty and pilot studies) •International cooperative programs (COMPSIS, etc.)
The NRC does not have guidance on what is acceptable and what is not in the modeling of digital system reliability	Develop guidance for regulatory applications involving digital systems reliability • acceptance criteria • limitations • evaluation methods • reliability data	•NRC staff •Brookhaven National Lab

RESEARCH PROJECTS

- UVa Integrated digital systems modeling project
- UMd software metrics project
- BNL project on digital system risk
- Dynamic reliability modeling project and PRA integration (Ohio State)
- Other research
  - Halden
  - COMPSIS
  - NRC In-house effort

STRUCTURE OF CURRENT NRC RESEARCH

- UVa integrated digital systems modeling project will provide:
  - An integrated digital system assessment method that can be used by the NRC staff to independently assess digital system safety
  - Information on digital systems failure modes and reliability that will inform the review guidance
  - Information and models that can be used in development of digital systems risk and reliability assessments

STRUCTURE OF CURRENT NRC RESEARCH (CONT.)

- UMd software metrics project will provide:
  - An assessment method that can be used by the NRC staff to independently assess software quality and reliability
  - Quantitative information on the relative importance of software metrics will be used to inform the current review guidance
  - Input to guidance on quantitative software quality and reliability

STRUCTURE OF CURRENT NRC RESEARCH (CONT.)

- BNL project on digital system risk will provide:
  - Draft interim review guidance for risk-informed digital submittals
  - Review current methods and tools for modeling digital systems that will be used in guidance for risk-informed digital submittals
  - Review of digital failure databases
  - Digital system PRA modeling using tradition methods

STRUCTURE OF CURRENT NRC RESEARCH (CONT.)

- Ohio State dynamic reliability modeling project will provide:
  - Methods for integration of one or more of the digital modeling into current generation PRAs
  - Pilot studies of methods using full scope PRA plants models
  - Methods for estimate uncertainty in quantitative assessments

STRUCTURE OF CURRENT NRC RESEARCH (CONT.)

- Halden
  - Development of digital systems requirements assessment tools and methods for integration of quantitative and qualitative information
- COMPSIS
  - International effort to develop a database of software failures in computer systems important to safety in nuclear plants, and the lessons learned from these failures
- NRC In-House Effort
  - Several areas including development of guidance, an NRC database of digital system failure information for use in validating reliability modeling assumptions, and work to support PRA integration

INTERFACES

- As part of the development of guidance in this area the digital systems reliability program will work with several other parts of the NRC and outside stake holders
  - NRR/EEIB and NRR/SPSB in the areas of PRA quality issues, revision of I&C guidance, and integration with other risk initiatives
  - EPRI and other research organizations
  - Other regulatory bodies (NASA, FAA, FRA, etc.)
- Public meetings and workshops will be held to gather input, build consensus and identify options

SUMMARY

- US nuclear industry is moving forward to retrofit digital systems into current generation nuclear power plants and the NRC is working to provide tools, methods, and guidance to support reviews in this area
- Digital systems reliability program consists of three elements
  - Development of quantitative digital systems models that can be used to determine digital system failure modes and reliability estimates
  - Development of methods for integration of reliability models that are capable of modeling digital systems (Dynamic Fault trees, Markov, Dynamic flow graph, Petri nets, etc.) into current generation PRAs to support risk informed regulatory reviews
  - Establishment of regulatory guidance, including modeling quality and acceptance criteria
- Research at several universities, and national laboratories supplemented by in-house efforts are part of an integrated program to develop tools and methods to evaluate of these systems and to support the development of regulatory guidance.

9.

Risk Analysis, Vol. 19, No. 1, 1999

Context in the Risk Assessment of Digital Systems

Chris Garrett[1] and George Apostolakis[1]

As the use of digital computers for instrumentation and control of safety-critical systems has increased, there has been a growing debate over the issue of whether probabilistic risk assessment techniques can be applied to these systems. This debate has centered on the issue of whether software failures can be modeled probabilistically. This paper describes a "context-based" approach to software risk assessment that explicitly recognizes the fact that the behavior of software is *not* probabilistic. The source of the perceived uncertainty in its behavior results from both the input to the software as well as the application and environment in which the software is operating. Failures occur as the result of encountering some context for which the software was not properly designed, as opposed to the software simply failing "randomly." The paper elaborates on the concept of "error-forcing context" as it applies to software. It also illustrates a methodology which utilizes event trees, fault trees, and the Dynamic Flowgraph Methodology (DFM) to identify "error-forcing contexts" for software in the form of fault tree prime implicants.

KEY WORDS: Software failures; software hazard analysis; safety-critical systems; risk assessment; context.

1. INTRODUCTION

Due to its usefulness in evaluating safety, identifying design deficiencies, and improving interactions with regulatory agencies, probabilistic risk assessment (PRA) is playing an increasing role in the design, operation, and management of safety-critical systems in the nuclear power, chemical process, and aerospace industries.(1,2) However, as the use of digital computers for instrumentation and control (I&C) of such systems has increased, there has been a growing debate over the issue of whether PRA techniques can be applied to digital systems.(3,4)

PRA is typically performed using fault tree analysis, often in combination with other methods such as event trees, reliability block diagrams, and Markov models. The purpose of these techniques is, almost universally, to calculate the probability of occurrence of various system-level events, based on the probabilities of occurrence of basic events. Naturally, these basic events are usually individual component failures. It therefore comes as no surprise that, when considering the application of PRA to digital systems, attention turns almost immediately to the question of software failure probabilities. A considerable amount of effort has been devoted over the years to developing various models for estimating software failure probabilities, and there is controversy within the software engineering community as to which of these models is most appropriate and, indeed, whether any of them are even meaningful at all.(4)

Unfortunately, all of the attention being focused on the issue of software failure probabilities is, in effect, putting the cart before the horse. Failure probabilities in a PRA context are only meaningful when they are linked to well-defined failure *modes*. Due to the complexity commonly associated with digital I&C systems, there is the potential for a number of

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software related failure modes, most of which are application specific and unknown. Trying to estimate the probability that the software will fail without first understanding in what *ways* the software may fail, as well as what effects its failure will have on the system, does not make sense. In fact, it is precisely this lack of knowledge of the potential software failure modes that lies at the root of the controversy surrounding the estimation of software failure probabilities. In order to calculate the likelihood of any event (software failure or otherwise), the event in question must be well defined.

The majority of unanticipated software failure modes fall under the umbrella of what are commonly referred to as "systems aspects" (i.e., issues that transcend the functions of individual components and involve interactions between components within the system as well as the interaction of the system with the environment). Systems aspects have been the focus of a great deal of concern over the use of digital I&C in safety-critical applications. For instance, in the aerospace industry, attention to systems aspects has identified a number of safety-related issues involving human–machine interaction, task allocation and levels of automation, such as operator confusion caused by automatic changes in operating modes.[4] Issues such as these must be well understood before considerations of software "failures" and their impacts on the system can have any real meaning.

Currently, there are no risk assessment models that are capable of characterizing software "failures" in an application-independent way. Indeed, it is not always clear what a software failure is. Therefore, we need to take a different approach in which the software is no longer a source of uncertainty. In most cases, software behavior is deterministic. The source of uncertainty is really in the physical situation with which the system is presented (i.e., the context, which forces the software to produce an undesired result. We need more detailed, deterministic models of the software's interaction with the system so that we can treat it properly as part of the PRA logic models.

This paper describes an approach that, instead of treating the software as a component that fails randomly, recognizes the fact that its behavior is deterministic. Whether or not it "fails" is determined by the particular situation at hand (i.e., the context). The idea is to identify in which situations the software *will* fail, and then deal with those situations. In Sect. 2, we argue that software reliability is an ill-defined metric for the purpose of risk assessment, and that the concept of a software "failure," considered apart from the particular context in which it occurs, has no meaning. In Sect. 3, we suggest a new "context-based" approach to software risk assessment, which explicitly accounts for the context dependence of software behavior. Section 4 outlines a methodology for implementing the context-based approach, and Sect. 5 presents an illustrative example. Section 6 contains some conclusions.

2. UNSUITABILITY OF THE CONCEPT OF SOFTWARE RELIABILITY

Simply stated, software reliability assessment and software risk assessment are entirely unrelated. Reliability assessment is concerned with the probability that the execution of the code will deviate from its specifications, whereas risk assessment is concerned with the probability that a software action will lead to the occurrence of a hazardous condition.

Software reliability has been defined as "the probability of failure-free software operation for a specified period of time in a specified environment."[5] For the purpose of risk assessment, this definition is of no practical use for two reasons. The first has to do with the fact that this is simply the standard definition for the reliability of any system component, cast specifically in terms of the software. Insofar as it is system aspects where the effects of software are most strongly felt, it seems misguided to treat the software as a component-level entity. Indeed, it is misleading to think of the software as simply a "component" of the system, in the sense in which pumps and valves are considered to be components (i.e., as physical devices which perform a specific function and whose performance may vary over time). In fact, software is quite the opposite. In general, the software may perform *many* functions, and it will perform each of them without variation. Given a particular set of inputs, it will *always* produce the same output. When we speak of software "failures," we are actually talking about unintended functionality that is *always* present in the system. In a very real sense, the software is more a reflection of the *design* of the system than it is a component of the system. As such, it is more appropriate to think in terms of failures of the system rather than failures of the software.

The second reason is the fact that what is meant by "failure-free operation" is ill defined. It is not clear whether software can indeed "fail," not simply because it is incapable of ceasing to perform a function which it used to provide, but, more fundamentally, because there is no appropriate criteria for judging its "correctness." It is not sufficient to simply compare the output of the software against the software specifications. It is well known that a large number of software errors can be traced back to errors in the requirements specification.[6] Thus, the ultimate arbiter of whether a specific software action constitutes a "failure" of the software can be nothing other than the observation of an undesired event in the system that occurs as a result.

For example, consider an incident which occurred when, to save development costs, Great Britain decided to adopt air traffic control software which had been in use in the USA.[7] Because the software had been designed for use in the United States, it had no provision to take into account the 0° longitude line. As a result, when it was installed into British air traffic control systems, it folded its map of Great Britain in half about the Greenwich Meridian.

Was this a software failure? The software did what it had been designed to do, so in that sense, it shouldn't be considered a failure. However, in the context of the system in which it was operating, it clearly performed the wrong action. Certainly, a failure did occur, but not on the part of the software alone. It was the application of the (nominally correct) software in that particular context that resulted in a failure of the *system*. This example illustrates the point that software actions can contribute to *system* failures, but the concept of a *software failure* has, by itself, little meaning. It is only the application of the software in a particular context that determines whether it is correct or incorrect.

Additionally, software can have many functions, each contributing to the operation of the system in very different ways. The failures of different functions will not, in general, have equivalent effects on the system. For instance, some failures may result in a catastrophic failure of the system, whereas others may only result in degraded performance. The definition of software reliability does not discriminate between different types of failures. However, in PRA, because we are concerned with the consequences of failure with respect to the system, we must be able to account for the differences. The first thing that is needed, then, in developing an approach for performing PRA for systems involving software, is to eschew attempts to evaluate the reliability of the software, and instead focus on the context of the system in which the software is applied.

3. CONTEXT-BASED APPROACH TO SOFTWARE RISK ASSESSMENT

System failures resulting from software are due to design errors (i.e., incorrect or incomplete requirements, inappropriate algorithms, and/or coding errors). In some cases, they may also be due to inappropriate use of the software in an application for which it was not designed. In any case, they are not due to "random" changes in the software. The software behavior is deterministic. However, it is misleading to say that the software is either *correct* or it is *incorrect*. In fact, the "correctness" of the software is context-dependent. It is correct for some situations and it is incorrect for other situations. The key to assessing the risk associated with the use of a particular piece of software is to identify which situations are "incorrect" for the software, and then evaluate the probability of being in one of those situations.

This is similar to the concept of "error-forcing context" recently proposed for human reliability analysis in nuclear power plant PRAs.[8] An error-forcing context represents the combined effect of human factors and plant conditions that create a situation in which unsafe acts are likely. The idea of error-forcing context is based on the theory that unsafe acts occur (for the most part) as a result of combinations of influences associated with the plant conditions and associated human factors issues that trigger error mechanisms in the plant personnel. In addition to plant conditions, such as sensor information, the context can include such things as working conditions, adequacy of man–machine interface, availability of procedures and time available for action.[9,10] Many error mechanisms occur when operators apply normally useful cognitive processes that, in the particular context, are "defeated" or "fooled" by a particular combination of plant conditions and result in human error.

This understanding has led to the belief that it is necessary to analyze both the human-centered factors (e.g., things such as human–machine interface design, procedure content and format, training, etc.) and the conditions of the plant that precipitated the inappropriate action (such as misleading indications, equipment unavailabilities, and other unusual configurations or operational circumstances). This is in contrast to traditional human error analysis practice, which considers primarily the human-centered causes, with only a cursory acknowledgment of plant influences through such simplistic measures as the time available for action. The human-centered factors and the influence of plant conditions are not independent of one another. Rather, in many major accidents, a set of particularly unusual or abnormal system conditions create the need for operator actions, and these unusual system conditions lead to unsafe acts on the part of operators. Simply stated, unsafe acts are more likely to result from unusual contexts than from a "random" human error. Analyses of nuclear power plant accidents and near misses, support this perspective, indicating that the influence of abnormal contexts appears to dominate over random human errors.[8]

This state of affairs is entirely analogous to software. Software does not fail "randomly." Instead, it fails as a result of encountering some context (i.e., a particular set of inputs, in combination with a particular operating environment and application) for which it was not properly designed. The error mechanism involved is not one in which the software does something inexplicably "wrong." On the contrary, it does exactly what it was designed to do. It executes precisely the algorithm that was programmed for that situation, unceasingly and unerringly. The problem is the context itself, one which was unexpected or untested by the system developer, and as a result, is one for which the algorithm implemented in the software (which is presumably "correct" in other situations) turns out to be inappropriate. The software is "defeated" or "fooled" by the unexpected context. In fact, the term "error-forcing context" is even more appropriate for software than it is for humans. Because software is deterministic, encountering an error-forcing context is *guaranteed* to result in a failure. Human behavior, on the other hand, is not quite so limited, in which case it would perhaps be more appropriate to speak in terms of an error-*prompting* context.

Another very simple example of an error-forcing context for software, in addition to the air traffic control example cited above, is an incident in which an aircraft was damaged when, in response to a test pilot's command, the computer raised the landing gear while the plane was standing on the runway.[7] In the right context, (i.e., when the plane is in the air), this would have been the correct action for the software to perform. However, it is *not* the appropriate action when the plane is on the ground. The developers failed to disable the function when the plane is on the ground, and the result is the existence of an error-forcing context.

The above example is, of course, so simple that it appears obvious. However, in general, an error-forcing context can be more exotic, requiring the combination of a number of unusual or unexpected conditions. An incident occurred at the Canadian Bruce-4 nuclear reactor in January 1990 in which a small loss of coolant accident resulted from a programming error in the software used to control the reactor refueling machine. Because of this error, the control computer, when suspending execution of the main refueling machine positioning control subroutine in order to execute a fault-handling subroutine triggered by a minor fault condition detected elsewhere in the plant, marked the wrong return address in its memory. As a result, execution resumed at the wrong segment of the main subroutine. The refueling machine, which at the time was locked onto one of the reactor's pressure tube fuel channels, released its brake and dropped its refueling assembly by about three feet, damaging both the refueling assembly and the fuel channel.[11]

This failure did not occur simply by virtue of the fact that the wrong address was placed on the stack. The failure also required the additional condition of the refueling machine being locked onto a channel at the time. If, instead, the refueling machine had been idle when the fault-handling interrupt was received (and assuming that the same return address was specified erroneously), no failure would have been observed. This is a case where the execution of a segment of code which violates its specification may or may not result in a failure, depending on the state of the system at the time (the *error* always exists, but the failure requires the occurrence of an error-forcing condition in the system).

To correctly assess the potential impact of such failures, it is necessary to identify both the unusual or unexpected conditions in which failure is more likely (i.e., those conditions outside the range considered during design and testing of the software), as well as the deficiencies in the software's design and implementation that affect their applicability to these "off-nominal" conditions. In other words, we need to identify the "error-forcing context," or the confluence of unexpected system conditions and latent software faults that result in failure. This result by itself would be very useful to designers. If one wishes to go further and provide a system risk assessment that is consistent with operational experience, the task of quantification must be based upon the likelihood of such error-forcing contexts, rather than upon a prediction of "random" software failure. Quantification of failure probabilities based upon error-forcing contexts for software represents a fundamental shift away from software reliability modeling.

In quantifying the system failure probability, we must concern ourselves with evaluating the likelihood of a well-defined event. The question is what is a well-defined event involving software "failure?" Simply, it is the occurrence of a hazardous condition in the system in which the software is embedded. Also, because the software action itself is deterministic, we will find that the probability of a given system failure event must be equal to the probability of the corresponding error-forcing context. If we are able to identify the system's error-forcing contexts and express them as well-defined events (both of which are discussed in the next section), then we can quantify the probability of system failure.

Note that, in this formulation, the risk quantification problem has been transformed to a more complicated, but more rational, form. We are no longer depending on the value of a single parameter (the probability of software failure). Instead, we are looking for the probabilities of finding certain system parameters (both in the input to the software and in the operating environment) in states that will lead to system failure through inappropriate software action. For example, in the case of the Bruce reactor incident, we would be concerned with evaluating the probability of finding the refueling machine locked onto a channel *while* the computer is responding to a fault elsewhere in the plant.

In general, this state information may also involve time. For instance, one of the space shuttle simulations ran into trouble during a simulated abort procedure.(7) The crew initiated an abort sequence, and then was advised by "ground control" that the abort was no longer necessary, so they "aborted" the abort. After completing another simulated orbit, they decided to go through with the abort procedure after all, and the flight computer, which did not anticipate the possibility of two abort commands in the same flight, got caught in a two-instruction loop. In this case, the error-forcing context is actually a particular *sequence* of events occurring in time, or a trajectory.

4. METHODOLOGY FOR IMPLEMENTING THE CONTEXT-BASED APPROACH

To identify the error-forcing contexts, a methodology is needed which must be able to do the following:

(1) Represent all of those states of the system which are deemed to be hazardous (the states that result from a system failure event);

(2) Model the functional and dynamic behavior of the software in terms of transitions between states of the system;

(3) Given a system failure event, identify the system states that precede it (the error forcing contexts).

There are a number of methods that might be used to perform these tasks, most notably fault tree analysis, event tree analysis, and hazard and operability analysis (HAZOP). We note that some of these techniques have been applied to software.(12,13) The approach we will use here is a combination of fault tree and event tree analysis with the Dynamic Flowgraph Methodology(14,15) (DFM), which is essentially a more sophisticated version of HAZOP, and allows the integrated analysis of both hardware and software.

To identify the error-forcing contexts associated with a system, the relevant hazardous system states (failures) are specified by an event tree. The system failure states in the event tree are typically identified using fault trees which "hang" from the event tree branches. Event trees are commonly used in the nuclear reactor safety community for accident progression modeling. Their role is to provide boundary conditions for the fault tree analyses.

For systems that involve software, the fault trees can be developed and evaluated using DFM, which is a digraph-based method for modeling and analyzing the behavior and interaction of software and hardware within an embedded system. A DFM model represents both the logical and temporal characteristics of a system (the software, the hardware, and their interactions with each other and the environment) and is used to build fault trees that identify critical events and sequences. DFM provides an analytical framework for systematically identifying the principal failure modes of an embedded system, whether they are due to unanticipated inputs, hardware failures, adverse environmental conditions, or implementation errors. Software is represented in the DFM model by transition boxes, which represent functional relationships between system parameters (both software and hardware), and which are associated with a time lag. "Firing" of the transition boxes provides the means for modeling the dynamic behavior of the system as it advances from one state to the next as a result of software action.

A DFM analysis is almost identical to a HAZOP analysis except for two important differences. DFM is an automated technique, rather than a manual process, and its deductive analysis procedure generates fault trees and prime implicants[16,17] which identify the basic events which can lead to a specified top event (hazard state). A prime implicant is the multiple-valued logic equivalent of a minimal cut set, which is a minimal set of basic events of a binary fault tree that are sufficient to cause the top event. A prime implicant is any conjunction of primary events that is sufficient to cause the top event, but does not contain any shorter conjunction of the same events that is sufficient to cause the top event. The prime implicants of any fault tree are unique and finite. Also, because of the dynamic nature of the DFM model, the prime implicants of the resulting fault trees are time-dependent, specifying both the *state* of the system required to produce the top event, as well as the *time* at which it must occur.

The prime implicants of the DFM fault trees specify the conditions that are capable of producing the failure event. As will be illustrated in the example in the following section, the prime implicants consist only of *system* states (i.e., the values of software inputs, hardware configurations, and process variable values), there are no events referring to the "success" or "failure" of the software. Taken as a whole, the fault tree prime implicants and the event tree branches from which they "hang" specify the error-forcing context (encompassing both the operating environment and the software input) in which the system is vulnerable to software errors.

5. EXAMPLE

5.1. Main Feedwater System

Consider the event tree shown in Fig. 1.[18] This is the event tree corresponding to the initiating event "very small loss of coolant accident (LOCA)" for the Surry Nuclear Station, Unit 1. This is from one of the plant analyses conducted as part of the NUREG-1150 effort by the Nuclear Regulatory Commission. Across the top of the figure, the initiating event (very small LOCA) and the plant systems and actions that might affect the subsequent course of events are listed in order of the time sequence in which they are expected to influence events in the reactor. The tree illustrates the different possible sequences of events, depending on the success or failure of the systems at the top, and indicates their consequences in terms of the reactor core integrity (the core status column at the right of the figure). Each branch point in the tree distinguishes between the success or failure of the system above it. Taking the upper branch corresponds to success of the system, while dropping to the lower branch indicates failure.

Rather than looking in detail at every system included in the tree, let us just consider one. MFW corresponds to the main feedwater system. From the location of the "MFW" branch on the tree, we can see that, at the time this system is called upon in this particular accident scenario, one of the systems before it has failed (AFW, the auxiliary feedwater system), but the other three have been successful. These events serve to characterize the plant conditions at the time of loss of main feedwater, and comprise the *context* in which a fault tree analysis of the MFW event would be conducted for this particular accident scenario. If the main feedwater system in question involves software, we can then evaluate the impact that any software failure modes will have on this particular accident scenario by determining the likelihood of this set of events and the likelihood that the system parameters will be found in the ranges specified by the fault tree prime implicants.

To illustrate, consider the portion of a Main Feedwater System (MFWS) analyzed using DFM in Guarro *et al.*[16] and Yau *et al.*[17] The MFWS is designed to deliver water to the steam generators (SG) during power operations and after reactor trip. The main feed valve is controlled by the SG level control system. The function of the SG level control system is to maintain the water level at a pre-defined set point (68% narrow range level under normal operating conditions). The system consists of sensors that measure steam generator level, steam flow and feed flow, digital-to-analog and analog-to-digital converters, digital control software that executes on a clock cycle of 0.1s, and actuators that regulate the position of the main feed valve. The system is implemented as a three-element control system, where measurements of the steam generator level, the steam flow and the feed flow are taken every tenth of a second as sensor inputs to the software. The software then uses these inputs to generate a target position for the main feed valve. This command is the output to the valve actuators. A schematic of the SG control system is shown in Fig. 2.

Three sets of control logic are implemented by the steam generator level control system; they are Proportional Integral and Derivative (PID) logic, High Level Override (HLO) and Reactor Trip Override (RTO). Reactor Trip Override logic is used when

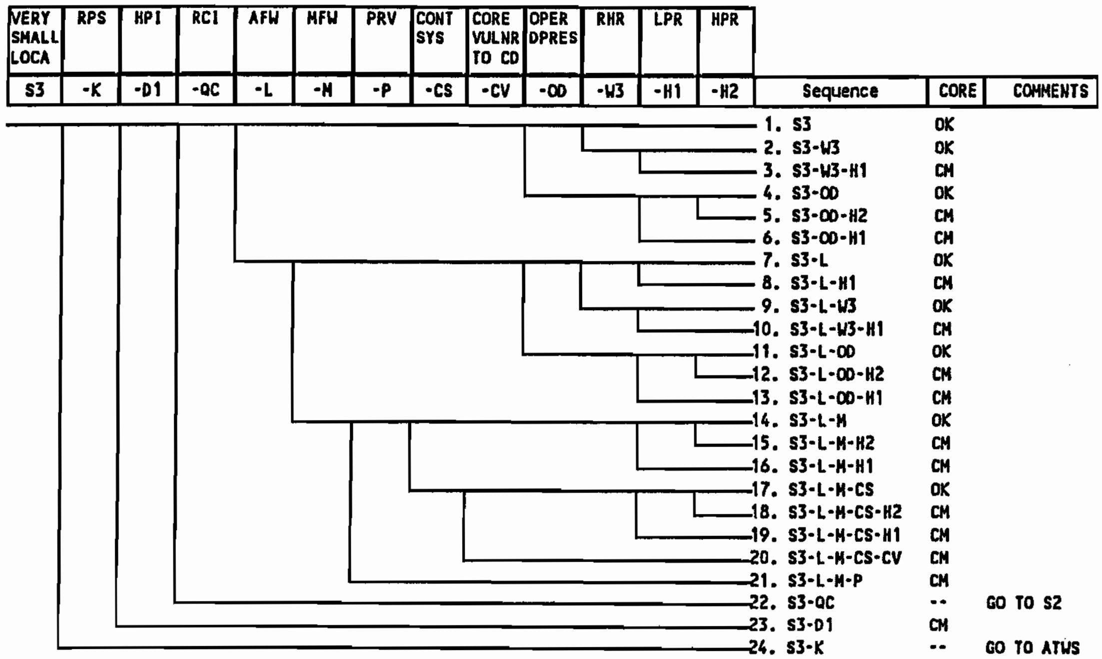

Fig. 1. Event tree for very small LOCA.[18]

the digital control software receives a reactor trip signal, in which case the target main feed valve position is then set to 5%. High Level Override logic is employed when the steam generator reading is greater than 89%, in which case the target main feed valve position is set to fully closed. The HLO control action is irreversible; this means that once an HLO signal is triggered, the system will not return to the normal PID control action unless the system is reinitialized. Proportional Integral and Derivative logic is implemented in all cases not covered by the other two sets of control logic.

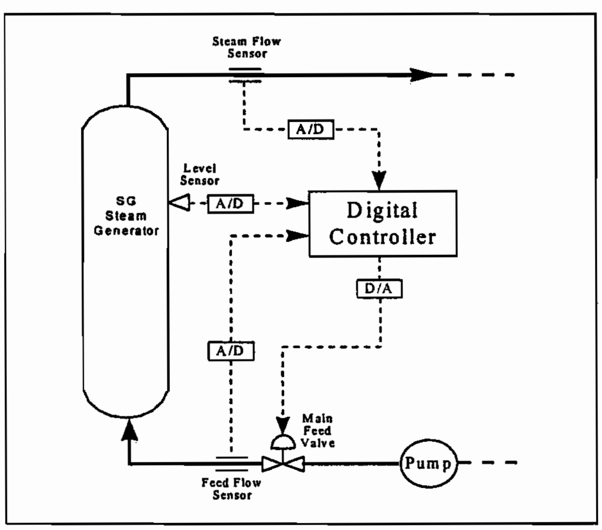

Fig. 2. Schematic of the steam generator level control system.[16,17]

5.2. Prime Implicants

The system was analyzed twice, with two different faults intentionally being injected into the system. For the first case, an error was introduced into the design specification of the control software. Instead of subtracting the derivative-lag signal of the steam flow-feed flow mismatch from the steam generator level, the faulted specification called for the addition of these two terms. The DFM model was constructed without assuming any prior knowledge of the software specification error, and the top event specified for analysis was defined as the steam generator "overflowing." The analysis was carried out for one step backward in the reference time frame, and ten prime implicants were identified. A typical prime implicant is the one shown in Table I.

The prime implicant reveals that the steam generator level sensor stuck at the low reading, combined with the level being very high, will cause the steam generator to overflow. The low reading provided by the level sensor will cause the control software to act

Table I. Prime Implicant 1.1

*Main feed valve good	@ $t = 0$ AND
*Main feed pump good	@ $t = 0$ AND
High level override inactive	@ $t = -1$ AND
Reactor trip override inactive	@ $t = -1$ AND
Main feed valve between 60–80%	@ $t = -1$ AND
Steam flow between 30–60%	@ $t = -1$ AND
SG level at level 8	@ $t = -1$ AND
*Steam flow sensor good	@ $t = -1$ AND
Level sensor stuck low	@ $t = -1$

as if there is not enough water in the steam generator and command the main feed valve to open, causing the SG to overflow. The asterisks in the table indicate necessary *non-failure* conditions in the prime implicant that result from the multiple-valued logic representation of the system model. For instance, the main feed pump being normal is part of the necessary condition in the prime implicant since a failed pump cannot sustain the feed flow into the SG that is necessary to cause overflow.

If a prime implicant does not contain basic component failure modes that can cause the top event directly, this usually means that a software error is identified. The event sequence leading from the prime implicant to the top event needs to be analyzed to locate the software error. The prime implicant shown in Table II, unlike that in Table I, does not contain any basic component failure modes, but consists of nonfailure hardware component conditions and software input conditions. This prime implicant points to the possibility of a software fault, but it is not directly obvious where the fault is and how the overflow condition is brought about. After reconstructing the sequence of events from the prime implicant to the top event, it can be determined that this prime implicant does indeed correspond to the inappropriate addition of the derivative-lag to the SG level.

Table II. Prime Implicant 1.2

*Main feed valve good	@ $t = 0$ AND
*Main feed pump good	@ $t = 0$ AND
High level override inactive	@ $t = -1$ AND
Reactor trip override inactive	@ $t = -1$ AND
Main feed valve between 60–80%	@ $t = -1$ AND
Feed flow between 60–80%	@ $t = -1$ AND
Steam flow between 30–60%	@ $t = -1$ AND
SG level at level 8	@ $t = -1$ AND
*Feed flow sensor good	@ $t = -1$ AND
*Steam flow sensor good	@ $t = -1$ AND
*Level sensor good	@ $t = -1$

Note that none of the basic events in prime implicant 1.2 refer explicitly to a software "failure." Instead, the basic events refer only to the values of software inputs and the states of hardware components. These basic events specify an error-forcing *context*, the conditions which must occur (and the times at which they must occur) in order for the preexisting software fault to be "activated."

For the second faulted-case analysis, it was assumed that an error had been introduced into the control software code. The assumption was that, instead of triggering the High Level Override (HLO) signal at 89% level, this programming error causes the HLO signal to be activated at 69% level. As the level set point is at 68%, a slight increase in SG level from the set point will cause the software to command the closing of the main feed valve to 5%.

A fault tree was developed for the top event "steam generator level dropped to 0% narrow range," using the DFM model of the faulted system. One prime implicant was identified, which is given in Table III.

The prime implicant does not contain any basic component failure events, but it encompasses input conditions that can trigger the error in the software. It is important to point out that, in general, the identification of a prime implicant does not imply that the occurrence of the prime implicant will *necessarily* lead to the top event. It simply points out the nexus of system conditions that must be present in order for the top event to occur. In other words, it is a necessary, but not sufficient, condition for the top event. This is a result of the fact that conditions in the prime implicant are expressed as ranges of continuous variables. The actual error-forcing condition may exist only within a subset of the range given by the prime implicant, whereas all other points within the range may not lead to the top event. For example, in prime implicant 2.1, the error is really only triggered if the steam generator level is above 69% (below 69%, the HLO override signal is not activated). However, because of the discretization scheme chosen during construction of the model, all steam generator levels between 65% and 71% are represented as the same state. Thus, even within the conditions

Table III. Prime Implicant 2.1

High level override inactive	@ $t = -5$ AND
Steam flow between 80–100%	@ $t = -5$ AND
SG level between 65–71%	@ $t = -5$ AND
SG pressure between 960–1185 psi	@ $t = -5$

specified by the prime implicants, there is still some uncertainty about where the actual error-forcing condition lies, if it in fact exists. This uncertainty can be reduced by either performing another analysis, with a finer discretization structure employed within the states specified by the prime implicants, or by testing the software in the neighborhood of the conditions specified by the prime implicant.

6. CONCLUDING REMARKS

We have described an approach to software risk assessment which explicitly recognizes that software behavior is deterministic. The source of the apparent "randomness" of software "failure" behavior is, instead, a result of both the input to the software as well as the application and environment in which it is operating (i.e., the *context*). In some contexts, the software is correct, but in others (i.e., the "error-forcing contexts"), it is not. One way of identifying these error-forcing contexts is by finding the prime implicants of system DFM models, subject to the boundary conditions specified by the associated event tree.

Having found prime implicants, one is faced with the question of what to *do* about them. Generally, when a fault is discovered in a piece of software, the usual remedy is to correct it. However, the cost of correcting software can be very large due to the fact that fixes may also introduce new errors and the verification and validation process must be started over again. If it should turn out that the prime implicant is sufficiently unlikely, then one might come to the conclusion that the software fault can be tolerated, or that the cost of fixing it is not justified by the decrease in risk that would result.

In order to support this kind of cost-benefit analysis for software, it is necessary to know both the probabilities of the prime implicants and the consequences of each fault tree top event. Performing the fault tree analysis as part of an accident sequence analysis, where the fault tree hangs from the event tree branch that represents the failure of the corresponding system, the event tree specifies the scenario and balance of plant conditions under which the top event occurs. This information can then be used to generate probability distributions for the conditions specified by the fault tree prime implicants. Note that the prime implicants do not contain events that say "software failure," rather, they identify states of physical system parameters and sequences of events for which the software has been incorrectly designed. By estimating their likelihood, we are estimating the probability of failure due to the occurrence of an "error-forcing" context. Also, note that the prime implicants refer to more than just the states of the input to the software, they also refer to the states of parameters in the *physical* system. The fault tree prime implicants specify all of the conditions (the error-forcing context) under which the system is vulnerable to a software error (as well as hardware failures).

The event tree also allows one to establish an upper bound on the allowable probability of failure for each branch in the tree. The further to the right on the tree that the event in question appears (meaning that it must occur in combination with a number of other failures in order to lead to system failure), the higher, in geneal, that upper bound will be, meaning that for some applications, the "ultra-high" reliability requirements commonly believed to be necessary for safety-critical software may not be necessary after all. For example, consider the event H2 at the right of the event tree in Fig. 1, which corresponds to failure of the charging pump system in high pressure recirculation mode (and, further, assume that there may be some software that is responsible for operating this system). Failure of this system during a very small LOCA, combined with failure of the operator to depressurize the reactor coolant system (accident sequence 5) leads to a core melt, while success of the system is "OK." Clearly, this is a safety-critical system. However, failure of this system in isolation will not lead to damage of the core. A number of other systems must fail *in addition to* the charging pump system. If the coincident failure of those other systems is sufficiently unlikely, then a reasonably large probability of failure of the charging pump system can probably be tolerated.

Sequence 5 contains the smallest number of failures involving H2 that will lead to core damage, so let us take a closer look at it. According to Ref. 18, the frequency of the initiating event S3 is 1.2×10^{-2}/yr, and the probability of failure of the operator to depressurize is less than 7.6×10^{-2}. Thus, if the maximum acceptable frequency of occurrence for this sequence is even as low as 10^{-6}/yr, that means that the maximum acceptable probability of H2 is only low as 10^{-3}, and furthermore, only a fraction thereof would be attributable to the controlling software, meaning that a decision to leave the error instead of "fixing" it may be justified. Also, for errors in this region, it may be practical to demonstrate an acceptable probability of occurrence by means of testing the fact that there are well-defined boundary conditions on the operational profile, and that the target failure probability is not infeasibly small, may lead to a manageable set of test cases.

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DRAFT REGULATORY GUIDE DG-1130

(Proposed Revision 2 of Regulatory Guide 1.152)

CRITERIA FOR USE OF COMPUTERS IN SAFETY SYSTEMS OF NUCLEAR POWER PLANTS

A. INTRODUCTION

Criterion 21, "Protection System Reliability and Testability," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires, among other things, that protection systems (or safety systems) be designed for high functional reliability commensurate with the safety functions to be performed. Criterion III, "Design Control," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 requires, among other things, that quality standards be specified and that design control measures be provided for verifying or checking the adequacy of design.

This regulatory guide describes a method acceptable to the NRC staff for complying with the NRC's regulations for promoting high functional reliability and design quality for the use of computers in safety systems of nuclear plants. The term "computer" means a system that includes computer hardware, software, firmware, and interfaces.

Regulatory guides are issued to describe to the public methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, to explain techniques used by the staff in evaluating specific problems or postulated accidents, and to provide guidance to applicants.

Regulatory guides are not substitutes for regulations, and compliance with regulatory guides is not required. Regulatory guides are issued in draft form for public comment to involve the public in developing the regulatory positions. Draft regulatory guides have not received complete staff review; they therefore do not represent official NRC staff positions.

The information collections contained in this draft regulatory guide are covered by the requirements of 10 CFR Part 50, which were approved by the Office of Management and Budget (OMB), approval number 3150-0011. The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

B. DISCUSSION

IEEE Std 7-4.3.2-2003, "Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations,"* was developed by the Nuclear Power Engineering Committee of the Institute of Electrical and Electronics Engineers (IEEE). The NRC staff has worked with IEEE in developing IEEE Std 7-4.3.2-2003 to ensure that the guidance provided by the consensus standard is consistent with the NRC's regulations. This standard evolved from IEEE Std 7-4.3.2-1993 and reflects advances in digital technology. It also represents a continued effort by IEEE to support the specification, design, and implementation of computers in safety systems of nuclear power plants. IEEE Std 7-4.3.2-2003 specifies additional computer-specific requirements to supplement the criteria and requirements of IEEE Std 603-1998, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations."

Instrumentation and control (I&C) systems designs that use computers in safety systems make extensive use of advanced technology (i.e., equipment and design practices). These designs are expected to be significantly and functionally different from current designs, and may include the use of microprocessors, digital systems and displays, fiber optics, multiplexing, and different isolation techniques to achieve sufficient independence and redundancy.

With the introduction of digital systems into plant safety system designs, concerns have emerged regarding the possibility that a design error in the software in redundant channels of a safety system could lead to common-cause or common-mode failure of the safety system function. Conditions may exist under which some form of diversity may be necessary to provide additional assurance beyond that provided by the design and quality assurance (QA) programs that incorporate software QA and verification and validation (V&V). The design techniques of functional diversity, design diversity, diversity in operation, and diversity within the four echelons of defense in depth (provided by the reactor protection, engineered safety features actuation, control, and monitoring I&C systems) can be applied as defense against common-cause failures. Manual operator actuations of safety and non-safety systems are acceptable provided the necessary diverse controls and indications are available to perform the required function under the associated event conditions and within the acceptable time.

The justification for equipment diversity, or for the diversity of related system software such as a realtime operating system, must extend to equipment components to ensure that actual diversity exists. For example, different manufacturers might use the same processor or license the same operating system, thereby incorporating common failure modes. Claims for diversity based only on different manufacturers are insufficient without consideration of the above.

With respect to software diversity, experience indicates that independence of failure modes may not be achieved in cases where multiple versions of software are developed from the same software

* IEEE publications may be purchased from the IEEE Service Center, 445 Hoes Lane, Piscataway, NJ 08854.

requirements. Other considerations, such as functional and signal diversity, that lead to different software requirements form a stronger basis for diversity.

Some safety system designs may use computers that were not specifically designed for nuclear power plant applications. Section 5.4.2 of IEEE Std 7-4.3.2-2003 provides general guidance for commercial grade dedication. Annex C of this standard provides useful information on providing confidence that an existing commercial computer is of sufficiently high quality and reliability to be used in a safety system.

IEEE Std 7-4.3.2-2003 does not provide guidance regarding security measures for computer-based system equipment and software systems. Regulatory Positions 2.1 through 2.9 were added to provide guidance specific to computer-based (cyber) safety system security.

Section 5.9 of IEEE Std 7-4.3.2-2003, Control of Access, refers to the applicable requirements in IEEE Std 603-1998 and states: "The design shall permit the administrative control of access to safety system equipment. These administrative controls shall be supported by provisions within the safety systems, by provision in the generating station design, or by a combination thereof." For digital computer-based systems, controls of both physical and electronic access to system software and data should be provided to prevent changes by unauthorized personnel. Controls should address access via network connections and via maintenance equipment. Additionally, the design of the plant data communication systems should ensure that the systems do not present an electronic path by which unauthorized personnel can change plant software or display erroneous plant status information to the operators. Useful information for establishing communication independence of plant equipment and systems is provided in Annex E of IEEE Std 7-4.3.2-2003.

Computer-based systems must be secure from electronic vulnerabilities as well as from physical vulnerabilities, which have been well addressed. Security of computer-based system software relates to the ability to prevent unauthorized, undesirable, and unsafe intrusions throughout the life cycle of the safety system. Computer-based systems are secure from electronic vulnerabilities if unauthorized access and use of those systems is prevented. The security of computer-based systems is established through (1) designing the security features that will meet user security requirements in the systems, (2) developing the systems without undocumented codes (e.g., back door coding, viruses, worms, Trojan horses, and bomb codes), and (3) installing and maintaining those systems in accordance with the users' security program.

Regulatory Positions 2.1 through 2.9 provide guidance specific to safety system security. The effectiveness of the security functions implemented in the software safety system should be confirmed during verification and validation (V&V) and in the configuration management process of the safety system software in each lifecycle phase.

IEEE Std. 7-4.3.2-2003 includes seven informative annexes, which are not part of IEEE Std 7-4.3.2-2003. These annexes are, therefore, not endorsed by the NRC staff. However, the staff believes that these annexes contain useful information.

C. REGULATORY POSITION

1. FUNCTIONAL AND DESIGN REQUIREMENTS

Conformance with the requirements of IEEE Std 7-4.3.2-2003, "Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," is a method acceptable to the NRC staff for satisfying the NRC's regulations with respect to high functional reliability and design requirements for computers used in safety systems of nuclear power plants.

2. SECURITY

This regulatory position uses the waterfall lifecycle phases as a framework for describing specific digital safety system security guidance. Lifecycles other than the waterfall lifecycle may be used. The digital safety system development process should address potential security vulnerabilities in each phase of the digital safety system lifecycle.

The typical waterfall lifecycle consists of the following phases:

- Concepts
- Requirements
- Design
- Implementation
- Test
- Installation and Checkout
- Operation
- Maintenance
- Retirement

The lifecycle phase-specific security requirements should be commensurate with the risk and magnitude of the harm resulting from unauthorized access, use, disclosure, disruption, or destruction of the digital safety system.

The user should establish a security quality assurance program and a security configuration management program as part of its security program. The security quality assurance program and security configuration management program can be incorporated into the existing quality assurance and configuration management programs.

The Quality Assurance organization should conduct periodic audits to determine the effectiveness of the digital safety system security program.

Regulatory positions 2.1 through 2.9 describe waterfall lifecycle phase-specific digital safety system security activities and recommendations.

2.1 Concepts phase

In the Concepts phase, the user and developer should delineate safety system security features that should be implemented to meet the desired system security capabilities. During this activity, the system architecture is selected and the desired safety system security functional capabilities are allocated to hardware, software, and user interface components.

The user and developer should perform security risk analyses to identify potential security vulnerabilities in the relevant phases of the system and software life cycle. The results of the analysis should be used to establish security requirements for the system (hardware and software).

Remote access to the safety system software functions or data from outside the technical environment of the plant (e.g., from the administrative or engineering buildings or from outside the plant) that involves a potential security threat to safety functions should not be implemented.

2.2 Requirements phase

2.2.1 System features

The users and developers should define the security functional and performance requirements; interfaces external to the system; and the requirements for qualification, human factors engineering, data definitions, user documentation for the software and hardware, installation and acceptance, user operation and execution, and user maintenance.

The security requirements are part of the overall system requirements. Therefore, the V&V process of the overall system should ensure the correctness, completeness, accuracy, testability, and consistency of the system software and hardware system requirements, which include security requirements.

Requirements specifying the use of pre-developed software (e.g., reuse software and commercial off-the-shelf software) should minimize the vulnerability of the safety system (e.g., by minimizing the number of pre-developed software functions used by the safety system to the extent necessary or using existing security functions of the pre-developed software).

2.2.2 Development activities

The developer should delineate its security policies to ensure the developed products (hardware and software) do not contain undocumented code (e.g., back door coding), malicious code (e.g., intrusions, viruses, worms, Trojan horses, or bomb codes), and other unwanted and undocumented functions or applications.

2.3 Design phase

2.3.1 System features

The safety system software security requirements identified in the system requirements specification should be translated into specific design configuration items in the software design description. The safety system software security design configuration items should address control over (1) access to the software functions, (2) use of safety system services, (3) data communication with other systems, and (4) the list of personnel who may access and use the system.

Design configuration items incorporating pre-developed software into the safety system should be specified such that vulnerability of the safety system security is minimized.

Access control should be based on the results of risk analyses. The results of the analyses may require more complex access control, such as a combination of knowledge (e.g., password), property (e.g., key, smart-card) and personal features (e.g., fingerprint), rather than just a password.

2.3.2 Development activities

The developer should delineate the standards and procedures that will conform with the applicable security policies to ensure the system design products (hardware and software) do not contain undocumented code (e.g., back door coding), malicious code (e.g., intrusions, viruses, worms, Trojan horses, or bomb codes), and other unwanted or undocumented functions or applications.

2.4 Implementation phase

In the software implementation phase, the system design is transformed into code, database structures, and related machine executable representations. The implementation activity addresses software coding and testing, including the incorporation of reused software products.

2.4.1 System features

The developer should ensure that the security design configuration item transformations from the system design specification are correct, accurate, and complete.

2.4.2 Development activities

The developer should implement security procedures and standards to ensure against tampering with the developed software. The developer's standards and procedures should include testing, including scanning, to ensure against undocumented codes or malicious codes that might (1) allow unauthorized access or use of the system or (2) cause systems to behave beyond the system requirements. There should be provisions against the incorporation of hidden functions in the application development software or the system software that could support potential unauthorized access. If provisions cannot be implemented for pre-developed software, the use of such software should be justified considering potential security threats.

The user and developer should review the possibility for deliberate modification of software to cause erroneous behavior of the software triggered by certain time or data constraints (e.g., viruses, worms, and Trojan horses).

2.5 Test phase

The objective of testing software security functions is to ensure that the software security requirements and system security requirements allocated to software are validated by execution of integration, system, and acceptance tests. Testing includes software testing, software integration testing, software qualification testing, system integration testing, and system qualification testing.

2.5.1 System features

The security requirements and configuration items are part of the overall system requirements and design configuration items. Therefore, testing security design configuration items is just one element of the overall system testing. The user and developer should test each system security feature to verify that the implemented system does not increase the risk of security vulnerabilities.

2.5.2 Development activities

The developer should perform testing and scanning to ensure the developed products (i.e., hardware and software) do not contain undocumented code (e.g., back door coding), malicious code (e.g., intrusions, viruses, worms, Trojan horses, or bomb codes), and other unwanted and undocumented functions or applications. Additionally, the developer should audit the configuration management processes to ensure that the software is developed in accordance with the appropriate configuration management procedures and standards.

2.6 Installation and Checkout phase

In installation and checkout, the safety system is installed and tested in the target environment. The system user should perform an acceptance review and test the safety system security features. The objective of installation and checkout security testing is to verify and validate the correctness of the safety system security features in the target environment.

2.6.1 System features

The user should ensure the the system features enable the user to perform post-installation testing of the system to the verify and validate that the security requirements have been incorporated into the system appropriately.

2.6.2 Development activities

A user or licensee should have a comprehensive digital system security program. The security policies, standards, and procedures should ensure that installation of the digital system will not compromise the security of the digital system, other systems, or the plant. This may require the user to perform a security assessment, which includes a risk assessment, to identify the potential security vulnerabilities caused by installation the digital system. The risk assessment should include an evaluation of new security constraints in the system; an assessment of the proposed system changes and their impact on system security; and an evaluation of operating procedures for correctness and usability. The results of this assessment should provide a technical basis for establishing certain security levels for the systems and the plant.

2.7 Operation phase

The operation lifecycle process involves the use of the safety system by the end user in its intended operational environment.

The user should monitor and record access and use of the system to ensure that its digital system security policies are implemented properly. The monitoring should include real-time monitoring and periodic audits. The type of monitoring is determined by the risk analyses performed in earlier lifecycle phases. The audit should include the security of any equipment that is connected to the system for maintenance.

The user should evaluate the impact of safety system changes in the operating environment on safety system security; assess the effect on safety system security of any proposed changes; evaluate operating procedures for compliance with the intended use; and analyze security risks affecting the user and the system. The user should evaluate new security constraints in the system; assess proposed system changes and their impact on system security; and evaluate operating procedures for correctness and usability.

2.8 Maintenance phase

The maintenance phase is activated when the user changes the system or associated documentation. The types of changes may be categorized as:

- Modifications (i.e., corrective, adaptive, or perfective changes)
- Migration (i.e., the movement of software to a new operational environment)
- Retirement (i.e., the withdrawal of active support by the operation and maintenance organization, partial or total replacement by a new system, or installation of an upgraded system).

System modifications may be derived from requirements specified to correct errors (corrective), to adapt to a changed operating environment (adaptive), or to respond to additional user requests or enhancements (perfective).

2.8.1 Maintenance activities

Modifications of the safety system should be treated as development processes and should be verified and validated as described above. Security functions should be assessed as described in the above regulatory positions and should be revised as appropriate to reflect requirements derived from the maintenance process.

When migrating software, the user should verify that the migrated software meets the safety system security requirements. The maintenance process should continue to conform to existing safety system security requirements unless those requirements are to be changed as part of the maintenance activity.

2.8.2 Quality Assurance

If the safety system security functions were not verified and validated previously using a level of effort commensurate with the safety system security functional requirements, and appropriate documentation is not available or adequate, the user should determine whether the missing or incomplete documentation should be generated. In making this determination of whether to generate missing documentation, the minimum safety system security functional requirements should be taken into consideration.

The user should establish a security configuration management program as part of its security program. The security configuration program may be incorporated into the existing configuration management program.

2.8.3 Incident Response

The user should develop an incident response and recovery plan for responding to digital system security incidents(e.g., intrusions, viruses, worms, Trojan horses, or bomb codes). The plan should be developed to address various loss scenarios and undesirable operations of plant digital systems, including possible interruptions in service due to the loss of system resources, data, facility, staff, and/or infrastructure. The plan should define contingencies for ensuring minimal disruption to critical services in these instances.

2.8.4 Audits and assessments

The user should perform periodic computer system security self-assessments and audits, which are key components of a good security program. The user should assess proposed safety system changes and their impact on safety system security; evaluate anomalies that are discovered during operation; assess migration requirements; and re-perform V&V tasks to ensure that vulnerabilities have not been introduced into the plant environment.

2.9 Retirement phase

In the Retirement lifecycle phase, the user should assess the effect of replacing or removing the existing safety system security functions from the operating environment. The user should include in the scope of this assessment the effect on safety and non-safety system interfaces of removing the system security functions. The user should document the methods by which a change in the safety system security functions will be mitigated (e.g., replacement of the security functions, isolation from other safety systems and user interactions, or retirement of the safety system interfacing functions).

3. REFERENCED STANDARDS

Section 2 of IEEE Std 7-4.3.2-2003 references several industry codes and standards. If a referenced standard has been separately incorporated into the NRC's regulations, licensees and applicants must comply with the standard as set forth in the regulations. If the referenced standard has been endorsed by the NRC staff in a regulatory guide, the standard constitutes an acceptable method of meeting a regulatory requirement as described in the regulatory guide. If a referenced standard has been neither incorporated into the NRC's regulations nor endorsed in a regulatory guide, licensees and applicants may consider and use the information in the referenced standard, if appropriately justified, consistent with regulatory practice.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this guide. No backfitting is intended or approved in connection with the issuance of this guide.

This proposed revision has been released to encourage public participation in its development. Except when an applicant or licensee proposes or has previously established an acceptable alternative method for complying with specified portions of the NRC's regulations, the methods described in this guide will be used in the evaluation of (1) submittals in connection with applications for construction permits, design certifications, operating licenses, and combined licenses for use of computers in safety systems, and (2) submittals from operating reactor licensees who voluntarily propose to initiate safety system modifications if there is a clear nexus between the proposed modifications and this guidance.

REGULATORY ANALYSIS

BACKGROUND

With the introduction of computers in safety systems, concerns have arisen over the possibility that the use of computer software could result in a common mode failure. Because of these concerns, the NRC staff has placed significant emphasis on defense in depth against propagation of common mode failures within and between functions. The two principal factors for defense against common mode failures are quality and diversity. Each postulated common mode failure should be analyzed using best-estimate methods to address vulnerabilities to common mode failures. Design qualification and quality assurance programs are intended to provide protection against design deficiencies and manufacturing errors. The guidelines in IEEE Std. 603-1998 and IEEE Std 7-4.3.2-2003 should be applied to the development of digital computer systems for purposes of developing high-quality hardware and software.

1. PROBLEM

IEEE Std 7-4.3.2-1993 was endorsed by Revision 1 of Regulatory Guide 1.152 in January 1996. The development processes for computer systems continue to evolve. The revision of this standard (IEEE Std 7-4.3.2-2003) represents a continued effort by IEEE to support the specification, design, and implementation of computers in safety systems. The regulatory guide should, therefore, be revised to reflect the current state of the technology.

2. OBJECTIVE

The objective of the regulatory action is to update NRC guidance for the use of computers in safety systems and to provide guidance on safety system security.

3. TECHNICAL APPROACH

This regulatory guide endorses the guidance of IEEE Std 7-4.3.2-2003. Issuing a regulatory guide is consistent with the NRC policy of evaluating the latest versions of national consensus standards in terms of their suitability for endorsement by regulatory guides. This guide provides a standardized approach so that the nuclear industry and the NRC staff may have a common understanding on criteria for the use of computers in safety systems.

The significant changes in IEEE Std 7-4.3.2-2003 are:

1. The “Software Quality Metrics” section was added. The industry practice is moving towards the use of software quality metrics to assure/monitor/improve software quality in addition to the verification and validation (V&V) that has traditionally been applied.

2. The “Qualification of Existing Commercial Computers” section was expanded to provide additional guidance that addresses the move toward the use of more commercial hardware and software in safety systems.

3. The “Software Tools” section was revised to address expanded use of software tools and methods.

4. The “Verification and Validation” section was revised to reference IEEE Std 1012-1998.

5. The “Software Configuration Management” section was expanded to provide additional guidance by identifying the key requirements for configuration management for safety system software using the guidance in IEEE Std 828-1998 and IEEE Std 1042-1987.

6. A “Software Project Risk Management” section was added to provide additional guidance consistent with IEEE Std 1540-2001 on risk management, and IEEE Std 12207.0-1996 on software lifecycle processes.

7. A “Fault Detection and Self-Diagnostics” section was added to discuss features that are unique to software and computer systems.

8. The “Identification” section was expanded to include software-specific requirements by extending the IEEE Std 603-1998 identification requirements to software.

9. The Annex on “Dedication of Existing Commercial Computers” was updated to more completely address COTS issues.

10. The Annex on “Identification and Resolution of Hazards” was revised to represent current practices and processes for hazards analysis.

In addition, the staff has provided guidance specific to computer-based (cyber) safety system security.

4. CONCLUSION

It is recommended that the NRC revise Regulatory Guide 1.152, since this action should enhance the licensing process. The staff has concluded that the proposed action will reduce unnecessary burden on both the NRC and its licensees, and it will result in an improved process for the use of computers in safety systems. Furthermore, the staff sees no adverse effects associated with revising Regulatory Guide 1.152. Use of this revision is optional by licensees of the currently operating nuclear power plants.

BACKFIT ANALYSIS

The regulatory guide does not require a backfit analysis as described in 10 CFR 50.109(c) because the use of this revision of Regulatory Guide 1.53 is voluntary by the licensees of currently operating nuclear power plants.

(11)

AP1000

AP1000 Design Certification Review

Westinghouse Electric Company

Presentation to

Advisory Committee on Reactor Safeguards

June 3, 2004

BNFL Westinghouse

AP1000

Safety Goal Risk Measures

- **NRC Safety Goal Policy Statement**
  - no significant additional risk to life and health
- **Quantitative Health Objectives - metrics for Safety Goal**
  - fatality and cancer risks < 0.1% of sum from other causes
- **Quantitative Health Objectives - numerics**
  - risk of prompt fatality < 5E-07 per reactor year
  - risk of latent cancer fatality < 2E-06 per reactor year
- **AP1000 PRA Results**
  - risk of prompt fatality 8.4E-11 per reactor year
  - risk of latent cancer fatality 8.6E-10 per reactor year

AP1000 Risk Quantification

- The AP1000 comparison to Safety Goal shows that additional uncertainties associated with severe accident analysis, such as those discussed today, can readily be tolerated without challenging the Safety Goal measures.

Summary of Issue 5

- Exothermic intermetallic reactions could lead to a vessel failure and produce a fuel-coolant interaction (FCI) greater than currently evaluated.
- ACRS would like to review FCI models and see justification that such a FCI does not fail containment.

AP1000

Ex-Vessel FCI in AP1000

- **FCI Analysis submitted for AP600**
  - TEXAS code used to determine FCI loads
  - Side, hinged failure is limiting case
  - Upper bound containment vessel strain is 3.8%
  - Tests on vessel material show 22 to 32% strain for ultimate load
  - FCI does not fail containment
- **AP600 analysis applied to AP1000**
  - Similar vessel lower head geometry
  - Similar lower plenum debris characteristics
    - materials
    - temperatures
  - Same vessel failure modes
  - AP1000 vessel closer to containment floor
- **This is consistent with NRC staff findings**

AP1000

FCI in AP1000

- **Lower metal layer exothermic reaction scenario would challenge vessel bottom.**
- **Vessel bottom failure not the limiting case**
  - Bottom of vessel close to floor
  - Limited pre-mixing volume
  - Limited debris participating in the FCI
- **The lower metal layer exothermic reaction failure scenario is bounded by side, hinged failure scenario – containment does not fail.**

AP1000

Summary of Issue 6

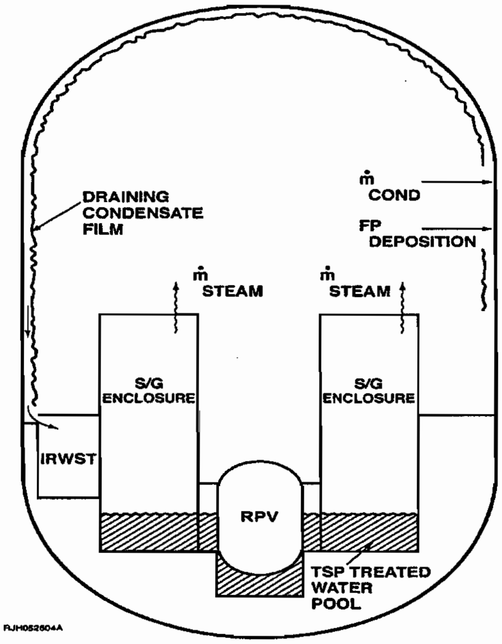
Organic Iodine Production: The acidification of containment water as a result of radiolysis of organic material could give rise to significant airborne fission product iodine in gaseous organic form. We need to review how Westinghouse and the staff have dealt with this potential.

AP1000

Issue 6: Organic Iodine Production

- Formation of organic iodine as a result of radiolysis or organic materials involves the availability of elemental iodine (I_2).
- Elemental iodine can be produced from iodide (I^-) in water pools or films where pH is not controlled to be 7 or greater.
- AP1000 containment design includes TSP to control the pH of the water pool that collects in the lower compartment and reactor cavity following an accident.
- However, no specific pH control treatment for the condensate film draining down the containment dome and shell is provided.
- Cesium iodide can be deposited on the draining film and provide a source of I^- that could potentially be converted in the film to I_2 given the film was acidified.
- Film residence time depends on the steam condensation rate and limits the amount of film acidification and CsI deposition. A range of 40 to 260 seconds has been estimated for condensation rates of 29 to 2.3 kg/s.

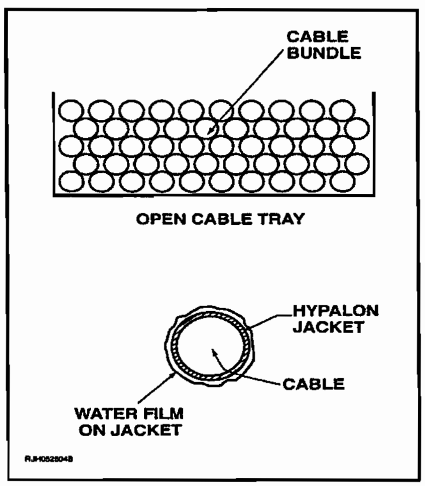
AP1000



AP1000

Issue 6: Organic Iodine Production

- Radiolytic decomposition of electric cable jacket material can produce HCl. If the HCl could escape the uncovered and covered cable trays, it could eventually mix with the containment atmosphere and be delivered to the draining film with the condensing steam.



Issue 6: Organic Iodine Production

- Draining film could be acidified by radiolytic formation of nitric acid or possibly deposition of other acids generated in containment.
- The radiation field in containment varies as the FPs are released and then removed by various deposition mechanisms.
- The estimated range of film pH due to nitric acid generation is 5.6 to 6.5 and 4.8 to 6.7 due to HCl deposition during the first 10 hr of the accident when $[I^-] \geq \sim 10^6$ g-mole/liter.
- A very small integral amount of CsOH (270 gram) from deposited aerosol fission products would be sufficient to neutralize all this nitric and hydrochloric acid for this 10 hour interval. Less than 0.1% of the potentially available CsOH in the core would completely neutralize the film.

Issue 6: Organic Iodine Production

- It is known that a variety of fission product chemical species constitute the source term (ST). CsOH has been judged to be a dominant specie and used as a surrogate chemical specie for the balance of Cs available following CsI formation.
- If the balance of the core Cs is considered to be CsOH, an initial core inventory of approximately 373 kg of CsOH would be available for release. This is several orders of magnitude larger than that estimated to neutralize the draining film.
- Continuing ST research such as in the PHEBUS facility suggests multiple Cs compounds are formed and may be released as aggomerated aerosols of multiple fission product species. Thus, some uncertainty exists regarding the dominant chemical specie but it doesn't eliminate the existence of CsOH as one specie.
- Interestingly, the PHEBUS FTP1 test results indicate that the aerosols injected into containment do not possess a strong acidic character. A small increase in sump pH was measured when the deposited fission products were washed into the sump.

Issue 6: Organic Iodine Production

AP1000

- The limited inventory of CsOH required to neutralize the film leads to the expectation that the draining condensate film pH will be 7 or greater.
- Significant conversion of the deposited iodide (I^-) would not be expected nor would the formation of additional organic iodine.

Issue 6: Organic Iodine Production

AP1000

- As a sensitivity study, it can be assumed that no CsOH is deposited on the draining film.
- Regulatory design basis source term (ST) definition considers that 3% of the elemental iodine that is released from RCS is converted to organic iodine in containment. (Note: 5% of iodine released from RCS is taken as elemental so 0.03 x 0.05 = 0.15% of released iodine is in organic form per the ST definition.) This source term has been used in the AP1000 design basis dose assessments.
- Models have been formulated (NUREG/CR-5950) for estimating the fraction of I^- converted to I_2 in water as a function of the water's pH and the I^- concentration.
- Experimental studies indicate that when a threshold radiation dose to water is exceeded, the conversion will reach the steady-state value. Specifically, NUREG/CR-5950 states that for this model:

  "Experimentally, it has been observed that at dose rates > ~0.3 Mrad/hr, steady state would be reached within a few hours."
- The draining film residence times are much shorter than an hour, which suggests that the steady-state conversion fractions would not be obtained.

AP1000

Issue 6: Organic Iodine Production

- To estimate the potential dose impact of additional organic iodine generation due to the lack of specific pH control of these draining films, it will be assumed that the conversion model applies and conversion of iodine form occurs instantaneously.
- Based on the estimated ranges of film pH for either HNO_3 formation or HCl deposition and the range of iodide concentrations due to CsI deposition, the conversion fraction is estimated to vary between 0 and 0.5 over the 10 hour interval.
- As a conservatism, a conversion fraction of 1.0 is assumed to assess the potential dose impact – not pH dependent.
- All the I^- in the deposited CsI, is assumed to be converted to elemental iodine in the draining film and also assumed to be in the equilibrium distribution of the aqueous, $(I_2)_{aq}$, and the gaseous, $(I_2)_{gas}$, molar concentrations.

AP1000

Issue 6: Organic Iodine Production

- An expression for the iodine partition coefficient, PC (I_2), defined as the ratio of aqueous to gaseous concentrations is provided in NUREG/CR-5950 to be given as a function of the water temperature:

  $\log_{10} PC (I_2) = 6.29 - 0.0149\ T$

  where T is in °K
- The draining condensate film temperature is used to determine the iodine partition coefficient over the 10 hr interval. The fraction of $(I_2)_{gas}$ in the film is assumed to all be released as elemental iodine into the containment gas space. This corresponds to approximately 6.4% of the iodine aerosol released per the design basis ST.
- With the assumed 3% conversion to the organic form for the elemental iodine released to the containment atmosphere, the impact on the organic iodine source term is to increase it from 0.15% to 0.33%.
- The fraction of $(I_2)_{aq}$ that remains in the film is not expected to produce organic iodine since the containment dome and shell are coated with inorganic zinc that does not contain organic material.

AP1000

Issue 6: Organic Iodine Production

- This sensitivity study includes several significant conservatisms:
  - core melt,
  - conservative source that includes 3% conversion of elemental to organic iodine,
  - conservative containment leak rate,
  - conservative weather (χ/Q quantification),
  - zero CsOH release,
  - for control room no operation of HVAC nor re-supply of compressed air until 7 days.
- The impact on the doses of the additional organic iodine is:

  Site Boundary 24.7 rem increases to 24.71 rem

  LPZ 22.8 rem increases to 23.16 rem

  Control Room 4.8 rem increases to 5.07 rem
- The sensitivity study results indicate that sufficient margin exists in the design basis dose assessment to accommodate these postulated consequences of no explicit pH control for the draining condensate films even if no CsOH deposition is considered.

AP1000

Summary of Issue 7

There is experimental evidence that a free-standing steel containment can fail in a catastrophic manner when its failure pressure is exceeded. Such a failure mode can lead to very rapid depressurization and, potentially, to resuspension of fission products that have been previously deposited or settled out. While the surrounding concrete structure of the AP1000 design may impede such a catastrophic depressurization, we would, nevertheless, like to see a sensitivity study on the fission product source term to assess the potential maximum effect on the risk of latent fatalities as compared to the Safety Goal.

Issue 7: Catastrophic Containment Failure

AP1000

- Failure of water cooling of containment vessel is 1E-06/demand
- Even with loss of water cooling, likelihood of catastrophic failure of the AP1000 steel containment due to overpressure is low, approximately 0.02, given failure of PCS cooling and no corrective operator actions.
- At least 24 hours are available for operators to take preventive actions. Any of the following actions could prevent the possibility of containment failure:
  - Climb up to the PCS valves and manually crank open one of the PCS drain valves that failed to open remotely.
  - Align another water supply to the outside surface of the containment; connections are provided for PCS Ancillary Water, Fire Water and Demin water.
  - Vent the containment to relieve the excess pressure.
- SAMG procedures guide operators to take these actions.

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Issue 7: Catastrophic Containment Failure

AP1000

- The potential for rapid containment depressurization causing resuspension of deposited fission products for a set of LWR reference plants was evaluated as part of the IDCOR program.
- The range of containment volumes and catastrophic break sizes include the applicable AP1000 characteristics.
- The IDCOR report concludes that resuspension due to dispersion following catastrophic containment failure would be insignificant even for large failure areas (10 m^2) and dry particle deposits.
- Wetted deposits are harder to disperse than dry deposits (deposited and settled aerosols).
- The conditions inside the AP1000 containment with or without failure of the PCS to remove decay would remain wet with steam.

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Issue 7: Catastrophic Containment Failure

AP1000

- The expected wet physical state of the deposited fission products greatly reduces the potential for resuspension.
- Thus, it is also concluded for AP1000 that catastrophic containment failure would not significantly enhance the fission product source term.
- The risk significance of any source term increase due to resuspension would be very small since the frequency of catastrophic failure induced releases is very low.
- This low frequency and the availability of preventive operator actions to potential catastrophic containment failure would prevent any discernible impact on compliance with the Safety Goal.

Safety Goal Risk Measures

AP1000

- **NRC Safety Goal Policy Statement**
  - no significant additional risk to life and health
- **Quantitative Health Objectives - metrics for Safety Goal**
  - fatality and cancer risks < 0.1% of sum from other causes
- **Quantitative Health Objectives - numerics**
  - risk of prompt fatality < 5E-07 per reactor year
  - risk of latent cancer fatality < 2E-06 per reactor year
- **AP1000 PRA Results**
  - risk of prompt fatality 8.4E-11 per reactor year
  - risk of latent cancer fatality 8.6E-10 per reactor year

AP1000

AP1000 Risk Quantification

- The AP1000 comparison to Safety Goal shows that additional uncertainties associated with severe accident analysis, such as those discussed today, can readily be tolerated without challenging the Safety Goal measures.

/2

AP1000 Design Certification Review

UNITED STATES NUCLEAR REGULATORY COMMISSION

June 3, 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 the ACRS interim letter issues

AP1000 Review Status

- 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 Draft SER with 174 Open Items
- May 18, 2004 – NRC provided responses to the issues in the ACRS's Interim Letter
- Processing Final SER

U.S. Nuclear Regulatory Commission Slide 3 June 3, 2004

Upcoming Schedule Milestones

- 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
- December 2005 – Final Design Certification Rule issued

U.S. Nuclear Regulatory Commission Slide 4 June 3, 2004

Issue 1 - Automatic Depressurization System (ADS)-4 Squib Valve Function

- Issue:
  - Agreed with the staff that ITAAC assures the valves meet the design basis specifications
- Response:
  - Simple design - ASME Code Section III Class 1
  - Redundant and Diverse Actuation
  - PRA Sensitivity Study
  - ITAAC

Issue 2 - Assurance of Long-Term Cooling (Strainer Blockage)

- Issue:
  - AP1000 is a robust design to prevent screen blockage.
  - Recommended ITAAC to ensure compliance with GSI 191
- Response:
  - ITAAC
  - COL Action Items
  - Containment recirculation screens redesign

Issue 3 - Code Deficiencies

- Issue:
  - When deficiencies are identified in codes, the weaknesses should be corrected.
- Response:
  - TRACE code is being assessed using APEX AP1000, ATLATS, and UPTF data.
  - If desired, the staff can describes the results to the ACRS when completed.

Issue 4 - Range of Pi-Group Values

- Issue:
  - The staff should verify that a Pi group range of 0.5 to 2 is appropriate.
- Response:
  - This range has been used as a de facto standard in scaling analyses.
  - This issue is generic, not an issue specific only to AP1000.
  - Staff plans to develop and document procedures to define appropriate Pi group range.

Issue 5 - In-Vessel Retention/Fuel-Coolant Interactions

- Issue:
  - IVR assessment needs to consider the effects of exothermic intermetallic reactions.
  - Would like to review the FCI models and justification that intermetallic reactions will not result in energetic FCI that could fail the containment.
- Response:
  - Staff provided the ACRS a copy of their contractors IVR and FCI report for AP1000

Issue 6 - Organic Iodine Production

- Issue:
  - Acidification of water film on the inside of the containment wall (as a result of radiolysis of organic material) could result in re-evolution of iodine in the gaseous organic form.
- Response:
  - W first presented their sensitivity study during a public meeting yesterday.
  - The staff plans to perform an audit of the sensitivity study within the next week.
  - The staff may perform independent evaluations.
  - If desired, the staff can describes its evaluation to the ACRS on June 25, 2004.

Issue 7 - Catastrophic Failure of the Steel Containment

- Issue:
  - A free-standing steel containment can fail in a catastrophic manner when its failure pressure is exceeded. This failure mode can lead to rapid depressurization and resuspension of deposited fission products.
  - Like to see a sensitivity study on the fission product source term to assess the effect on the risk of latent fatalities as compared to the Safety Goal.
- Response:
  - Frequency of catastrophic containment failures are small
  - Resuspension would not have a noticeable impact on the Commission's safety goals.

Additional ACRS Comments

- Materials
  - Comment:
    - Ongoing and future studies may suggest material and environmental changes that will be addressed at the COL stage.
  - Response:
    - Clarified the Part 52 change process
- Aerosol Removal Coefficient (lambda)
  - Comment:
    - The ACRS looks forward to reviewing the staff's aerosol removal analysis using the MELCOR code.
  - Response:
    - Provided staff evaluation

Conclusion

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

U.S. Nuclear Regulatory Commission   Slide 13   June 3, 2004

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ACRS Meeting
U. S. Nuclear Regulatory Commission
Rockville, Maryland
June 3, 2004

FCI MODELING USING PM-ALPHA/ESPROSE.m

by:

H. Esmaili and M. Khatib-Rahbar
Energy Research, Inc.
6167 Executive Blvd.
Rockville, Maryland 20852

Advisory Committee on Reactor Safeguards (ACRS) Meeting

Rockville, Maryland

June 3, 2004

ERI
Energy Research, Inc.

PM-ALPHA/ESPROSE.m

- 2D version released to ERI in mid-1990s by NRC
- Developed by UCSB (Theofanous, et al.)
- Newer version also made available to ERI but not used in the present analyses
- Numerical approach based on the KFIX code
- Models have some experimental validation basis

ERI
Energy Research, Inc.

PM-ALPHA

- PM-ALPHA simulates the premixing phase
- Uses multifield Eulerian formulation
  - Fuel melt
  - Liquid coolant
  - Vapor
- Constitutive laws provide interfacial heat & mass transfer, phase change, fuel breakup through a number of correlations
  - Breakup model solves an interfacial area transport equation:

$$\frac{\partial A_f}{\partial t} + \nabla \cdot \left(A_f \overline{u_f}\right) = S_{fr} + S_b$$

A_f is fuel surface area, u_f is fuel velocity,S is the source term representing fragmentation & breakup

ERI
Energy Research, Inc.

PM-ALPHA (Cont)

- The breakup mechanism is given by:

$$S_b = -\frac{6\theta_f}{D_f^2}\left(\frac{dD_f}{dt}\right) = -\frac{6\theta_f}{D_f^2}\left[\max\left(\frac{|\overline{u_f}|}{\beta_b}, \frac{D_f}{L}|\overline{u_f}|\right)\right]$$

D_f = melt diameter
θ_f = melt volume fraction
β_b = User specified paramter
L = Total available fall distance

Breakup process is terminated once particle diameter reaches capillary size:

$$\sqrt{\sigma / g(\rho_f - \rho_l)}$$

ERI
Energy Research, Inc.

ESPROSE.m

- ESPROSE.m simulates the propagation phase of the explosion once it is triggered.
- Uses multi-field Eulerian formulation
  - Fuel melt
  - Liquid coolant
  - m-field (vapor field ahead of the explosion front – denotes a homogeneous mixture of fragmented debris and coolant behind the explosion from where fragmentation occurs)
- Fuel fragmentation is principal mechanism that drives the propagation phase of the steam explosion

ERI
Energy Research, Inc.

ESPROSE.m (Cont.)

- Rate of fragmentation for a single melt particle:

$$\frac{dm_f}{dt} = \frac{\rho_f \pi D_f^3}{6 t_{fr}}$$

- Where

  D_f = fuel particle diameter

  ρ_f = particle density

  t_{fr} = characteristic time for fragmentation

- The dimensionless fragmentation time is defined by:

$$t_{fr}^{+} = \frac{\left|\overline{u_f} - \overline{u_i}\right| t_{fr}}{D_f} \left(\rho_f \big/ \rho_i \right)^{-1/2} = \beta_f \big/ Bo^{1/4} = \frac{\beta_f}{\{3 C_d \rho_i D_f \left|\overline{u_f} - \overline{u_i}\right|^2 / 16\sigma\}^{1/4}}$$

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Energy Research, Inc.

ERI

ESPROSE.m (Cont.)

- Therefore:

$$\frac{dm_f}{dt} = \frac{\pi D_f^2 \left|\vec{u}_f - \vec{u}_c\right|}{6 t_b^*} (\rho_f - \rho_c)^{1/2}$$

$$= \left(\frac{3}{6}\right)^{1/4} \left(\frac{\pi}{6\beta_f}\right) C_d \left(\frac{1}{\sigma}\right)^{1/4} D_f^{3/4} \left|\vec{u}_f - \vec{u}_c\right|^{3/2} \rho_c^{3/4} \rho_f^{1/2}$$

However, if $\frac{dm_f}{dt}$ ¡ÝUser specified rate then,

$$\frac{dm_f}{dt} = \text{User specified rate}$$

ERI Energy Research, Inc.

ESPROSE.m (Cont.)

- Total rate of fragmentation per unit volume is:

$$F_r = n_p \frac{dm_f}{dt} = \frac{6\theta_f}{\pi D_f^3} \frac{dm_f}{dt} = \frac{\rho'_f}{t_{fl}}$$

Where

n_p = number of particles per unit volume
θ_f = volume fraction of the melt
ρ'_f = macroscopic density of the melt

ERI Energy Research, Inc.

ESPROSE.m (Cont.)

- To account for presence of both vapor and liquid in the mixture:

$$F_r = \rho_f^{'}\left(\frac{\alpha}{t_{fl}} + \frac{1-\alpha}{t_{fm}}\right)$$

F_r is introduced as the source term in the fuel and debris continuity equations. It also appears in the fuel and m-field energy equation

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ESPROSE.m (Cont.)

- The rate of energy addition to the m-field is

$$E_f = F_r I_f = F_r\left\{C_{pf}(T_f - T_{ref}) + I_f^*\right\}$$

I_f = Internal enegy of the melt

I_f^* = heat of fusion of the melt

C_{pf} = heat capacity of the melt

T_f = temperature of the melt

T_{ref} = reference temperature

Therefore, explosive load is a function of <u>melt quantity, temperature, particle size and rate of fragmentation</u>

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ST Analysis: evaluation of aerosol removal rates (1)

General Remarks

Application of AST requires T-H scenario and aerosol models not specified by NUREG-1465.

Westinghouse calculation based on a single T-H scenario and mechanistic aerosol model.

Adopted scenario (3BE-1) is a double-ended DVI 4" line break with a failure to activate the intact train. The spillage floods the containment and spills into the vessel.

- scenario acceptance based on the following:

    It is representative of the "3BE" accident class, which is the dominant contributor to the core damage frequency for the AP1000.

    The T-H conditions for 3BE accidents are typical for majority of severe accident sequences (fully depressurized and reflooded.)

    AST was intended to be representative of low pressure core-melt accidents.

The staff accepts the 3BE-1 accident sequence as a basis for the AP1000 dose analysis.

ST Analysis: evaluation of aerosol removal rates (2)

Westinghouse analysis

Initially Westinghouse intended to use AP600 removal rates for AP1000 aerosol. After the staff raised concerns, Westinghouse submitted BE analysis using MAAP calculated T-H and aerosol mechanistic code STARNAUA. Credit was given for gravitational settling, diffusiophoresis (steam condensation) and thermophoresis (temperature gradient).

Staff accepted these phenomena as removal mechanisms, however questioned the Westinghouse calculated removal rate values.

Staff's analysis

Staff performed an independent aerosol removal rates analysis with an alternative T-H (MELCOR) as an input to Monte Carlo sampling. MELCOR calculated removal rates were also reviewed.

ST Analysis: evaluation of aerosol removal rates (3)

Staff's analysis

13 parameters affecting aerosol behavior were sampled to achieve 95% confidence level (200 tries.) Engineering judgement was used for the choice of parameters as well as for the range and distribution of their values. The sampled parameters are:

- aerosol size
- distribution,
- aerosol void fraction
- particle shape factors,
- aerosol material density,
- non-radioactive aerosol mass,
- particle slip coefficient,
- sticking probability for agglomeration,
- boundary layer thickness for diffusion deposition,
- thermal accommodation coefficient for thermophoresis,
- ratio of thermal conductivity of particle to gas,
- turbulent energy dissipation, and
- multipliers on heat and mass transfer to containment shell.

ST Analysis: evaluation of aerosol removal rates (4)

Regulatory issue

Traditional regulatory approach is to accept a "bounding" value. In the case of probability distributions the widely accepted bounding values are 5% or 95%-tiles.

For AP1000, staff used the median (50%-tile) for the following reasons:

- staff believes that the selected scenario belongs to a "worst case" category,
- median value is the least affected by the user's subjective judgements,
- since the choice of the initial ranges and distributions of the selected parameters is highly subjective, staff introduced a "conservative bias" in its selection,
- there is a precedence of staff accepting the median value in a pilot case of Perry steam line deposition, based on RES opinion that it is appropriate given other conservatisms built into the other parts of the analysis,
- staff's dose calculation code requires yet another "averaging" of the removal rates for the specified time periods, introducing additional subjectivity to the analysis,
- the fully integrated MELCOR calculated removal rates are mostly well above the 5%-tile.

ST Analysis: evaluation of aerosol removal rates (5)

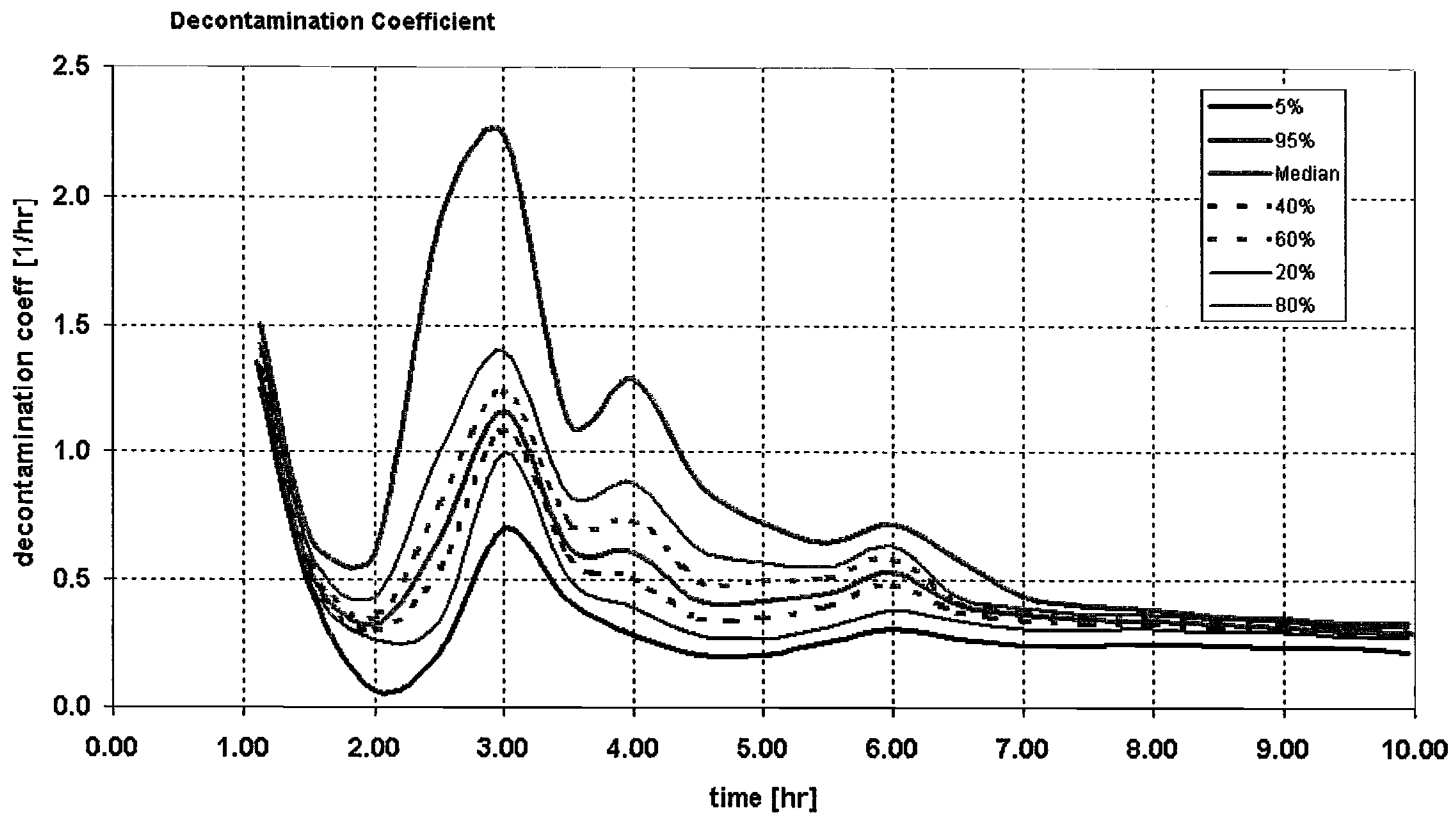
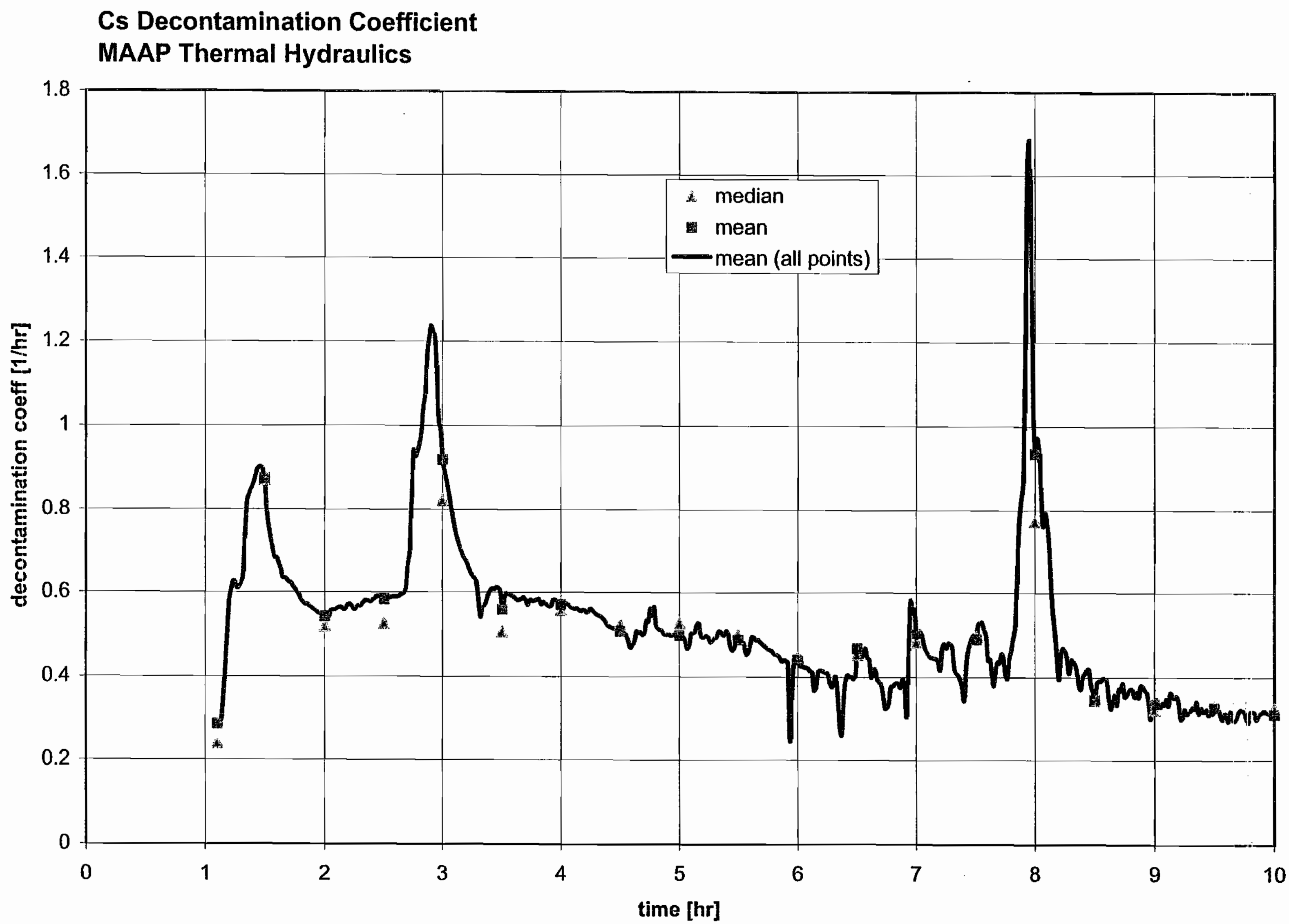
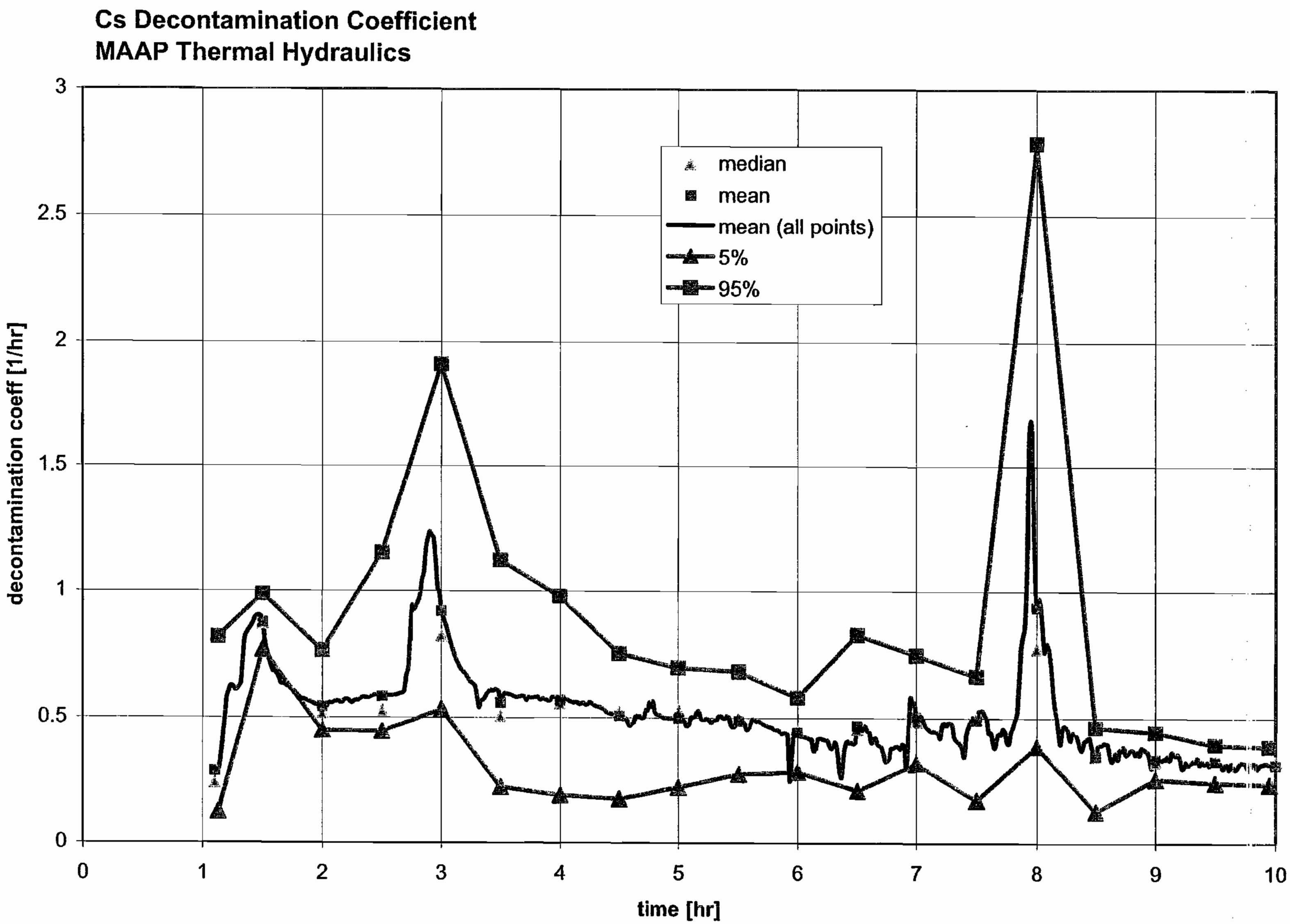


Figure 1: Uncertainty bands of aerosol removal rates (lambdas)


Cs Decontamination Coefficient
MAAP Thermal Hydraulics
median
mean
mean (all points)
decontamination coeff [1/hr]
time [hr]
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
1
2
3
4
5
6
7
8
9
10



Cs Decontamination Coefficient
MAAP Thermal Hydraulics
decontamination coeff [1/hr]
time [hr]
median
mean
mean (all points)
5%
95%
0
0.5
1
1.5
2
2.5
3
0
1
2
3
4
5
6
7
8
9
10

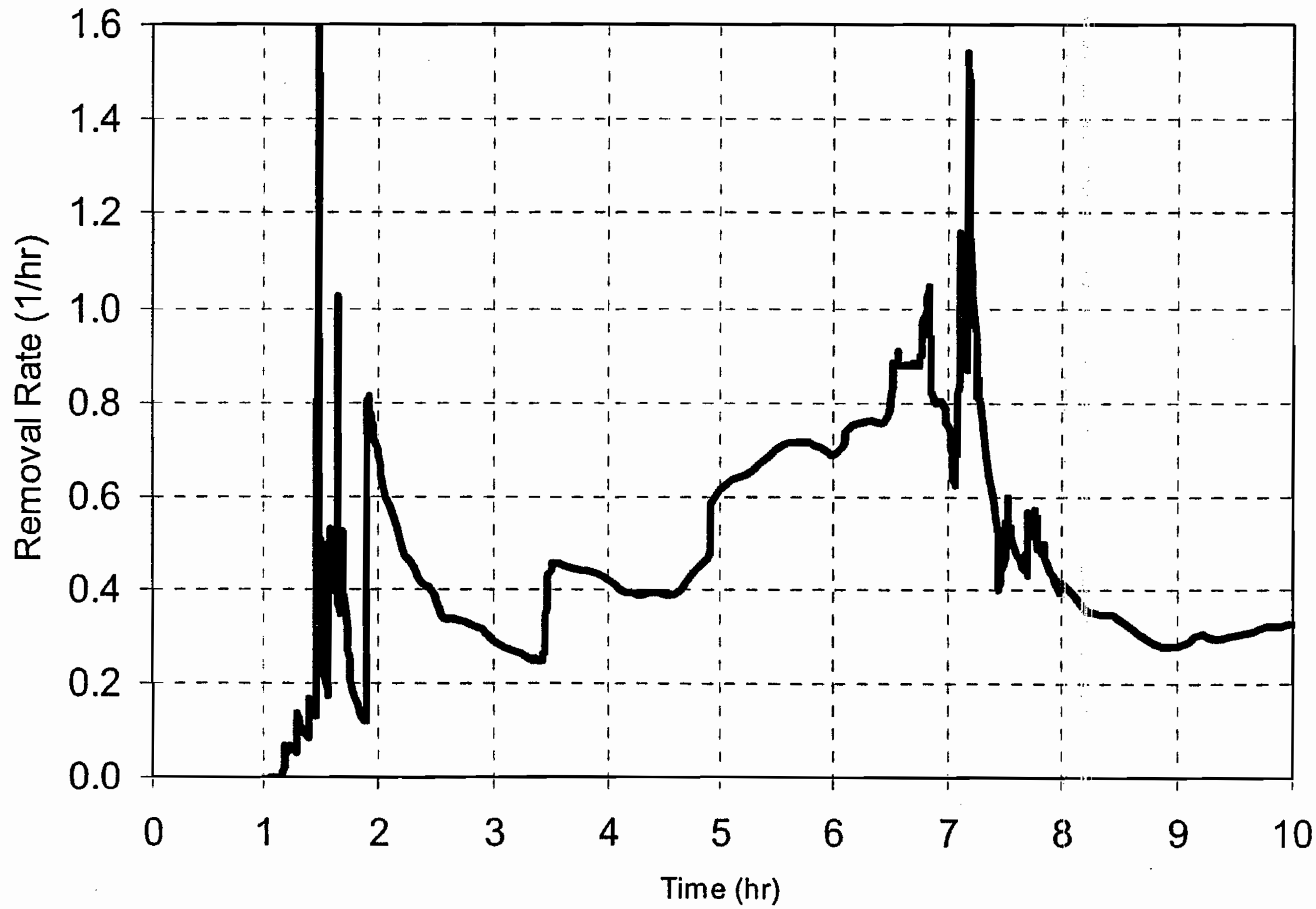
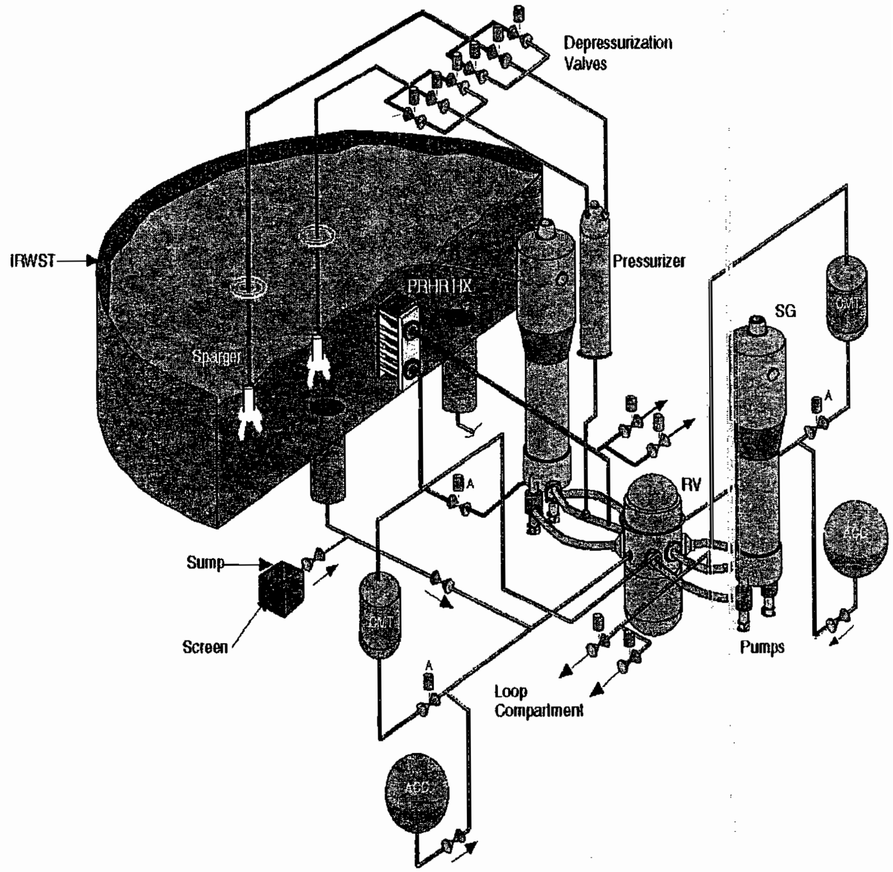


Figure 3 Average Aerosol Rate Removal Constant in Containment, MELCOI Scenario 3BE with Vessel Reflood Through Broken DVI Line

AP1000 Passive Core Cooling System

- AP600 System Configuration Retained
- Capacities Increased to Accommodate Higher Power
  - CMT Increased 25%
  - IRWST Injection Increased 80%
  - Sump Injection Increased 110%
  - ADS 4 Increased 90%
  - PRHR HX Capacity Increased 72%
- System Performance Maintained
  - No core uncovery for SBLOCA
    - DVI line break
  - Large margin to PCT limit



Standard Review Plan Update Process

Presentation to the Advisory Committee on Reactor Safeguards

Rob Kuntz
Aida Rivera

Organizational Effectiveness Branch
Program Management, Policy Development and Planning Staff
Office of Nuclear Reactor Regulation

June 3, 2004

5

Purpose

- Present summary of changes to SRP sections 5.2.3, 5.3.1, and 5.3.3 and request a waiver of ACRS review of the revised sections

- Inform the ACRS of NRR's process and plan to update SRP sections during FY05 and FY06.

- Obtain ACRS agreement on the potential work load, and the schedule established for SRP updates during FY05.

Agenda

- Summary of changes
- Background
  - October 31, 2003 SRM
- SRP development process
- Plan
- Summary

Summary of changes

SRP Section	Technical changes	Editorial changes	Added /revised references	Total number of changes
5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS	0	5	17	22
5.3.1 REACTOR VESSEL MATERIALS	0	12	22	34
5.3.3 REACTOR VESSEL INTEGRITY	0	4	7	11

- Since technical changes were not required to update these Standard Review Plan sections, ACRS review is not considered to be necessary. The technology for light water reactor applications in the areas covered by these sections has remained essentially unchanged.

Background

- October 31, 2003 SRM - SRM in response to October 2, 2003 ACRS meeting.
- SRM asked staff to provide the Commission the status, approach and plans for maintaining a current and effective set of guidance documents (including SRP).

Background

- Prior to the issuance of the SRM, NRR staff had begun preliminary work on an SRP update plan.
- Included:
  - Scoping process
  - Prioritization process
  - Scheduling

Scoping Process

- Purpose- determine the extent of the update and estimate the resources required to complete the update.
- Questions asked during the scoping:
  - What version is currently used for reviews of license amendments?
  - Is there guidance that has superceded the version used?
  - Does updating the SRP Section require public comment, ACRS comment, and/or CRGR comment?
  - Does updating the SRP section require updating of other guidance?
  - Estimated number of hours required to complete the revision.
- Updating the entire SRP will require approximately 35 FTE.

Prioritization Process

- Purpose – create a prioritized list of SRP sections that will be used to determine which SRP sections are scheduled to be updated each fiscal year.
- 3 criteria used to prioritize the sections:
  - Safety Significance
  - Recent Industry Activities
  - Stakeholders/Commission Interest
- As resources are allocated in the budget for updating the SRP, the highest priority SRP sections will be updated.

Plan

- Updates to the SRP will be accomplished according to NRR Office Instruction (OI) LIC-200, "Standard Review Plan (SRP) Process."
- The budget proposed for SRP work for FY05 and FY06 is approximately 6 FTE for each fiscal year.
- NRR's plan is to update around 35 SRP sections in FY05 and FY06.

Bundling

- Purpose - create groups (bundles) of SRP sections in order to make the SRP update process easier on both NRR staff and ACRS.
- Examples of topics for bundles:
  - Reactor Vessel – materials and internals
  - Containment
  - Instrumentation and control systems
- Results -
  - FY05 - 35 sections divided in 13 bundles
  - FY06 - 35 sections divided in 11 bundles

Scheduling

- Intermediate milestones were established to distribute resources for review of SRP sections throughout the year.

- Each bundle for fiscal year 2005 (FY05) will be completed within a specified quarter.

- The quarter was assigned based on the estimated hours the staff provided during the scoping process and resource availability.

Summary

- The update of the SRP will be accomplished according to NRR OI LIC-200, "Standard Review Plan (SRP) Process."

- During FY05, ACRS will be receiving 13 bundles of SRP updates, approximately 3 bundles per quarter.

Objectives

- ACRS response to waiver request for review of revised SRP sections 5.2.3, 5.3.1, and 5.3.3
- ACRS agreement on the potential work load, and the schedule established for SRP updates during FY05.

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Schedule of SRP Sections to be Updated in FY05

Revision Quarter	Bundle	Section	Title	Primary Division	ACRS Review
1st	1	4.5.2	Reactor Internal and Core Support Materials	DE	1st quarter (December 2004)
	2	6.5.2	Containment Spray as a Fission Product Cleanup System	DE	1st quarter (December 2004)
	3	8.4	Station Blackout [Future]	DE	2nd quarter (February 2005)
2nd	4	3.9.6	Inservice Testing of Pumps and Valves	DE	2nd quarter (March 2005)
		3.9.5	Reactor Pressure Vessel Internals	DE	
		3.9.3	ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures	DE	
		3.9.4	Control Rod Drive Systems	DE	
	5	7.1-A	Acceptance Criteria & Guidelines for I&C Systems Important to Safety	DE	2nd quarter (March 2005)
		7.3	Engineered Safety Features Systems	DE	
		7.4	Safe Shutdown Systems	DE	
	6	2.5.2	Vibratory Ground Motion [Future]	DE	3rd quarter (April 2005)
3rd	7	6.2.1	Containment Functional Design	DSSA	3rd quarter (June 2005)
		6.2.5	Combustible Gas Control in Containment	DSSA	
		6.4	Control Room Habitability System	DSSA	
		6.2.1.3	Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents	DSSA	
		6.2.3	Secondary Containment Functional Design	DSSA	
	8	19.1	Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed	DSSA	3rd quarter (June 2005)
	9	3.8.2	Steel Containment	DE	4th quarter (July 2005)
		3.8.1	Concrete Containment	DE	
		3.8.3	Concrete and Steel Internal Structures of Steel or Concrete Containments	DE	
	10	14.2.1	Generic Guidelines for Extended Power Uprate Testing Programs	DIPM	4th quarter (September 2005)
		12.5	Operational Radiation Protection Program	DIPM	
4th	12	9.4.1	Control Room Area Ventilation System	DSSA	4th quarter (September 2005)
		2.3.4	Short-Term Dispersion Estimates for Accidental Atmospheric Releases	DSSA	
		3.5.1.6	Aircraft Hazards	DSSA	
		9.5.1	Fire Protection Program	DSSA	
	11	4.2	Fuel System Design	DSSA	1st quarter FY06 (October 2005)
		4.3	Nuclear Design	DSSA	
		4.4	Thermal and Hydraulic Design	DSSA	
		4.6	Functional Design of Control Rod Drive System	DSSA	
		NEW*	Spent Fuel Criticality	DSSA	
		5.4.12	Reactor Coolant SystemHigh Point Vents	DSSA	
	13	19.0	Probabilistic Risk Assessment	DSSA	1st quarter FY06 (October 2005)

Note: Schedule is subject to change
* Number of section not yet decided

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G:PPHandout

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
513	11	11.1

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

Authors

JOHN T. LARKINS

List of Documents Attached

PLANNING & PROCEDURES MINUTES

11

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

From Staff Person
JOHN T. LARKINS

June 3, 2004
G:PlanPro(ACRS):ppmin.513

INTERNAL USE ONLY

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

The ACRS Subcommittee on Planning and Procedures held a meeting on June 1, 2004, in Room T-2B3, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 1:30 p.m. and adjourned at 2:35 p.m.

ATTENDEES
M. Bonaca
G. Wallis
S. Rosen

ACRS Staff
J. T. Larkins
R. P. Savio
S. Duraiswamy
J. Gallo
M. Sykes
M. Snodderly
H. Nourbakhsh
J. Delgado
C. Santos

NRC Staff
D. Weaver

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

Member assignments and priorities for ACRS reports and letters for the June ACRS meeting are attached (pp. 7-9). Reports and letters that would benefit from additional consideration at a future ACRS meeting were discussed.

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the June ACRS meeting be as shown in the attachment (pp. 7-9).

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through September 2004 is attached (pp. 7-9). The objectives are to:

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

During this session, the Subcommittee also discussed and developed recommendations on items included in Section IV of the Future Activities list (pp. 10).

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 met with the NRC Commissioners on June 2, 2004, to discuss the following topics:

- Overview (MVB)
- PWR Sump Performance (JDS)
- PRA Quality (for decisionmaking) (GEA)
- Risk-Informing 10 CFR 50.46 (WJS)
- NRC Safety Research Program Report (DAP)
- ESBWR Pre-Application Review (TSK)
- Interim Review of the AP1000 Design (TSK)

Any follow-up items resulting from this meeting should be discussed by the Committee at the June ACRS meeting.

RECOMMENDATION

The Subcommittee recommends that the Committee discuss the follow-up items resulting from the meeting with the NRC Commissioners and develop a course of action to bring them to closure expeditiously.

4) Revision to ACRS Action Plan

A copy of the revised Action Plan was sent to the members following the June ACRS meeting. Members were requested to provide their comments to Mrs. Weston by June 4, 2004. The current revision to the Action Plan reflects incorporation of the limited comments received from the members.

RECOMMENDATION

The Subcommittee recommends that the Committee endorse the revised Action Plan for publication with a provision that the ACRS Executive Director make editorial/clarification changes, as warranted, prior to publication.

5) Visit to a Nuclear Plant and Regional Office

Several members (Bonaca, Ford, Leitch, Ransom, Rosen, Sieber, and Wallis) are scheduled to visit the D.C. Cook Nuclear Plant on Wednesday, June 9, 2004, and the NRC Region III Office on Thursday, June 10, 2004. Reservations have been made at the Hyatt Lisle Hotel, (630-852-1234) 1400 Corporetum Drive, Lisle, Illinois.

RECOMMENDATION

The Subcommittee recommends that Mrs. Weston provide additional details on this matter, including an agenda for the meeting with the Region III personnel.

6) Tour of Test Facilities Used for the ACR-700 Design

During the May 2004 ACRS meeting, Drs. Ford, Kress, Ransom, and Wallis expressed interest in touring the Chalk River Facility used for the ACR-700 design and participating in a joint meeting of the ACRS Subcommittees on Future Plant Designs and on Materials and Metallurgy to discuss various aspects of the ACR-700 design, including materials issues. The tour and the meeting were originally scheduled for July 25-30, 2004. However, a workshop regarding the AP1000 design has been scheduled by the NRC staff on July 26-29, 2004, to be held in China. Dr. Kress was invited to join the NRC Panel to participate in this workshop. During the 512^{th} ACRS meeting in May 2004, the Committee approved Dr. Kress' participation in such workshop. Accordingly, the trip to the Chalk River Facility in Canada will be postponed, possibly till August, September, or October 2004.

RECOMMENDATION

The Subcommittee recommends that the members select the month (August, September, or October, 2004) and dates for touring the Chalk River Facility and holding a meeting. Also, Dr. Kress should prepare a trip report following the Workshop in China on the AP1000 design.

7) EDO Response to the Anonymous E-mail Regarding the NRC Staff's Process for Reviewing the TRACE Computer Code

A member of the public sent an anonymous e-mail to Dr. Wallis on February 20, 2004, criticizing the process being used by the NRC staff in the development and review of the TRACE computer code. As agreed to by the Committee during its March 2004 meeting, the ACRS Executive Director referred this matter to the EDO for action.

In a memorandum dated April 16, 2004, to the ACRS Executive Director (pp. 11-12), the EDO addressed the issues raised in the anonymous e-mail. In this response, the EDO states that the concerns expressed in the anonymous e-mail can be discussed by the ACRS Subcommittee on Thermal-Hydraulic Phenomena during its review of the TRACE code.

RECOMMENDATION

The Subcommittee recommends that Dr. Ransom, lead member for reviewing the TRACE code, discuss the concerns expressed by the author of the anonymous e-mail and the associated staff's response during future meetings involving the review of the TRACE code and provide a report to the Committee on the findings.

8) Vermont Yankee Power Uprate

In letters dated March 15 and 31, 2004 (pp. 13-18) to the NRC Chairman, the State of Vermont Public Service Board requested that the ongoing NRC review of the Vermont Yankee power uprate request by Entergy include several new facets, including an "independent assessment" of the plant.

On May 4, 2004, NRC Chairman Diaz responded to the Vermont Public Service Board (pp. 19-22) and explained that the NRC has decided to conduct a detailed engineering inspection that it believes will be appropriate for addressing its oversight responsibilities and is also responsive to the concern expressed by the Vermont Public Service Board.

The Chairman also noted that the NRC has been developing a new engineering inspection program which it intends to pilot at selected plants. The new program incorporates the best practices of existing and past engineering inspections and it would be appropriate to conduct this new engineering inspection at Vermont Yankee.

The ACRS will also review the Vermont Yankee power uprate request. The NRC staff will provide the results of its review efforts, including relevant inspection findings, to the ACRS for review. After the ACRS completes its review, it will make an independent recommendation regarding whether the proposed power uprate amendment should be approved.

RECOMMENDATION

The Subcommittee recommends that the Thermal-Hydraulic Phenomena Subcommittee, which is responsible for reviewing the Vermont Yankee power uprate application, take note of the NRC Chairman's statement that the ACRS will review the inspection findings associated with the Vermont Yankee power uprate.

9) Interview of Candidates for Potential Membership on the ACRS (Closed)

During the June ACRS meeting, the ACRS Member Candidate Screening Panel and the ACRS members interviewed five candidates for potential membership on the ACRS.

RECOMMENDATION

The Subcommittee recommends that the members provide feedback on the Candidates to the ACRS Chairman, who will meet with the Screening Panel on June 4, 2004, to discuss potential recommendation for a new member.

10) ACRS Assessment of the Quality of the NRC Research Programs

In a letter dated April 26, 2004, the ACRS outlined a strategy for assessing the quality of the NRC research projects. Out of eight projects provided by RES, the Committee selected the following projects for assessment during the remainder of FY 2004 and made assignments as noted below.

- Sump Blockage - Mr. Rosen (Chair), Kress, and Wallis
- MACCS Code - Dr. Kress (Chair), Apostolakis, and Sieber

In a letter dated May 20, 2004 (pp. 23), Mr. Paperiello, RES Director, states that RES appreciates the Committee's willingness to assist RES in assessing the quality of the NRC research projects.

Since the Committee has committed to provide a summary report on its assessment of the above two research projects at the end of this fiscal year, it would be helpful if Mr. Rosen and Dr. Kress provided a status report on the progress of their Panels in assessing the quality of the assigned projects in July 2004.

RECOMMENDATION

The Subcommittee recommends that Mr. Rosen and Dr. Kress provide a status report at the July 2004 ACRS meeting.

11) Member Issues

Mr. Sieber suggests (pp. 24) that instead of sending a CD, which contains background materials for the ACRS full Committee meetings, the cognizant staff engineers e-mail the meeting documents to the members.

Even though Mr. Sieber's suggestion is a good one, we would like to bring the following requirement of the Federal Advisory Committee Act (FACA) to the attention of the members. FACA requires that all documents provided to the Committee and all reports, letters, agendas, and studies prepared by or for the Committee be maintained for the duration of the Committee. Having all documents for each meeting stored in a single CD will be a more efficient way of complying with the FACA requirement. As has been the practice, the staff engineers will e-mail the status reports and background documents to the members ahead of the meeting.

RECOMMENDATION

The Subcommittee recommends that the ACRS staff keep sending CDs containing documents for ACRS meetings and that the cognizant engineers e-mail the status reports and background documents to the members prior to the meeting.

12) Travel Request

Mr. Rosen requests (pp. 25) Committee approval and support to attend the NEI Fire Protection Information Forum to be held in Florida between September 18-23, 2004.

RECOMMENDATION

The Subcommittee recommends that the Committee approve Mr. Rosen's travel request. Mr. Rosen should prepare a trip report following the conference.

OIC
JTL 5/28

ANTICIPATED WORKLOAD
JUNE 2-4, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	--	Snodderly	Draft Final 10 CFR 50.69. Risk-Informed Categorization and treatment of Structures, Systems, and Components	A	To support the staff schedule	–
		Nourbakhsh	Metrics for quantitative assessment of the effectiveness (quality) of the research projects	--	--	--
Bonaca	Leitch	Sykes	Revised License Renewal Review Process - **Information Briefing**	–	–	–
	All Members	Larkins	Meeting with the NRC Commissioners (June 2, 2004, 1:30pm -3:30pm)	–	–	–
Ford	--	Duraiswamy	Update to SRP Sections (5.2.3, 5.3.1, and 5.3.3) and SRP update process	Larkinsgram	–	–
Kress	--	El-Zeftawy	Response to the March 17, 2004 ACRS Report on AP1000	--	--	–
Sieber	Apostolakis	Sykes	Digital I&C Systems research activities	A	To provide Committee's views	--

ANTICIPATED WORKLOAD
JULY 7-9, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Kress	–	El-Zeftawy/ Duraiswamy	Draft final 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants"	A	To support the staff schedule	–
		El-Zeftawy	Final SER associated with the AP1000 design certification	A	To support the staff schedule	–
Powers	Rosen/Kress	Nourbakhsh/ Duraiswamy	Status of activities associated the assessment of the quality NRC research projects	–	–	–
	–	Nourbakhsh/ Duraiswamy	Response to SRM on divergence in regulatory approaches between U.S. and several other countries (**Tentative**)	B	To respond to the Commission SRM	–
Rosen	–	Sykes	Proposed Rule on the Post-Fire Operator Manual Actions (**Tentative**)	A	To support the staff schedule	–
Shack	–	Snodderly	Draft NUREG on 10 CFR 50.46 LB LOCA frequency reevaluation	(Report as Needed)	–	–
Sieber	–	Sykes/Santos	Proposed Generic Communication on the use of ultrasonic flow measurement devices for measuring feedwater flow rates in nuclear plants	A	To support the staff schedule	–
Wallis	–	Caruso	Generic Letter on potential impact of debris blockage on emergency recirculation during design-basis accidents at PWRs	A	To support the staff schedule	–

ANTICIPATED WORKLOAD
SEPTEMBER 8-11, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Bonaca	–	Sykes	Final review of the License Renewal Application for Dresden and Quad Cities Nuclear Plants	A	To support the staff schedule	–
		Savio/Major	Safeguards and Security matters	A	To provide Committee's views	–
Powers	–	Weston	MOX Fuel Fabrication Facility (**Tentative**)	A	To support the staff schedule	–
	Rosen/ Kress	Nourbakhsh/ Duraiswamy	Assessment of the quality of NRC research on sump blockage and on MACCS code	Report to be completed in October	–	–
Ransom	Kress	Caruso/Weston	Maximum Extended Load Line Limit Analysis Plus (MELLA +) Licensing Topical Report **(Tentative)**	B	To provide feedback to the staff	–
		Caruso	Proposed resolution of GSI-185, "Control of Recriticality Following Small-Break LOCAs in PWRs" **(Tentative)**	A	To support the staff schedule	–
Shack	–	Nourbakhsh/Santos	PTS technical basis reevaluation **(Tentative)**	A	To support the staff schedule	–
Sieber		Weston	Mitigating System Performance Index (MSPI) program	B	To provide Committee's reviews	–
Wallis	–	Caruso	Safety Evaluation Report for the Evaluation Guidelines Regarding Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at PWRs	A	To support the staff schedule	–

Items Requiring Committee Action

1 **Review draft Regulatory Guide, DG-1130, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants"** (Open)

Member: John Sieber **Engineer:** Marvin Sykes

Estimated Time:

Purpose: Determine a Course of Action

Priority: High

Requested by: RES Satish Aggarwal

The staff provided copies of the draft regulatory guide DG-1130, requesting that the ACRS review the draft final Guide after reconciliation of public comments. The staff plans to issue this draft regulatory guide for public comment. The draft regulatory guide was developedto reflect the current state-of-the-technology reflected in IEEE Std 7-4.3.2-2003 and provides guidance specific to computer-based safety-system security.

The Subcommittee recommends that Mr. Sieber propose a course of action.

2 **Draft SRP Chapter 13.0, Section 13.1.2-13.1.3 and Supporting Draft NUREG** (Open)

Member: Stephen Rosen **Engineer:** Ralph Caruso

Estimated Time: 1 Hour

Purpose: Determine a Course of Action

Priority: High

Requested by: NRR

The NRR staff has forwarded to the Committee a revision to Chapter 13 of the SRP, and a draft final NUREG, "Regulatory Guidance for Assessing Exemption Requests From Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)," and has requested the Committee to inform them whether it wishes to discuss these documents before they are issued for public comments. The revisions to the SRP are simple, and only make reference to the draft NUREG, which provides detailed guidance to the staff in assessing requests for exemptions to 50.54(m). Copies of the request and the supporting documents have been provided to the members.

The Subcommittee recommends that Mr. Rosen propose a course of action.

April 16, 2004

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

FROM: William D. Travers ***/RA Carl Paperiello Acting For/***
Executive Director for Operations

SUBJECT: RESPONSE TO ANONYMOUS MESSAGE CONCERNING THE TRACE COMPUTER CODE DEVELOPMENT AND REVIEW PRACTICES

This is a response to your memorandum dated March 8, 2004, concerning an anonymous e-mail received by Advisory Committee on Reactor Safeguards (ACRS) member Dr. Graham Wallis. The anonymous e-mail contained criticisms of the NRC code development process and the numerical solution method used in the TRACE computer code. We will address each issue below.

(1) Independent verification of the coding

The author of the e-mail expresses concern that the NRC thermal-hydraulic system analysis codes are not required to meet the same quality assurance standards for verification and validation as commercial codes used in licensing calculations.

Quality assurance for the TRACE project follows the guidelines described in NUREG-1737, "Software Quality Assurance Procedures for NRC Thermal-Hydraulics Codes." The verification process for TRACE development is for a developer (other than the developer who did the update) to perform a review of all updates to the code as part of the agency's software QA process. Each code update and the testing of each update are independently reviewed. All code updates also have associated documentation and testing for traceability and all updates and code versions are archived. Great effort is taken to ensure that no errors in coding are introduced into TRACE, but all large software projects (both NRC and industry) contain errors. The TRACE development project has a system for reporting and correcting code errors that are found by code developers and users. The TRACE source code and documentation are also made available for scrutiny by members of the TRACE user community.

The NRC has completed a comprehensive consolidation of the features of several codes into the TRACE code. TRACE has had only a preliminary assessment to demonstrate that it is equivalent to the codes it was designed to replace. The Office of Nuclear Regulatory Research (RES) is currently in the process of developing a detailed code assessment plan for TRACE. This is an enormous effort to put together input models for many different integral and separate effects tests, to identify the key parameters for those tests, and to generate the plots to compare code predictions with data. Equally important is the development of an "auto-validation" tool that greatly enhances our ability to repeat the entire assessment matrix with each major code release. One of the capabilities of this tool is the identification of figures-of-merit which allows quantification of code fidelity. TRACE will have a comprehensive code fidelity. TRACE will have a comprehensive code assessment matrix and RES will repeat this matrix with every major code release and will perform part of the matrix with every code version.

(2) Numerical methods used in TRACE

The author expresses concern about the lack of theoretical rigor used to describe the Stability Enhancing Two Step (SETS) solution method used in TRACE and its historical development. The author asserts that the SETS method does not satisfy the original finite difference equations (FDEs) and the TRACE solution does not satisfy the nonlinear equation-of-state (EOS).

The SETS numerical method used in TRACE has been used for more than 20 years and has undergone extensive comparisons to analytical solutions, other numerical solutions, and experimental data. No specific problems unique to the SETS numerical method used in TRACE have been identified. A detailed response by RES to the concerns about the SETS is provided in Attachment 1.

RES welcomes open discussions of its analysis codes and methods. Discussion of scientific theories, methods, and results should always be open, vigorous, and subject to rational success criteria. The ACRS Thermal-Hydraulic Phenomena Subcommittee has undertaken a review of TRACE to conduct these types of discussions. The initial meeting was held November 19-20, 2003, and future meetings are planned. The concerns expressed in the anonymous e-mail can be discussed during these meetings about TRACE.

Attachment: As stated

112 State Street
Drawer 20
Montpelier, VT 05620-2701
Tel.: (802) 828-2358

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State of Vermont
Public Service Board

March 15, 2004

Mr. Nils J. Diaz, Chairman
United States Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: Vermont Public Service Board Request for Independent Engineering Assessment of Vermont Yankee Nuclear Power Station License No. DPR -28 (Docket 50-271) Technical Specification Proposed Change No. 263 Extended Power Uprate

Dear Chairman Diaz:

As you know, Entergy Nuclear Operations, Inc. ("Entergy"), is seeking approvals from both the United States Nuclear Regulatory Commission ("NRC") and the Vermont Public Service Board ("Board") in regard to a proposed 20 percent power uprate at the Vermont Yankee Nuclear Power Station ("Vermont Yankee"). We noted in your February 20, 2004, letter to Michael Kansler, President of Entergy, that your staff has determined that Vermont Yankee's extended power uprate ("uprate") application is now acceptable for review, and that your review is expected to be completed over the next 12 months.[1]

Entergy has also submitted a request to the Board for a Certificate of Public Good permitting Vermont Yankee to increase electrical generation by up to 20 percent. In determining whether Entergy should receive a Certificate of Public Good, the Board must consider several statutory criteria, including economic impacts upon the people of Vermont.

Because of this statutory standard, assessing the reliability effects of the proposed uprate upon Vermont Yankee's expected output is critical to our review. Very few nuclear plants (and even fewer of Vermont Yankee's age) have seen uprates in the 17-20 percent range. Among those

1. Letter to Michael Kansler, President Entergy Nuclear Operations, Inc. (TAC No. MC0761).

few, reductions in output have been more than incidental. From Vermont's perspective, the proposed uprate raises serious engineering questions that only the NRC appears qualified to independently assess. Thus, we are writing to ask the NRC to augment its scheduled review of Vermont Yankee along the lines set out below.

During our investigation of Entergy's request, we heard testimony as to the need for an independent review of the proposed extended power uprate. We also heard testimony from Entergy, State officials, and advocates describing the NRC's review process, and the role of the Advisory Committee on Reactor Safety (ACRS). Testimony identified the ACRS as independent of the NRC staff who conduct the initial review of the technical aspects of the proposed changes, and the importance of an independent review of its staff's findings and conclusions.

We understand that, under certain circumstances, the NRC has agreed to sponsor a more detailed review of certain engineering aspects of a nuclear plant's operation in order to establish the effectiveness of regulatory oversight. In 1996, for example, the NRC conducted such a review at the Maine Yankee Atomic Power Station ("Maine Yankee"), where there were concerns about the analysis supporting an increase in the rated thermal power at which Maine Yankee could operate. We understand that the review undertaken at Maine Yankee was performed by a "team comprised of staff who were independent of any recent or significant regulatory oversight responsibility"[2] for Maine Yankee, and that it was coordinated with the State to facilitate participation by the State representatives consistent with NRC policy. We also recognize and greatly appreciate that the Commission has subsequently incorporated into its current uprate review process much of what was developed during the 1996 Maine Yankee assessment.

We ask that, as the NRC conducts its current uprate analysis of Vermont Yankee, it do so in a way that will provide Vermont with a level of assurance about reliability equivalent to an independent engineering assessment. Such an assessment contains the following features:

- It would be independent in the same sense as the independent safety assessment of Maine Yankee, *i.e.*, it should be performed by experts "independent of any recent or significant regulatory oversight responsibility" related to Vermont Yankee.[3]
- The assessment would be a vertical slice review of two safety-related systems and two Maintenance Rule, non-safety systems affected by the uprate. The level of effort necessary for this work has been described to us in testimony as requiring about four experts for about four weeks.[4] This will provide a valuable check of the reliability of the systems that are reviewed and allow for correction of any problems.
- The independent engineering assessment should (as we believe is expected) be reviewed by the ACRS in the context of their evaluation of the power uprate.

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2. Independent Safety Assessment of Maine Yankee Atomic Power Company, U.S. Nuclear Regulatory Commission, October 1996; Vermont Public Service Board Docket No. 6812, exh. NEC-DL-3 at 1.
3. *Id.*
4. Lochbaum pf. 12/18/03 at 8–9; tr. 1/13/04 at 110–111 (Lochbaum).

We are making this unusual request of the NRC because Vermont must be reasonably assured that Vermont Yankee — a resource for which two of the state's largest retail electricity providers have contracted nearly one third of their power for the next nine years — continues to be a reliable source of electricity. While the reliability of Vermont Yankee has always been of great concern to the Board, it is especially important in the case of this proposed 20 percent extended power uprate. Thus, we request this review, as set out above, because the record presented in our proceeding strongly suggests that an uprate of the magnitude proposed here raises significant reliability issues upon which the NRC's assessment will be of extraordinarily high value.

Thank you very much for your consideration of this matter. We would welcome a response at your earliest convenience.

Sincerely,

Michael H Dworkin

Michael H. Dworkin

for
Vermont Public Service Board

Michael H. Dworkin, Chairman
David C. Coen, Board Member
John D. Burke, Board Member

Cc: Mr. Ledyard B. Marsh, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O-8E1A
Washington, D.C. 20555-0001

Mr. Richard D. Ennis, Project Manager
Licensing Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
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State of Vermont
Public Service Board

March 31, 2004

Mr. Nils J. Diaz, Chairman
United States Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: Vermont Public Service Board Request for
Independent Engineering Assessment of
Vermont Yankee Nuclear Power Station
License No. DPR -28 (Docket 50-271)
Technical Specification Proposed Change No. 263
Extended Power Uprate

Dear Chairman Diaz:

We wrote to you on March 15, 2004, requesting that the United States Nuclear Regulatory Commission ("NRC") conduct its review of the proposed extended power uprate at the Vermont Yankee Nuclear Power Station ("Vermont Yankee") in a "way that will provide Vermont with a level of assurance about reliability equivalent to an independent engineering assessment." We asked for this assessment because of our significant concerns with the effect that the uprate may have upon the future reliability of Vermont Yankee.

Today, the owner of Vermont Yankee, Entergy Nuclear Vermont Yankee ("Entergy"), submitted a filing with the Vermont Public Service Board ("Board") that included a letter from the NRC to Vermont Senator James M. Jeffords. That letter, from William D. Travers, Executive Director for Operations, suggested that the NRC was planning to conduct a baseline inspection program for the power uprate rather than expanding the review. It is unclear whether that letter to Senator

Jeffords was intended to be the NRC's response to this Board. We have also received notice that the NRC will hold a meeting tonight in Vernon to discuss the power uprate with members of the public.

At the present time, the Board has pending motions to reconsider our Order approving the proposed power uprate. As a result, we cannot actively debate the issues raised in our Order. However, we want to make very clear that the views expressed in our previous letter are unchanged, although we have not yet considered the pending motions for reconsideration (one of which seeks a more extensive independent assessment). In particular, we reiterate our request that the NRC's review of the proposed power uprate include the following features:

- It would be independent in the same sense as the independent safety assessment of Maine Yankee, *i.e.*, it should be performed by experts "independent of any recent or significant regulatory oversight responsibility" related to Vermont Yankee.

- The assessment would be a vertical slice review of two safety-related systems and two Maintenance Rule, non-safety systems affected by the uprate. The level of effort necessary for this work has been described to us in testimony as requiring about four experts for about four weeks. This will provide a valuable check of the reliability of the systems that are reviewed and allow for correction of any problems.

- The independent engineering assessment should be (as we believe is expected) reviewed by the ACRS in the context of their evaluation of the power uprate.

We want to stress that our request is not based upon a concern about the safety of Vermont Yankee; safety is clearly an issue over which the NRC has jurisdiction and considerable expertise. Instead, our concern stems from the potential impact that the power uprate could have upon reliability, which would affect the value to Vermont of existing purchase agreements for power from Vermont Yankee. A number of nuclear plants that have undergone extended power uprates have experienced increased outages or power derates. The problems that led to these outages may not have been safety-related, but they have affected the output of these nuclear plants. Our request is based upon our obligation to ensure that such outages are unlikely at Vermont Yankee.

Because of factors that are unique to Vermont Yankee, we also do not expect that granting our request will establish poor precedent. As we said in our previous letter, the record evidence we eard shows that the proposed uprate at Vermont Yankee is larger than those that have occurred at other nuclear plants. Moreover, Vermont Yankee is one of the older nuclear facilities.

Thank you very much for your consideration of this matter.

Sincerely,

Michael H. Dworkin, Chairman

David C. Coen, Board Member

John D. Burke, Board Member

Cc: Mr. Ledyard B. Marsh, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
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Mr. Richard B. Ennis, Project Manager
Licensing Project Directorate I
Division of Licensing Project Management
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CHAIRMAN

May 4, 2004

Mr. Michael H. Dworkin, Chairman
Vermont Public Service Board
112 State Street, Drawer 20
Montpelier, Vermont 05620-2701

Dear Mr. Dworkin:

I am responding on behalf of the U.S. Nuclear Regulatory Commission (NRC) to your letters dated March 15 and 31, 2004, regarding the request by Entergy Nuclear Vermont Yankee, LLC, and Entergy Nuclear Operations, Inc. (Entergy), to amend the Vermont Yankee Nuclear Power Station license to increase the power level of the facility. In those letters, the Vermont Public Service Board requested that the NRC conduct its review of the proposed power uprate in a way that would provide Vermont a level of assurance about plant reliability equivalent to an independent engineering assessment. The NRC has decided to conduct a detailed engineering inspection that we believe will be appropriate for addressing our oversight responsibilities and is also responsive to the Board's concerns. This inspection will be performed as part of a new engineering inspection program that the NRC has been developing to enhance the Reactor Oversight Process.

NRC regulations and its oversight process focus on ensuring nuclear safety, whether the facility is operating at power or shut down. The NRC's statutory authority does not extend to regulating the reliability of electrical generation. The NRC recognizes, however, that there is some overlap between attributes that result in safe operation and those that contribute to overall plant reliability.

The Commission understands that the Board is concerned about the reliability of Vermont Yankee following an increase in power level, especially in light of operational issues that have occurred at some other plants that have recently implemented extended power uprates. The NRC recognizes the importance of these issues and is taking steps to ensure that they are satisfactorily addressed to maintain safety. For example, in response to instances of steam dryer cracking at some boiling water reactors, outside technical experts are assisting NRC staff in performing an audit of General Electric's analyses related to steam dryer performance and specific issues related to Vermont Yankee. We continue to engage the industry to ensure resolution of these issues and will consider additional regulatory action, if needed.

The NRC's established review process for power uprate applications is independent, thorough, and comprehensive. A description of the review process is enclosed. Engineering assessments have always been an integral part of the NRC's safety activities. Under our current Reactor Oversight Process, NRC resident inspectors and regional specialists routinely evaluate the work performed by the licensee's engineering organization to determine whether engineering analyses adequately support safe operation. Over the past several months, the NRC has been developing a new engineering inspection program which we intend to pilot at selected plants. The NRC staff considered a number of factors, including the Board's request for an independent engineering assessment, and concluded it is appropriate to conduct this engineering inspection at Vermont Yankee. This new engineering assessment inspection incorporates the best practices of the existing and past engineering inspections. The NRC will use this inspection to verify that design bases have been correctly implemented for a sampling of components across multiple systems and to identify latent design issues. The inspection process uses operating experience, risk assessment, and engineering analysis to select risk-significant components and operator actions, and will ensure that adequate safety margins exist. Although the specific sampling of components is still being developed, it will include components from multiple systems that are potentially affected by a power uprate such as the emergency core cooling systems, the containment system, power conversion systems, and auxiliary systems. The inspection will be performed by a team of approximately six inspectors, including some NRC inspectors who do not have recent oversight experience with Vermont Yankee and at least two contractors with design experience. Three weeks of on-site inspection and over 700 hours of direct inspection time will be conducted. This level of effort exceeds that of the biennial safety system design inspection. The Commission believes it is appropriate for addressing the NRC's oversight responsibilities and is also responsive to the Board's concerns. The NRC staff will inform the State of Vermont of the schedule for this inspection to facilitate participation by State representatives, consistent with NRC policy.

The NRC Advisory Committee on Reactor Safeguards (ACRS) will also review the Vermont Yankee power uprate request. The ACRS is a statutory committee that reports directly to the Commission and is structured to provide a forum where experts representing many technical perspectives can provide advice that is factored into the NRC's decision-making process. The NRC staff will provide the results of its review efforts, including relevant inspection findings, to the ACRS for review. After the ACRS completes its review, it will make an independent recommendation regarding whether the proposed power uprate amendment should be approved.

The NRC will not approve the Vermont Yankee uprate, or any proposed change to a plant license, unless the NRC staff can conclude that the proposed change will be executed in a manner that assures the public's health and safety. In response to your request, the NRC staff has taken a close look at proposed inspections and technical reviews to ensure that they will identify and address potential safety concerns for operating at uprated power conditions. The staff has concluded that the detailed technical review, prescribed in the Extended Power Uprate Review Standard, coupled with the normal associated program of power uprate and engineering inspections, will provide the information necessary for the NRC staff to make a decision on the safety of operation of Vermont Yankee under uprated power conditions. The Commission believes that the results of NRC reviews and inspections, particularly the new engineering inspection, will assist in addressing the Board's concerns regarding the future reliability of Vermont Yankee. The NRC staff is prepared to meet with the Board to explain further our review process and scope, including the engineering assessment inspection.

Sincerely,

Nils J. Diaz

Enclosure:
Established NRC Power Uprate Review Process

Established NRC Power Uprate Review Process

The NRC's established review process for power uprate applications is independent, thorough, and comprehensive. A team of engineers with specialties in a minimum of 17 different technical areas will review the Vermont Yankee power uprate application. The NRC plans to expend about 4000 hours to perform a comprehensive assessment of the engineering, design, and safety analyses related to the uprate. The NRC's "Review Standard for Extended Power Uprates" guides the staff in its review of the application. The Review Standard also provides guidance for determining when and what type of audits should be performed at the plant or vendor sites, as well as for performing our own confirmatory analyses and independent calculations to supplement the review.

The NRC's review of the power uprate application also includes on-site inspections. NRC inspections will review selected activities and modifications made to allow operation at higher power levels to verify that changes to plant systems will support safe plant operation and are in accordance with Vermont Yankee's licensing and design bases. The NRC will use Inspection Procedure 71004, "Power Uprates," as well as a number of our baseline inspection procedures to inspect issues specifically related to power uprate. These inspections will assess changes that could impact the integrity of barriers (e.g., higher flow rates which could increase vibration at specific support points), safety evaluations, plant modifications, post maintenance and surveillance testing, heat exchanger performance, and integrated plant operation. Additionally, our other baseline inspection activities, while not specifically directed at power uprate activities, will provide additional information about Vermont Yankee's ability to operate safely at a higher power level.

The NRC will adjust, as necessary, our technical review, audit plans, confirmatory analyses, or inspection activities if any issues are identified which may have a bearing on our decision on the Vermont Yankee power uprate application. For example, a recent examination of the steam dryer at Vermont Yankee identified cracks on both interior and exterior structures of the steam dryer. The steam dryer is an important component in the process for converting steam to electrical energy, but is not used to mitigate any accidents. The NRC is interested in steam dryer cracking because of the potential for parts to break loose and impact the performance of safety-related equipment. Entergy has indicated that the cracks are in low-stress, low-steam flow areas of the dryer and not in the areas where cracks were observed at other plants that implemented extended power uprates. NRC inspectors monitored Entergy's steam dryer inspection activities, and we will thoroughly review Entergy's follow-up actions as part of our evaluation of Vermont Yankee's request to operate at a higher power level.

Assessment of engineering has always been an integral part of the NRC's safety mission. In the 1990s, the NRC performed extensive reviews at plants across the country to determine if licensees were operating plants in accordance with their design bases. As part of this review, two team inspections were conducted at Vermont Yankee in 1997. One of these inspections was led by staff from NRC headquarters and included six contractors. In 1998, the NRC conducted an engineering inspection, as well as a team inspection to address operability issues resulting from Vermont Yankee's configuration improvement program. Under our current Reactor Oversight Process, NRC resident inspectors and regional specialists routinely evaluate the work performed by the licensee's engineering organization to determine whether the engineering analyses adequately supports safe operation. Our inspectors conduct both routine engineering inspections, as well as an in-depth team inspection every two years. Since the Reactor Oversight Process was implemented in 2000, the NRC has conducted two such safety system design team inspections.

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

May 20, 2004

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

FROM: Carl J. Paperiello, Director
Office of Nuclear Regulatory Research

Carl J. Paperiello

SUBJECT: ACRS ASSESSMENT OF THE QUALITY OF RESEARCH PROGRAMS

Your letter dated April 26, 2004, stated that the Advisory Committee on Reactor Safeguards (ACRS or the Committee) agreed to assist the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research (RES), in assessing the quality and effectiveness of the NRC's research programs. As you know, we have undertaken an initiative to develop a quality metric for our research as part of our overall program to measure and continually improve the performance of the RES staff. We appreciate the Committee's willingness to support our initiative.

I understand that ACRS has decided to initially review two specific projects to enable the Committee to further develop and refine its review process. Specifically, those projects are sump blockage research and computer program improvements to the MELCOR Accident Consequence Code System (MACCS). As you are no doubt aware, those specific projects represent important aspects of our overall research program, and I believe ACRS has chosen wisely.

The cognizant RES staff have already engaged with the ACRS staff and are providing the relevant documentation for the Committee's review. Our management lead for this initiative is Mr. Michael E. Mayfield, Director of the RES Division of Engineering Technology. Please do not hesitate to contact Mr. Mayfield or me regarding this important activity.

From: "Stephen L. Rosen" <historyart@computron.net>
To: <JDSIEBER@aol.com>, <jtl@nrc.gov>
Date: 5/23/04 4:50PM
Subject: Re: electronic data transfer.

The problem I have with Jack's suggestion to transmit everything to us electronically and skip the CDs that Ethel is making is that I have a slow dial-up connection to the internet and I would never get the material. While this may be a personal problem that I have, Jack's suggestion would not work for me.

Steve

----- Original Message -----
From: JDSIEBER@aol.com
To: jtl@nrc.gov
Cc: mvbonaca@snet.net ; apostola@mit.edu ; GMLeitch@aol.com ; graham.b.wallis.@dartmouth.edu ; ransom@ecn.purdue.edu ; dapower@sandia.gov ; wjshack@anl.gov ; historyart@computron.net ; FPCTFord@aol.com ; sxd1@nrc.gov
Sent: Friday, May 21, 2004 12:17 PM
Subject: electronic data transfer.

John,

I have a suggestion for you to consider. You have adopted our suggestion to eliminate the paper notebooks that we receive at our regularly scheduled meetings and substituted CDs (prepared by Ethel) which are sent to us in the mail (usually by FedEx overnight delivery). For me, that is a good move forward as to reducing paper that I must deal with at my home and records that I must carry back and forth to DC for meetings. Electronic information also helps me to set up electronic files which are much simpler to set up than paper files. Also, the use of electronic methods of getting information to us is faster and cheaper. To me, this is a win-win for everybody who has the capability of using electronic data transmittal. The downside is that Members need to have reasonably modern computers, and perhaps a laptop (if they have a senior memory, as do I) to carry around to help them remember things. I suspect that our members are adequately equipped with computers, so that should not be a challenge.

I have another suggestion that might help streamline the process better. I note that Ethel is assigned the task of gathering all of the material to be inserted into the Notebooks and she makes a CD of that information and sends the CD to us FedEx. I also note that the staff engineers send the notebook information to Ethel (and sometimes to us) electronically (by E-mail). Why not ask the staff engineers to send the notebook material directly to us by E-mail and skip the process of having Ethel make the CDs and mail them to us. That would work just as good for me.

I buy flash memory sticks (USB compatible) and I keep my ACRS files on a few sticks. We could download the notebook material to a flash memory stick and that would be faster, better and cheaper than having Ethel make the CDs for us and mail them. (Flash memory is not cheap, I pay $46 for 256Mb of USB flash memory at Costco, in Cranberry, PA.) Between the NRC web site and the Emails we regularly receive, we certainly have access to all the information that we would otherwise get in paper form.

I suggest that you think about my suggestion and perhaps consult with the P and P Subcommittee as to whether the Members would agree to this change in the transmittal of documents to us.

Thanks,

Jack Sieber

CC: <mvbonaca@snet.net>, <apostola@mit.edu>, <GMLeitch@aol.com>, <graham.b.wallis.@dartmouth.edu>, <ransom@ecn.purdue.edu>, <dapower@sandia.gov>,

Janet/Patty Disk:Travel.Frm
9/9/94

ACRS SPECIAL TRAVEL ENDORSEMENT FORM

THIS FORM IS TO BE USED TO REQUEST ACRS ENDORSEMENT OF SPECIAL TRAVEL REQUESTS BY MEMBERS WHEN NRC SUPPORT FOR PARTIAL OR FULL REIMBURSEMENT OF EXPENSES AND/OR TIME IS DESIRED. THIS PROCEDURE IN NO WAY LIMITS THE FREEDOM OF A MEMBER TO PARTICIPATE IN A MEETING AS AN INDIVIDUAL AT PERSONAL EXPENSE. PLEASE SUBMIT THIS FORM TO THE PLANNING AND PROCEDURES SUBCOMMITTEE AT LEAST 60 DAYS PRIOR TO THE MEETING, IF POSSIBLE. SUPPLEMENTAL INFORMATION MAY BE ADDED AS DETAILS DEVELOP.

Member Name: S Rosen Date Submitted: June 2, 2004

Dates of Planned Trip: 9/18/04 to 9/23/04

Destination: Key Biscayne, Florida

Meeting or Facility to be Visited: NEI Fire Protection Info Forum

Purpose/Relevance to ACRS Business: Rosen is Chairman of ACRS Fire Protection subcommittee. Issues likely to be discussed are important to ACRS including: a) post fire safe shutdown circuit analysis; b) revised fire protection SRP; c) Operator manual actions rulemaking

Participation (Invited Speaker, paper presented, etc.):

Justification (Foreign Travel Only):

NRC SUPPORT REQUESTED

Air Fare: Yes X No Per Diem: Yes X No ~~3~~ Days 6

Registration: $ 40 Compensation: Yes X No Days ~~6~~ 3

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