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

May 26, 2005

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

Dear Chairman Diaz:

SUBJECT: SUMMARY REPORT - 522nd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, May 5-6, 2005 AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 522nd meeting, May 5-6, 2005, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following report, letter, and memorandum:

REPORT:

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

- Report on the Safety Aspects of the License Renewal Application for Arkansas Nuclear One, Unit 2 , dated May 13, 2005.

LETTER:

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

- Guidance for Assessing Exemption Requests from Nuclear Power Plant Licensed Operator Staffing Requirements, dated May 13, 2005.

MEMORANDUM:

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

- Draft Regulatory Guide DG-8029 (Proposed Revision 2 of Regulatory Guide 8.7), "Instructions for Recording and Reporting Occupational Radiation Dose Data," dated May 6, 2005.

HIGHLIGHTS OF KEY ISSUES

1. Final Review of the License Renewal Application for Arkansas Nuclear One, Unit 2

The Committee met with NRC staff and representatives of Entergy Operations, Inc. to review the license renewal application for Arkansas Nuclear One, Unit 2 (ANO-2) and the associated final Safety Evaluation Report (SER). The applicant requested approval for continued operation of this unit for 20 years beyond the current license expiration date of July 17, 2018. The applicant discussed the operating experience, major equipment replacement, and specific actions that have been or will be taken to manage the effects of aging on structures, systems, and components that are within the scope of license renewal. The staff presented the results of its review of the license renewal application and the audits and inspections conducted at the site. In the final SER, the staff concluded that the applicant has satisfied the requirements of 10 CFR 54.29(a).

Committee Action

The Committee issued a report to the NRC Chairman, dated May 13, 2005, concluding that the programs established and committed to by the applicant provide reasonable assurance that ANO-2 can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public. The Committee recommended that the application for renewal of the operating license for ANO-2 be approved.

Draft Final Revisions to Standard Review Plan (SRP), Chapter 13, "Conduct of Operations"

The Committee met with NRC staff to discuss the draft final revisions to SRP Sections 13.1.2 and 13.1.3, "Operating Organization," and the associated supporting document, NUREG-1791, "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operating Staffing Requirements Specified in 10 CFR 50.54(m)." The current regulation prescribes the licensed operator staffing required for the current generation of light water reactors. These requirements may not be appropriate for advanced reactors and operating plants with significant modifications to their control rooms; thus, applicants may wish to seek exemptions from this requirement. The changes made to Chapter 13 of the SRP reference NUREG-1791, which contains guidance for assessing requests for exemption from 10 CFR 50.54(m).

Committee Action

The Committee issued a letter to the NRC Executive Director for Operations, dated May 13, 2005, recommending that the revised SRP Sections 13.1.2 and 13.1.3 be issued. The Committee also recommended that Sections 10.1.3.2 and 10.3.3 of NUREG-1791 be revised to emphasize the importance of objective measures to evaluate the safety implications of staffing schemes, and to explore the development of objective criteria for using simulation data in the evaluation.

Advanced Reactors for Hydrogen Production

The Committee met with representatives from the Department of Energy's (DOE's) Office of Nuclear Energy, Science & Technology to discuss DOE's plans for hydrogen production using the next generation nuclear plant. Presentations by DOE included organizational structure and programs with emphasis on research and development (R&D) activities associated with the hydrogen production initiative. Because of the need for high temperatures to produce hydrogen in various processes, DOE's R&D activities have centered around Generation IV Very High Temperature Reactor (VHTR) design concepts. Utilizing a VHTR concept, two general options for hydrogen production were presented:(1) thermochemical cycle, and (2) high-temperature electrolysis. Technical challenges for each of the options were discussed.

The Committee focused on the primary issue of concern, which involves the coupling between the hydrogen production facility and the nuclear reactor. The distance between the facilities is expected to be a key licensing issue. Future studies by DOE will be directed at evaluating the separation distances and engineering features required to mitigate the impact of hydrogen hazard on the nuclear side of the plant. Aside from the coupling, DOE's goal is to have the hydrogen facility regulated separately from the nuclear facility, a process that is not unlike the one currently in place today. DOE expects to demonstrate the VHTR for hydrogen production before 2020, and to have the plant commercially available by 2025.

Committee Action

This was an information briefing, and the Committee took no action. The ACRS Action Plan for FY 2005-2008 identified Advanced Reactor Designs for Hydrogen Production as a proactive initiative in order to identify, early in the process, any new technical, safety, and policy issues requiring Commission attention. As part of this effort, the Committee plans to remain informed of DOE's hydrogen production program by meeting with DOE representatives at future opportune times. Two areas continue to remain of primary interest: early identification of potential safety and policy issues for DOE advanced reactor activity to generate hydrogen from nuclear heat, and identification of long-term research issues that will require the development of a new infrastructure to support the regulatory process.

Proactive Initiative on Safety Management

The Committee discussed options for future ACRS activity on safety management as a proactive initiative. The ACRS safety management proactive initiative follows in the wake of the Commission's August 30, 2004 Staff Requirements Memorandum (SRM) (SECY-04-0111) on Safety Culture, which directed the staff to develop a process for determining the need for performing safety culture evaluations for plants in the degraded cornerstone column. Although Committee members expressed an interest in the various options, the Committee decided to wait for NRC staff's response plan prior to proceeding further. The plan is expected to be issued in spring 2005, and stakeholder interactions are expected to begin in September 2005.

Committee Action

The Committee will review the NRC staff's plans in response to the Commission SRM and subsequently determine any future course of action.

Steam Generator Tube Integrity Program

The Committee met with representatives of the NRC staff regarding the objectives, technical approach, and results of the steam generator tube integrity program being conducted by the Argonne National Laboratory. The NRC staff provided an overview of the steam generator tube integrity program, including the task to evaluate and validate models for leak/rupture behavior of degraded steam generator tubes under normal and accident conditions. This project is one of the four projects selected by the Committee for its 2005 assessment of the quality of NRC research program.

Committee Action

The Committee plans to discuss the preliminary assessment of the quality of the research project on steam generator tube integrity during the July 6-8, 2005 ACRS meeting.

Digital Instrumentation and Control Systems Research Plan

The Committee met with representatives of the NRC staff to discuss the draft NRC Digital Systems Research Plan for FY 2005-2009. The NRC staff provided an overview of the current version of the research plan to assist the Committee's upcoming review of some of the ongoing projects. The staff discussed six research areas: systems aspects of digital technology, software quality assurance, risk assessment of digital systems, security aspects of digital systems, emerging digital technology and applications, and advanced nuclear power plants digital systems. A member of the staff also made the Committee aware of alternative views of the research plan.

Committee Action

This briefing was provided for information only. The Committee plans to review details of the specific research programs in upcoming subcommittee meetings.

Operating Experience

The Committee heard a report from the Chairman of the ACRS Subcommittee on Plant Operations about plant operating experience and another report from the ACRS staff about staff activities involving shutdown risk.

Committee Action

This briefing was provided for information only. This Committee plans to follow up on new developments involving shutdown risk.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's April 28, 2005 letter of response to the March 11, 2005 ACRS report on "Pressurized Thermal Shock (PTS) Reevaluation Project: Technical Basis for Revision of the PTS Screening Criteria in the PTS Rule."

  The Committee decided that it was satisfied with the EDO's response. The Committee plans to discuss whether the revised NUREG-1809, "Thermal-Hydraulic Evaluation of Pressurized Thermal Shock," is responsive to the Committee's comments and recommendations.

- The Committee considered the EDO's April 4, 2005 response to the ACRS's letter of February 24, 2005, concerning the Committee's review of the proposed Waterford 3 Extended Power Uprate (EPU). In its letter, the Committee recommended that (1) the application by Entergy for the EPU should be approved, subject to (a) the staff's approval of the alternate source term (AST) application, and (b) documentation of the resolution of the boron precipitation issue during long-term cooling for Waterford 3 by the submittal of the analysis details and their acceptance in the staff's SE; (2) the staff should waive the requirement for large-transient testing for this application; and (3) the staff should review the generic potential for boron concentration and precipitation to interfere with core cooling following a LOCA.

  The Committee decided that it was satisfied with the EDO's response. The staff plans to pursue the issue of boron concentration modeling with the PWR vendors and will review post-LOCA boron precipitation as part of each future PWR power uprate licensing action. However, the staff believes that the safety significance of the issue is not sufficiently high as to meet the threshold for evaluation as a generic safety issue. Instead, the staff believes that the industry should address the issue as part of the long-term cooling requirements of 10 CFR 50.46.

- The Committee considered the EDO's April 22, 2005 letter of response to the March 11, 2005 ACRS report on its review of the revised draft NUREG Report, "Estimating Loss-of-Coolant Accident Frequencies Through the Elicitation Process."

  The Committee decided that it was satisfied with the EDO's response. The Committee plans to review the draft final NUREG report once the NRC staff has resolved public comments.

- The Committee considered the EDO's April 25, 2005 letter of response to the March 14, 2005 ACRS report on its review of the proposed rule for a voluntary alternative to 10 CFR 50.46, "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements."

  The Committee decided that it was satisfied with the EDO's response. The Committee plans to review the associated draft proposed Regulatory Guide and the draft final rule once the NRC staff has resolved public comments.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from April 10, 2005 through May 5, 2005, the following Subcommittee meetings were held:

- Fire Protection - May 4, 2005

  The Subcommittee met with representatives of the NRC staff and Electric Power Research Institute (EPRI) to discuss the fire risk assessment methodology described in NUREG/CR-6850, "EPRI/NRC-RES Fire Methodology for Nuclear Power Facilities." Representatives of the NRC staff and EPRI also briefed the Subcommittee on a draft NUREG, "Verification and validation of selected Fire Models for Nuclear Power Plant Application."

- Planning and Procedures - May 4, 2005

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

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The Committee would like to have a briefing by the staff on its evaluation of recent shutdown events. (ACRS POC: John Lamb)

- The Committee plans to review the draft final NUREG Report, "Estimating Loss-of-Coolant Accident Frequencies Through the Elicitation Process, " once the NRC staff has resolved public comments. (ACRS POC: Michael Snodderly)

- The Committee plans to review the associated draft proposed Regulatory Guide and the draft final rule for a voluntary alternative to 10 CFR 50.46 once the NRC staff has resolved public comments. (ACRS POC: Michael Snodderly)

- The Committee plans to discuss whether the revised NUREG-1809, "Thermal-Hydraulic Evaluation of Pressurized Thermal Shock," is responsive to the Committees comments and recommendations included in the May 11, 2005 ACRS report. (ACRS POC: Cayetano Santos)

PROPOSED SCHEDULE FOR THE 523rd ACRS MEETING

The Committee agreed to consider the following topics during the 523rd ACRS meeting to be held on June 1-3, 2005:

- Interim Review of the License Renewal Application for the Point Beach Nuclear Plant, Units 1 and 2

- Draft Commission Paper on Policy Issues Related to New Plant Licensing

- Fire Risk Requantification and Probabilistic Risk Analysis Methodology for Nuclear Power Plants
- Draft Commission Paper on Proposed Alternatives to the Existing Single Failure Criterion
- Draft Safety Evaluation Report Related to Grand Gulf Early Site Permit Application
- Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants"
- Status Reports on the Quality Assessment of Selected Research Projects

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

CERTIFIED

Date Issued: 6/15/05
Date Certified: 6/23/05

CERTIFIED

Date Issued: 6/15/05
Date Certified: 6/23/05

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May 5-6, 2005

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REPORTS:

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- Report on the Safety Aspects of the License Renewal Application for Arkansas Nuclear One, Unit 2 , dated May 13, 2005.

LETTERS:

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

- Guidance for Assessing Exemption Requests from Nuclear Power Plant Licensed Operator Staffing Requirements, dated May 13, 2005.

MEMORANDA:

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

- Draft Regulatory Guide DG-8029 (Proposed Revision 2 of Regulatory Guide 8.7), "Instructions for Recording and Reporting Occupational Radiation Dose Data," dated May 6, 2005.

APPENDICES

I. *Federal Register Notice*
II. Meeting Schedule and Outline
III. Attendees
IV. Future Agenda and Subcommittee Activities
V. List of Documents Provided to the Committee

MINUTES OF THE 522nd MEETING OF THE
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
May 5-6, 2005
ROCKVILLE, MARYLAND

The 522nd meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on May 5-6, 2005. Notice of this meeting was published in the *Federal Register* on April 20, 2005 (75 FR 20608) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting schedule and outline (Appendix II). The meeting was open to public attendance. There were no written statements or requests for time to make oral statements from members of the public regarding the meeting.

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

ATTENDEES

ACRS Members: ACRS Members: Dr. Graham B. Wallis (Chairman), Dr. William J. Shack (Vice Chairman), Mr. John D. Sieber, (Member-at-Large), Dr. George E. Apostolakis, Dr. Mario V. Bonaca, Dr. Richard S. Denning, Dr. Thomas S. Kress, Dr. Dana A. Powers, Dr. Victor H. Ransom, and Mr. Stephen L. Rosen. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

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

Dr. Graham B. Wallis, Committee Chairman, convened the meeting at 8:30 a.m. and reviewed the schedule for the meeting. He summarized the agenda topics for this meeting and discussed the administrative items for consideration by the full Committee.

II. Final Review of the License Renewal Application for Arkansas Nuclear One, Unit 2 (ANO-2) (Open)

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

The Committee met with NRC staff and representatives of Entergy Operations, Inc. to review and discuss the license renewal application for Arkansas Nuclear One, Unit 2 (ANO-2) and the associated final Safety Evaluation Report (SER). The applicant requested approval for continued operation of this unit for 20 years beyond the current license expiration date. The operating license for ANO-2 expires on July 17, 2018.

ANO-2 is a Combustion Engineering pressurized water reactor rated at 3026 MWt. This power rating includes a 7.5% power approved in 2002. The applicant has implemented several plant improvement initiatives such as replacing the steam generators, upgrading the feedwater control system, upgrading the turbine, and replacing piping affected by flow accelerated corrosion. In the future, the applicant plans to replace the pressurizer, the reactor vessel head, and service water piping. ANO-2 was the second application to be evaluated by the staff using a new audit and review process to confirm consistency with and the acceptability of deviations from the Generic Aging Lessons Learned (GALL) Report. As a result of the staff's review, several components were brought into scope of license renewal and one aging management program was added.

The draft SER issued in November 2004 contained no open items, no confirmatory items, and three proposed license conditions. In the final SER dated April 2005, the staff concluded that the applicant has satisfied the requirements of 10 CFR 54.29(a).

Committee Action

The Committee issued a report to the NRC Chairman, dated May 13, 2005, concluding that the programs established and committed to by the applicant provide reasonable assurance that ANO-2 can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public. The Committee recommended that the application for renewal of the operating license for ANO-2 be approved.

III. Draft Final Revisions to Standard Review Plan (SRP), Chapter 13, "Conduct of Operations"

[Note: Dr. Medhat El-Zeftawy was the Designated Federal Official and Mr. Eric A. Thornsbury was the cognizant staff engineer for this portion of the meeting.]

The Committee met with the NRC staff to discuss the draft final revisions to SRP Sections 13.1.2 and 13.1.3, "Operating Organization," and the associated supporting document, NUREG-1791, "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operating Staffing Requirements Specified in 10 CFR 50.54(m)." The current regulation prescribes the licensed operator staffing required for the current generation of light water reactors. These requirements may not be appropriate for advanced reactors and operating plants with significant modifications to their control rooms; thus, applicants may wish to seek exemptions from this requirement. The changes made to Chapter 13 of the SRP references NUREG-1791 and contains guidance for assessing requests for exemption from 10 CFR 50.54(m).

Committee Action

The Committee issued a letter to the NRC Executive Director for Operations, dated May 13, 2005, recommending that the revised SRP Sections 13.1.2 and 13.1.3 be issued. The Committee also recommended that Sections 10.1.3.2 and 10.3.3 of NUREG-1791 be revised to

emphasize the importance of objective measures to evaluate the safety implications of staffing schemes, and to explore the development of objective criteria for using simulation data in the evaluation.

IV. Proactive Initiative on Advanced Reactors for Hydrogen Production

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

Dr. Thomas Kress, Chairman of Future Plant Designs Subcommittee, opened by stating that the purpose of the meeting is to become familiar with the Department of Energy's (DOE's) nuclear hydrogen initiative, and to start thinking about regulatory requirements, safety implications, and the need for new research tools. Dr. Kress also noted DOE's expectations from the meeting, was to obtain initial ACRS feedback on potential regulatory and safety issues. Dr. Kress then turned the meeting over to John Gross, the Acting Director for the Office of Advanced Nuclear Research, in DOE's Office of Nuclear Energy, Science & Technology

Mr. Gross opened by describing DOE's organizational structure starting with the Office of Advanced Nuclear Research (OANR). Three major programs within OANR relate to the next generation of advanced reactor designs: (1) Advanced Fuel Cycle Initiative (AFCI), (2) Generation IV (Gen IV), (3) Nuclear Hydrogen Initiative (NHI). Because of the need for high temperatures to produce hydrogen, Mr. Gross indicated that DOE's R&D nuclear hydrogen production activities centered aroundGeneration IV Very High Temperature Reactor (VHTR) design concepts. He noted that there were two options for hydrogen production both requiring very high temperatures: (1) thermochemical cycle and (2) high temperature electrolysis. Mr. Gross then presented the time line for meeting DOE's goal and objectives:

2010 - decision to select a fast reactor technology
2010 - Yucca mountain opens
2010-2020 - deployment of first advanced light water reactor
2017-2020 VHTR demonstration
2017 - alternate fuel cycles to be brought on-line (more proliferation resistance)
2020 - decision made on second repository
2025+ Gen IV designs come on-line
2040 - conversion to closed fuel cycles

Mr. Gross then turned the presentation over to Paul Pickard (SNL), Nuclear Technology Integrator for NHI. Mr. Pickard explained that DOE is looking at a range of reactors for the next generation plant, but is primarily focusing on the very high temperature reactor (VHTR) for both its efficiency and for its use in hydrogen production. High temperature gas cooled and molten salt cooled graphite reactors were the two concepts seriously being considered.

For the hydrogen production facility, Mr. Pickard indicated that the thermochemical cycles generally require temperatures to be in the 1000C range and for the most part, are sulfur based involving the decomposition of sulfuric acid (beginning at about 800-900C). The alternative is

use high temperature electrolysis. Because high temperature electrolysis uses steam instead of water, it has a higher efficiency (about 10-15%) than the water based process.

During the discussion, Dr. Kress indicated that unless oxygen is put back into the environment, large scale operation of hydrogen production facilities could lead to depletion of oxygen in the environment. Mr. Pickard indicated that the oxygen created during the process would be used somewhere or would be dispersed, but at the time did not enter into the decision making process for the plant's design. He noted that the U.S. demand for hydrogen is about 10 million tons a year compared to world-wide demand of about 50 million tons. Most of the U.S. produced hydrogen is used for ammonia production (50%) and oil refining (37%). Additionally, as the grade of crude oil continues to decrease, the need for hydrogen will increase in order to meet the demand necessary to refine the crude oil into gasoline.

Mr. Pickard noted that the challenge for the high temperature electrolysis process is in having to maintain and operate millions of cells that are basically fuel cells running in reverse, i.e., where instead of generating a voltage from the cell, a voltage is applied to the cell. For this type of process, the economies of scale transforms the need to manufacture a massive number of cells, which increases the cost and makes high temperature management more difficult. The advantage of the fuel cell program, however, is that it allows for a wider temperature range, uses a steam process that eliminates the need for hazardous chemicals.

For the thermochemical cycle, Mr. Pickard indicated that the challenge is having to deal with a series of chemical species that are very corrosive at high temperature, and potentially poisonous to operators. Therefore, material issues will clearly be the most significant challenge for the thermochemical cycle. In either case, both processes have a similar challenge in that they will need high temperatures and an interface between a nuclear facility and a hydrogen production facility.

A number of key Issues were then identified and discussed during the meeting:

- the need to investigate accident scenarios that could result from a mix of hydrogen and oxygen production products.
- the effects of toxic gases on operating crews, specifically with respect to the potential release of a large quantity of sulfur dioxide.
- the need to understand the impact of the chemical plant (accidents) on the nuclear plant.
- alignment of the safety analysis performed by the chemical industry, with the probabilistic risk analysis performed by the nuclear industry (significant differences exist today between the two types of analyses).
- establishing an appropriate separation distance between the hydrogen production facility and the nuclear plant.

- how to inspect the central pipe of a concentric pipe system that transports high temperature helium between the hydrogen and nuclear facilities.

- the need for large helium pipes (if helium is used) between the nuclear and hydrogen facility, or the need for materials that can withstand the corrosive behavior from molten salt if molten salt is used in the intermediate loop.

- how to store large quantities of hydrogen storage during production, with a production rate that could include a couple of kilograms per second.

- Security implications and the accidents that will need to be considered.

In closing, Mr. Pickard indicated that much of the work presented had been started this year, and hopes to have a good piece of the work completed within the next year and a half, to two years. The major issue is the one involving separation of the facilities. ACRS member Dana Powers indicated that there is no graded approach to treating separation between the hydrogen and nuclear facilities, either it must be fully addressed or does not need to be considered (for example, if it is greater than 5 miles away). For the thermochemical process, the effect of tons of sulfur dioxide (not pounds) on the nuclear facility will need to be considered, a quantity that no other licensee had to deal with before. Dr. Kress then indicated that it may be better to have only one facility generate hydrogen, and another to generate electricity rather than combining the two into a single unit. This would allow operating crews to focus on one rather than two complex functions. Additionally, one may be able to use a less hazardous material on-site which could be transported off-site to generate hydrogen, effectively removing the hydrogen hazard. At this time, there is no known process.

In conclusion, the primary issue of concern involves the coupling between the hydrogen production facility and the nuclear reactor. Future studies by DOE will be directed at evaluating the separation distances and engineering features required to mitigate the impact of hydrogen or chemical hazard. Aside from the coupling, DOE's goal is to have the hydrogen facility regulated separately from the nuclear facility, a process which is not unlike the one currently in place today. DOE expects to demonstrate the VHTR for hydrogen production as a concept before 2020, and have the plant commercially available by 2025.

Committee Action

This was an information briefing with no Committee action at this time. The Committee plans to remain informed of DOE's hydrogen production program by meeting with DOE representatives in the future. Two areas continue to remain of primary interest: (1) early identification of potential safety and policy issues for DOE advanced reactor activity to generate hydrogen from nuclear heat and (2) identification of long-term research issues that will require new analytical tools or infrastructure development to support the regulatory process.

V. Proactive Initiative on Safety Management

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

Dr. John H. Flack, ACRS staff, stated that the purpose of this meeting was to discuss five options for taking action on safety management as an ACRS proactive initiative. The proactive initiative is intended to keep the Committee informed and out in front on evolving issues. The initiative is consistent with the Commission's Strategic Plan, specifically with respect to keeping abreast of new technologies and opportunities (safety strategy), enhancing NRC process and products by supporting the use of good science (effectiveness strategy), and ensuring excellence in Agency Management (management strategy). The ACRS safety management proactive initiative follows in the wake of the Commission's August 2004 SRM (SRM-04-0111) which directed the staff to develop a process for determining the need for safety culture evaluations of plants in the Reactor Oversight Program (ROP) degraded cornerstone column. The ACRS plans to review the NRC's staff's response to the Commission's SRM when it becomes available later this year. Although several Committee members expressed an interest in one or more of the proactive initiative options, the Committee believed it better to wait for the release of NRC staff's Response Plan before initiating any action. Following ACRS review of the staff's Response Plan, Committee members will decide what proactive initiatives (options) are warranted in light of NRC staff activities.

Committee Action

The Committee will review NRC's plans in response to the Commission's SRM-04-0111 and determine what additional actions are necessary. The proactive initiative is intended to compliment not duplicate the NRC staff planned activities in response to the Commission's SRM.

VI. Steam Generator Tube Integrity Program

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

The Committee met with representatives of the NRC staff regarding the objectives, technical approach, and results of the steam generator tube integrity program being conducted by the Argonne National Laboratory. The NRC staff provided an overview of the steam generator tube integrity program including the task to evaluate and validate models for leak/rupture behavior of degraded steam generator tubes under normal and accident conditions. This project is one of the four specific projects selected by the Committee for its 2005 assessment of the quality of NRC research program.

Committee Action

The Committee plans to discuss the preliminary assessment of the quality of the research project on steam generator tube integrity during the July 6-8, 2005 ACRS meeting.

VII. Digital Instrumentation and Control (I&C) Systems Research Plan

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

Dr. George Apostolakis, the cognizant Committee member for this issue, introduced this topic by describing the NRC Digital System Research Plan for fiscal years 2005-2009. Dr. Apostolakis mentioned internal discussions within the Office of Nuclear Reactor Regulation (NRR) concerning various views of the value of the proposed research. Mr. Michael Mayfield, Director of the Division of Engineering in NRR, pointed out that the memoranda that Dr. Apostolakis referred to an internal office process to collect, review, and resolve the various views within the office. Once that process is complete, NRR will provide official comments to the Office of Nuclear Regulatory Research (RES) for their consideration. As a matter of policy, Mr. Mayfield stated that NRR believes that an active research program in this area is useful, and they look forward to dialog with RES to address any office concerns. Dr. Apostolakis then asked Mr. William Kemper, RES, to begin.

NRC Staff Presentation

Mr. Kemper briefly discussed the goal of the presentation, to provide the Committee with the information required to determine what further interactions are needed. He then introduced Mr. Mike Waterman, Senior I&C Engineer, RES, to provide the bulk of the presentation.

The research plan attempts to provide a flexible, adaptable framework for identifying a research initiative for the other program offices, including NMSS and NSIR, in addition to NRR. The research plan is oriented toward providing a more consistent process for regulating nuclear application of digital technology, through the development of objective acceptance criteria. In addition to assessment tools and methods, Mr. Waterman also discussed the need for review and inspection procedures to codify the review process, and a training curriculum to teach new employees the review methods. Mr. Waterman discussed the rapid changes in digital technology and the need to keep our licensing processes current with the technology, while at the same time moving toward a more risk-informed, performance-based process. The goal is not to replace existing review methods, but to improve them.

Mr. Waterman's presentation stepped through the six areas of research proposed in the draft research plan: system aspects of digital technology, software quality assurance, risk assessment of digital systems, security aspects of digital systems, emerging digital technology and applications, and advanced nuclear power plant digital systems. Within each research area, Mr. Waterman discussed the ongoing and planned projects.

Staff from other offices contributed their opinions on the research plan from the audience. Mr. Scott Morris, Chief of the Reactor Security Section, NSIR, commented on the agency's need for a more comprehensive cyber security policy, both for our licensees and internally. Mr. Jose Calvo, Chief of the NRR Electrical and Instrumentation and Controls Branch, also

provided his views of the research plan, which may differ somewhat from the official NRR comments.

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

- Dr. Wallis commented on the value and need for objective acceptance criteria.

- Dr. Apostolakis suggested that part of the research should explore whether or not a risk-informed, performance-based approach can even be developed in the near future due to the inherent differences between digital systems and traditional hardware systems.

- Dr. Bonaca asked about the increasing level of complexity we are seeing in digital systems, and whether or not it is necessary. Mr. Waterman's view was that the complexity comes partly from the desire to increase the number of functions handled by the digital system and partly from the natural capability of digital systems to provide additional, though unnecessary, functions.

- Dr. Powers commented on the development of tools by the NRC. He commented on the philosophy of developing independent NRC tools that are adequate enough to help the staff pose questions to the licensee, as opposed to developing tools that have sufficiently high quality to design and certify systems on their own. Mr. Kemper agreed with the first approach, where he hopes that licensees will address the issues raised by the new tools, but that the NRC develops the ability to independently assess systems with their own tools.

- Dr. Powers and Dr. Apostolakis asked about the staff's involvement in activities in this area internationally and in other industries. Mr. Kemper answered that the staff has interfaces with NASA, the military, and other government agencies, as well as keeping up with international activities. Dr. Apostolakis reminded the staff of the need to examine applications from other industries from the nuclear power perspective, which tends to question more issues.

- Dr. Apostolakis commented on the use of Markov models to assess reliability through fault injection presented previously to the Committee. He stated that he does not believe these approaches have been scrutinized enough to be accepted yet, though methods such as fault injection add confidence in the reliability of digital systems even without producing a reliability value.

- Dr. Apostolakis also asked the staff if they have objective criteria for determining when enough research has been performed on any particular topic. Mr. Waterman and Mr. Kemper answered that the staff is addressing the need for research as they progress, for example by using a phased process that periodically assesses the viability of a research project.

Committee Action:

This briefing was provided for information only. The Committee plans to review the draft final Digital Systems Research Plan and some of its ongoing projects at a later Full Committee meeting.

VIII. Executive Session

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

A. Reconciliation of ACRS Comments and Recommendations/EDO Commitments

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

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

- The Committee considered the EDO's April 28, 2005 letter of response to the March 11, 2005 ACRS report on "Pressurized Thermal Shock (PTS) Reevaluation Project: Technical Basis for Revision of the PTS Screening Criteria in the PTS Rule."

  The Committee decided that it was satisfied with the EDO's response. The Committee plans to discuss whether the revised NUREG-1809, "Thermal-Hydraulic Evaluation of Pressurized Thermal Shock," is responsive to the Committee's comments and recommendations.

- The Committee considered the EDO's April 4, 2005 response to the ACRS's letter of February 24, 2005, concerning the Committee's review of the proposed Waterford 3 Extended Power Uprate (EPU). In its letter, the Committee recommended that (1) the application by Entergy for the EPU should be approved, subject to (a) the staff's approval of the alternate source term (AST) application, and (b) documentation of the resolution of the boron precipitation issue during long-term cooling for Waterford 3 by the submittal of the analysis details and their acceptance in the staff's SE; (2) the staff should waive the requirement for large-transient testing for this application; and (3) the staff should review the generic potential for boron concentration and precipitation to interfere with core cooling following a LOCA.

  The Committee decided that it was satisfied with the EDO's response. The staff plans to pursue the issue of boron concentration modeling with the PWR vendors and will review post-LOCA boron precipitation as part of each future PWR power uprate licensing action. However, the staff believes that the safety significance of the issue is not sufficiently high as to meet the threshold for evaluation as a generic safety issue. Instead, the staff believes that the industry should address the issue as part of the long-term cooling requirements of 10 CFR 50.46.

- The Committee considered the EDO's April 22, 2005 letter of response to the March 11, 2005 ACRS report on its review of the revised draft NUREG Report, "Estimating Loss-of-Coolant Accident Frequencies Through the Elicitation Process."

  The Committee decided that it was satisfied with the EDO's response. The Committee plans to review the draft final NUREG report once the NRC staff has resolved public comments.

- The Committee considered the EDO's April 25, 2005 letter of response to the March 14, 2005 ACRS report on its review of the proposed rule for a voluntary alternative to 10 CFR 50.46, "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements."

  The Committee decided that it was satisfied with the EDO's response. The Committee plans to review the associated draft proposed Regulatory Guide and the draft final rule once the NRC staff has resolved public comments.

B. Report on the Meeting of the Planning and Procedures Subcommittee

The Committee heard a report from the ACRS Chairman and the Executive Director, ACRS, regarding the Planning and Procedures Subcommittee meeting held on May 4, 2005. The following items were discussed:

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

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

Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through July 2005 were addressed. The objectives were:

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

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

Revised ACRS Subcommittee Structure

During the April ACRS meeting, a proposed revision to the ACRS Subcommittee structure was provided to the members, and comments was requested by April 22, 2005. The current version reflects incorporation of the comments received. The revised Subcommittee structure became effective on May 9, 2005.

Staff Requirements Memorandum Resulting from the ACRS/Commission Meeting

In an April 25, 2005 Staff Requirements Memorandum that resulted from the April 7, 2005 ACRS meeting with the NRC Commissioners, the Commission stated the following:

- The Committee should provide the Commission a list of research projects it intends to review in the short term as part of its assessment of research quality, with an indication of the methodology the Committee will use for the review. The Commission appreciates the Committee's undertaking of this effort in addition to providing a biennial report to the Commission on the NRC Safety Research Program.

- The Committee should review the staff's Regulatory Guide that will implement the revised 10 CFR 50.46.

- The Committee should consider reviewing upcoming revisions to NUREG-0800 (Standard Review Plan) Sections that involve significant changes.

Proposed Plan for Preparing the 2006 ACRS Report to the Commission on the NRC Safety Research Program

The 2006 ACRS report on the NRC Safety Research Program was due to the Commission on March 15, 2006. The Committee established the format, schedule, and assignments for this report.

Attendance at Foreign Meetings

Each year some members attend technical conferences held in foreign countries. In order to ensure that adequate resources are available to support attendance at such meetings, the ACRS Executive Director would like to know the members' plans to attend such conferences. Since all foreign travels are approved by the NRC Chairman or his designee, members who plan to attend foreign technical conferences should fill out all necessary forms to be sent to the NRC Chairman.

Browns Ferry Plant Visit

During the April 2005 ACRS meeting, the Committee decided to visit the Browns Ferry Plant on Tuesday, August 23, 2005, and hold a meeting with the Regional Office on August 24 (afternoon) and 25 (morning). It is anticipated that all members will participate. Those members who do not plan to visit the Browns Ferry plant and participate in the meeting with the Regional Office should inform Mr. Caruso. Details of the arrangements for the plant visit and agenda for the meeting with the Regional Office will be provided to the members during the June ACRS meeting.

Proposed Strategy for ACRS Review of the License Renewal Applications

During the April 6, 2005 Planning and Procedures Subcommittee meeting and at the April 2005 full Committee meeting, Dr. Bonaca, Chairman of the Plant License Renewal Subcommittee, discussed the anticipated workload in the license renewal area. Subsequently, Dr. Powers sent an email to all the members that proposed a strategy for ACRS review of the license renewal applications. He stated that Dr. Bonaca has done an outstanding job in the license renewal area. However, Dr. Bonaca cannot bear up under the anticipated load of license renewal applications. Furthermore, if Dr. Bonaca took the lead on all license renewal applications, the Committee would lose his wise counsel on other issues that the ACRS must address. Dr. Powers proposed the following strategy for dealing with license renewal applications:

- The Committee should divide the responsibility for reviewing the forthcoming license renewal applications among the members — one application to each member.

- Dr. Bonaca could assist the members or take lead responsibility for reviewing particularly troublesome applications.

Other members agreed with the proposal by Dr. Powers. Dr. Apostolakis suggested that the Committee fill the existing vacancy with a member who has plant operating experience.

In an email dated April 27, 2005, Dr. Bonaca suggested the following:

- The Committee should try to bring in a new member, with operating experience, on board prior to 2006. If that happens, the workload in the license renewal area could be split between the new member and Dr. Bonaca.

- If a new member with operating experience is not on board by 2006, Dr. Bonaca would take review responsibility to every other license renewal application. Review of other applications should be assigned to other members.

Quadripartite Meeting Status

In March 2005, Dr. Larkins, on behalf of the ACRS, sent an email to the respective chairpersons at GPR (France), NSC (Japan) and RSK (Germany) to start the planning process for the 2006 Quadripartite Meeting. The email contained a copy of the proposed topics for the meeting, dates for the meeting (October 18-20, 2006), and questions about 1) extending the invitation to Swiss/Sweden as well as to other Countries such as Korea and China; and 2) conference location and other logistical issues. All three Countries have indicated that they have received this email and are preparing a formal response. NSC is amenable to extending the invitation to other Countries, while RSK and GPR prefer to limit invitations to Countries with advisory committees equivalent to those in the member Countries. Comments received from GPR, NSC, and RSK were received.

We have contacted the Department of Energy and have received approval for the group to tour Yucca Mountain. Meanwhile, Link Technologies, Inc. explored the availability and costs associated with potential venues in the Washington, DC or Las Vegas area for the conference site.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 522nd ACRS Meeting, May 5-6, 2005.

The 522nd ACRS meeting was adjourned at 6:00 p.m. on May 6, 2005.

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

June 15, 2005

MEMORANDUM TO: ACRS Members

FROM: Noble S. Green, Jr.
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 522nd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - MAY 5-6, 2005

Enclosed are the proposed minutes of the 522nd meeting of the ACRS. This draft is being provided to give you an opportunity to review the record of this meeting and provide comments. Your comments will be incorporated into the final certified set of minutes as appropriate, which will be distributed within six (6) working days from the date of this memorandum.

Attachment:
As stated

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

June 23, 2005

MEMORANDUM TO: Noble S. Green, Jr., Technical Secretary
Advisory Committee on Reactor Safeguards

FROM: Graham B. Wallis
ACRS Chairman

SUBJECT: CERTIFIED MINUTES OF THE 522nd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), May 5-7, 2005

I certify that based on my review of the minutes from the 522nd 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.

dated November 29, 2004. Supporting documentation is available for inspection at the NRC's Public Electronic Reading Room at *http://www.nrc.gov/reading-rm/adams.html.* A copy of the Finding of No Significant Impact can be found at this site using the Agencywide Documents Access and Management System (ADAMS). These documents may also be viewed electronically on the public computers located at the NRC's Public Document Room (PDR), O1–F21, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852. The PDR reproduction contractor will copy documents for a fee. 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 at 1–800–397–4209 or (301) 415–4737, or by e-mail to *pdr@nrc.gov.*

Dated at Rockville, Maryland, this 13th day of April 2005.

For the Nuclear Regulatory Commission.

Jeremy A. Smith,

Project Manager, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards.

[FR Doc. E5–1854 Filed 4–19–05; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

The ACRS Subcommittee on Planning and Procedures will hold a meeting on May 4, 2005, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

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

The agenda for the subject meeting shall be as follows:

Wednesday, May 4, 2005–10 a.m.–11:30 a.m.

The Subcommittee will discuss proposed ACRS activities and related matters. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Sam Duraiswamy (telephone: 301–415–7364) between 7:30 a.m. and 4:15 p.m. (ET) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public.

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

Dated: April 13, 2005.

Michael L. Scott,

Branch Chief, ACRS/ACNW.

[FR Doc. E5–1851 Filed 4–19–05; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Subcommittee Meeting on Fire Protection; Notice of Meeting

The ACRS Subcommittee on Fire Protection will hold a meeting on May 4, 2005, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Wednesday, May 4 , 2005—8:30 a.m. until 3 p.m.

The purpose of this meeting is to discuss the NRC/EPRI joint work on the improved fire risk assessment methodology. The Subcommittee will discuss NUREG/CR–6850, "EPRI/NRC–RES Fire PRA Methodology for Nuclear Power Facilities." The Subcommittee will also discuss the NRC staff's efforts on verification and validation of fire models. The Subcommittee will hear presentations by and hold discussions with the NRC staff, representatives of the EPRI, and other interested persons regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Hossein P. Nourbakhsh (Telephone: 301–415–5622) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted.

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

Dated: April 13, 2005.

Michael L. Scott,

Branch Chief, ACRS/ACNW.

[FR Doc. E5–1852 Filed 4–19–05; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting Notice

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on May 5–6, 2005, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Wednesday, November 24, 2004 (69 FR 68412).

Thursday, May 5, 2005, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting. *8:35 a.m.–10 a.m.: Final Review of the License Renewal Application for Arkansas Nuclear One, Unit 2 (ANO–2)* (Open)—The Committee will hear presentations by and hold discussions with representatives of the Entergy Operations, Inc. and the NRC staff regarding the license renewal application for ANO–2 and the associated final Safety Evaluation Report prepared by the NRC staff.

10:15 a.m.–11:45 a.m.: Draft Final Revisions to Standard Review Plan (SRP), Chapter 13, "Conduct of Operations" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft final revisions to Sections 13.1.2–13.1.3, "Operating Organization," of SRP Chapter 13 and related NUREG–1791, "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing

Requirements Specified in 10 CFR 50.54 (m)."

12:45 p.m.–2:45 p.m.: Advanced Reactor Designs for Hydrogen Production (Open)—The Committee will hear presentations by and hold discussions with representatives of the Department of Energy (DOE) regarding the status of DOE plans and research and development activities in support of advanced reactor designs for hydrogen production.

3 p.m.–4 p.m.: Significant Recent Operating Events (Open)—The Committee will hear a briefing by the Chairman of the ACRS Subcommittee on Plant Operations regarding significant recent operating events.

4 p.m.–5 p.m.: Proactive Initiative (Open)—The Committee will discuss proposed options for addressing ACRS proactive initiative on safety management.

5:15 p.m.–6:45 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting.

Friday, May 6, 2005, Conference Room T-2B3, Two White Flint North, Rockville, Maryland

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

8:35 a.m.–10 a.m.: Steam Generator Tube Integrity Program (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the objectives, technical approach, and results of the steam generator tube integrity program being conducted by the Argonne National Laboratory.

10:15 a.m.–11:45 a.m.: Digital Instrumentation and Control (I&C) Systems Research Plan (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the digital I&C systems research plan.

11:45 a.m.–12 Noon: 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.

1 p.m.–2 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 p.m.–6:30 p.m.: Preparation of ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

6:30 p.m.–7 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 5, 2004 (69 FR 59620). In accordance with those procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting. Persons desiring to make oral statements should notify the Cognizant ACRS staff named below five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Cognizant ACRS staff prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Cognizant ACRS staff if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, as well as the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Sam Duraiswamy, Cognizant ACRS staff (301–415–7364), between 7:30 a.m. and 4:15 p.m., 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: April 14, 2005.

Annette L. Vietti-Cook,

Secretary of the Commission.

[FR Doc. E5–1853 Filed 4–19–05; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–51541; File No. SR–NSCC–2005–02]

Self-Regulatory Organizations; National Securities Clearing Corporation; Notice of Filing of Proposed Rule Change To Enhance Automated Customer Account Transfer Service To Permit the Automated Notification of Changes to the Broker-Dealer of Record for Applicable Insurance Products

April 13, 2005.

Pursuant to section 19(b)(1) of the Securities Exchange Act of 1934 ("Act"),[1] notice is hereby given that on April 4, 2005, the National Securities Clearing Corporation ("NSCC") filed with the Securities and Exchange Commission ("Commission") and on April 12, 2005, amended the proposed rule change described in Items I, II, and III below, which items have been prepared primarily by NSCC. The Commission is publishing this notice to solicit comments on the proposed rule change from interested parties.

[1] 15 U.S.C. 78s(b)(1).

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

April 12, 2005

SCHEDULE AND OUTLINE FOR DISCUSSION
522nd ACRS MEETING
MAY 5-6, 2005

THURSDAY, MAY 5, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:~~30~~ 31 - 8:35 A.M. — Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)
   - 1.1) Opening statement
   - 1.2) Items of current interest

2) 8:35 - 10:00 A.M. — Final Review of the License Renewal Application for Arkansas Nuclear One, Unit 2 (ANO-2) (Open) (MVB/CS)
   - 2.1) Remarks by the Cognizant Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the Entergy Operations, Inc. and the NRC staff regarding the license renewal application for ANO-2 and the associated final Safety Evaluation Report prepared by the NRC staff.

10:00 - 10:15 A.M. — *****BREAK*****

3) 10:~~15~~ 17 - 11:~~45~~ 51 A.M. — Draft Final Revisions to Standard Review Plan (SRP), Chapter 13, "Conduct of Operations" (Open) (SLR/EAT/MME)
   - 3.1) Remarks by the Cognizant Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final revisions to Sections 13.1.2 - 13.1.3, "Operating Organization," of SRP Chapter 13 and related NUREG-1791, "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54 (m)."

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

Do not need the court Reporter for the remainder of the meeting, per Dr. Wallis

~~11:45~~ 11:51 - 12:45 P.M. — *****LUNCH*****

4) 12:~~45~~ 48 - 2:~~45~~ 36 P.M. — Advanced Reactor Designs for Hydrogen Production (Open) (TSK/JHF)
   - 4.1) Remarks by the Cognizant Subcommittee Chairman
   - 4.2) Briefing by and discussions with representatives of the Department of Energy (DOE) regarding the status of DOE plans and research and development activities in support of advanced reactor designs for hydrogen production.

2:45 - 3:0~~0~~2 P.M. — *****BREAK*****

5) 3:0~~0~~ 2 - 4:~~00~~ 18 P.M. — Significant Recent Operating Events (Open) (JDS/RC)
Briefing by the Chairman of the ACRS Subcommittee on Plant Operations regarding significant recent operating events.

6) 4:~~00~~ 18 - 5:0~~0~~ 7 P.M. — Proactive Initiative (Open) (GEA/JHF/MME)
Discussion of proposed options for addressing ACRS proactive initiative on safety management.

5:0~~0~~ 7 - 5:15 P.M. — *****BREAK*****

7) 5:15 - 6:~~45~~ 26 P.M. — Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
7.1) Arkansas Nuclear One, Unit 2 License Renewal Application (MVB/CS/SD)
7.2) Draft Final Revisions to SRP Chapter 13, "Conduct of Operations" (SLR/EAT/MME)

FRIDAY, MAY 6, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

9) 8:35 - 10:00 A.M. — Steam Generator Tube Integrity Program (Open) (DAP/HPN)
9.1) Remarks by the Cognizant Subcommittee Chairman
9.2) Briefing by and discussions with representatives of the NRC staff regarding the objectives, technical approach, and results of the steam generator tube integrity program being conducted by the Argonne National Laboratory.

10:00 - 10:15 A.M. — *****BREAK*****

10) 10:15 - 11:45 A.M. — Digital Instrumentation and Control (I&C) Systems Research Plan (Open) (GEA/EAT/MRS)
10.1) Remarks by the Cognizant Subcommittee Chairman
10.2) Briefing by and discussions with representatives of the NRC staff regarding the digital I&C systems research plan.

Will do after lunch — 11) 1:15 - 1:45 P.M. ~~11:45 - 12:00 Noon~~ — Reconciliation of ACRS Comments and Recommendations (Open) (GBW, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.
together with Future ACRS....

12:00 - 1:00 P.M. — *****LUNCH*****

1:00 - 1:15 P.M. ***Break***

Do not need Court Report after lunch per Dr. Wallis.

12) 1:~~00~~ 45 - 2:00 2:45 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GBW/JTL/SD)

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

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

2:45 - 3:00 P.M. xx Break xxx

13) ~~2:00 - 6:30~~ 3:00 6:00 P.M. Preparation of ACRS Reports (Open)

(3:30-3:45 P.M. BREAK)

Discussion of the proposed ACRS reports on:

13.1) Arkansas Nuclear One, Unit 2 License Renewal Application (MVB/CS/SD)

13.2) Draft Final Revisions to SRP Chapter 13, "Conduct of Operations" (SLR/EAT/MME)

14) 6:30 - 7:00 P.M. Miscellaneous (Open) (GBW/JTL)

Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

- **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

522nd ACRS MEETING
May 5-6, 2005

NRC STAFF (5/5/05)

J. Zimmerman, NRR
P. Hiland, NRR
D. Trimble, NRR
R. Pelton, NRR
J. Yerokun, RES
P. Lewis, RES
D. Desaulniers, NRR
S. Arndt, RES
C. Ader, RES
W. Beckner, NRR
R. Barrett, NRR
Y. Orechwa, NRR
F. Eltawila, RES
J. Muscara, RES
M.K. Bagehi, NRR
K.B. Welter, RES
S. Basin, RES
J.E. Rosenthal, RES
D. Carlson, RES
S. Rubin, RES
I. Schoenfeld, OE
J. Persensky, RES
B. Rogers, NRR
R. Nease, RIV
J. Drake, RIV
L. Smith, RIV
M. Pribish, RII
K. Cozius, NRR
M. Morgan, NRR
J. Eads, NRR
R. Dipert, NRR
G. Cranston, NRR
D. Merzke, NRR
P.T. Kuo, NRR
K.R. Hsu, NRR
L. Train, NRR
J. Medoff, NRR
S. Lee, NRR
A. Lee, NRR
M.A. Mitchell, NRR
J.G. Lamb, NRR
R. McNally, NRR
R. Sublaneh, NRR
Y. Diaz, NRR
J. Tsao, NRR
G. Georgiev, NRR
C.Y. Li, NRR
M. Hartzman, NRR
H. Ashar, NRR
J.S. Guo, NRR
J.S. Ma, NRR
J. Rowley, NRR
S.K. Mitra, NRR
M. McConnell, NRR
G. Galletti, NRR
A. Szabo, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

M. Stroud, Entergy
R.B. Ruclur, Entergy
R. Ahrabuld, Entergy
M. Rinckel, AREVA-FANP
D.J. Lach, Entergy
T. Ivy, Entergy
G.G. Young, Entergy
M. Fallin, Constellation Energy
B. Kalinowski, AEP
R. Grumbir, AEP
N. Mosher, Entergy
S. Traiforos, LINK
S. Pope, ISL
C. Plott, MA&D
B. Shermon, State of Vermont
C. Brinkman, Westinghouse
A.D. Henderson, Dept. Of Energy
P. Puchaird, Sandia
R. Versluis, DOE-NE
J. Weil, McGraw-Hill

APPENDIX III: MEETING ATTENDEES (Cont'd)

522nd ACRS MEETING
May 5-6, 2005

NRC STAFF (5/6/05)
J. Davis, RES
A. Lee, RES
R. Crotan, RES
T. Mintz, RES
J. Muscara, RES
R. Barrett, RES
K. Karwoski, NRR
M. Waterman, RES
L. Lund, NRR
J. Lamb, NRR
M. Mayfield, NRR
W.E. Kemper, RES
S. Arndt, RES
T. Govan, RES
G. Tartal, RES
T. Hilsmeir, RES
H. Hamzehee, RES
M. Evans, RES
P. Loeser, NRR
C. Antonesa, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

S. Traiford, LINK
S. Dollary, Inside NRC

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

May 11, 2005

SCHEDULE AND OUTLINE FOR DISCUSSION
523rd ACRS MEETING
JUNE 1-3, 2005

WEDNESDAY, JUNE 1, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - 9:45 A.M. Interim Review of the License Renewal Application for the Point Beach Nuclear Plant, Units 1 and 2 (Open) (MVB/CS)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and the Nuclear Management Company, LLC regarding the license renewal application for the Point Beach Nuclear Plant, Units 1 and 2 and the associated draft Safety Evaluation Report prepared by the NRC staff, as well as the progress being made by the NRC staff and the applicant in resolving the issue of potential common-mode failure of the auxiliary feedwater pumps due to operator actions specified in the plant procedures, and related issues.

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

3) 10:00 - 11:30 A.M. Draft Commission Paper on Policy Issues Related to New Plant Licensing (Open) (TSK/MME)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding the draft Commission paper on policy issues (integrated risk and level of safety) related to new plant licensing.

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

4) 12:30 - 2:00 P.M. Fire Risk Requantification and Probabilistic Risk Analysis (PRA) Methodology for Nuclear Power Plants (Open) (SLR/HPN)
   4.1) Remarks by the Subcommittee Chairman
   4.2) Briefing by and discussions with representatives of the NRC staff and Electric Power Research Institute (EPRI) regarding the draft final NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," and related matters.

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

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

5) 2:15- 4:15 P.M. Draft Commission Paper on Proposed Alternatives to the Existing Single Failure Criterion (Open) (WJS/MRS)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding the draft Commission Paper on the proposed risk-informed and performance-based alternatives to the existing single failure criterion.

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

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

6) 4:30 - 7:00 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

6.1) Interim Review of the License Renewal Application for the Point Beach Nuclear Plant, Units 1 and 2 (MVB/CS)

6.2) Draft Commission Paper on Policy Issues Related to New Plant Licensing (TSK/MME)

6.3) Draft Final NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities" (SLR/HPN)

6.4) Draft Commission Paper on Proposed Alternatives to the Existing Single Failure Criterion (WJS/MRS)

6.5) Response to the April 26, 2005 Staff Requirements Memorandum Regarding the ACRS Assessment of the Quality of the NRC Research Projects (Tentative) (DAP/HPN/SD)

THURSDAY, JUNE 2, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

8) 8:35 - 10:00 A.M. Draft Safety Evaluation Report Related to Grand Gulf Early Site Permit Application (Open) (DAP/MME)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff and System Energy Resources Inc. regarding the NRC staff's draft Safety Evaluation Report related to the Grand Gulf Early Site Permit Application.

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

9) 10:15 - 11:45 A.M. Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants" (Open) (GEA/HPN)

9.1) Remarks by the Cognizant ACRS Member

9.2) Briefing by and discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI) regarding the draft final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," which endorses, with certain exceptions, NEI document, NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48 (c)," and the NRC staff's resolution of public comments.

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

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

10) 12:45 - 1:45 P.M. Status Reports on the Quality Assessment of Selected Research Projects (Open) (DAP/GBW/WJS/RC/EAT)

10.1) Report by the Chairman of the ACRS Panel regarding the status of the assessment of the quality of the thermal-hydraulic test program at the Penn State University.

10.2) Report by the Chairman of the ACRS Panel on the assessment of the quality of the containment capacity study being performed by the Sandia National Laboratories.

11) 1:45 - 2:30 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GBW/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.

12) 2:30 - 2:45 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (GBW, et al./SD, et al.)

Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

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

13) 3:00 - 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

13.1) Interim Review of the License Renewal Application for the Point Beach Nuclear Plant, Units 1 and 2 (MVB/CS)
13.2) Draft Commission Paper on Policy Issues Related to New Plant Licensing (TSK/MME)
13.3) Draft Final NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities" (SLR/HPN)
13.4) Draft Commission Paper on Proposed Alternatives to the Existing Single Failure Criterion (WJS/MRS)
13.5) Draft Safety Evaluation Report on the Grand Gulf Early Site Permit Application (DAP/MME)
13.6) Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants" (GEA/HPN)
13.7) Response to the April 26, 2005 Staff Requirements Memorandum Regarding the ACRS Assessment of the Quality of the NRC Research Projects (Tentative) (DAP/HPN/SD)

FRIDAY, JUNE 3, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 8:30 - 5:00 P.M. Preparation of ACRS Reports (Open)
Continue discussion of the proposed ACRS reports listed under Item 13.

15) 5:00 - 5:30 P.M. Miscellaneous (Open) (GBW/JTL)
Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

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

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
522nd ACRS MEETING
May 5-6, 2005

[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 May 5-6, 2005
2	Final Review of the License Renewal Application for Arkansas Nuclear One, Unit 2 (ANO-2)
	2. Arkansas Nuclear One - Unit 2 License Renewal presentation by Entergy [PowerPoint Slides]
	3. Arkansas Nuclear One - Unit 2 License Renewal Safety Evaluation Report staff presentation by NRR [PowerPoint Slides]
3.	Draft Final Revisions to Standard Review Plan (SRP), Chapter 13, "Conduct of Operations"
	4. Control Room Staffing presentation by NRR and RES [PowerPoint Slides]
4.	Advanced Reactor Designs for Hydrogen Production
	5. Hydrogen Production Using Nuclear Energy presentation by DOE [PowerPoint Slides]
5.	Significant Recent Operating Events
	6. Operating Reactors Summary and Analysis - CY 2003 - 2004 briefing by John D. Sieber, Chairman of the ACRS Subcommittee on Plant Operations [Power Point Slides]
	7. Operating Experience Briefing 2005-03, April 29, 2005 [Background Handout]
	8. Operating Report by John D. Sieber (Predecisional) on Operating Reactors Summary and Analysis for CY 2003 - 2004, dated May 1, 2005
6.	Proactive Initiative
	9. Proactive Initiative Safety Management presentation by John H. Flack, ACRS, [PowerPoint Slides]
9.	Steam Generator Tube Integrity Program
	10. Steam Generator Tube Integrity Program presentation by RES [PowerPoint Slides]
10.	Digital Instrumentation and Control (I&C) Systems Research Plan
	11. NRC Digital System Research Plan - FY 2005 Through FY 2009 presentation by RES [PowerPoint Slides]
11.	Reconciliation of ACRS Comments and Recommendations
	12. Reconciliation of ACRS Comments and Recommendations [Handout #1]
12.	Future ACRS Activities/Report of the Planning and Procedures Subcommittee
	13. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - April 6, 2005 [Handout #10]

Appendix V
522nd ACRS Meeting

MEETING NOTEBOOK CONTENTS

Color Code List - 522nd ACRS Meeting
Overtime Schedule

TAB	DOCUMENTS
2	Review of the Plant License Renewal Application and Final Safety Evaluation Report for Arkansas Nuclear One, Unit 2 1. Table of Contents 2. Meeting Schedule 3. Status Report, dated May 5, 2005
3	Control Room Staffing Exemption Requests 1. Table of Contents 2. Proposed Schedule 3. Status Report 4. Attachments 1. NUREG-0800, "Standard Review Plan," Chapter 13.0, "Conduct of Operations," Sections 13.1.2-13.1.13, "Operating Organization," Draft Revision 5 2. NUREG-1791, "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)," Final Report
4	Advanced Reactor Designs for Hydrogen Production 1. Proposed Schedule 2. INEEL/EXT-04-01816, "Design Features and Technology Uncertainties for the Next Generation Nuclear Plant" - Independent Technology Review Group
10	Digital Instrumentation and Controls Research Plan 1. Table of Contents 2. Proposed Schedule 3. Status Report 4. Attachments 1. "NRC Digital System Research Plan, FY 2005 - FY 2009 (draft)," transmitted to ACRS by memorandum dated April 2005 2. Letter from Mario V. Bonaca, ACRS, to Luis A. Reyes, EDO, "Digital Instrumentation and Controls Research Program," 9 June 2004.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
522nd FULL COMMITTEE MEETING

May 5-6, 2005

TODAY'S DATE: May 5, 2005

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

NAME	NRC ORGANIZATION
Jake Zimmerman	NRR/DRIP/RLEP
Patrick Hiland	NRR/DIPM/IROB
Dave Trimble	NRR/DIPM/IROB/IOHS
RICK PELTON	NRR/DIPM/IROB/IOHS
Jim Yerokun	RES
Paul Jenius	RES
David Desaulniers	NRR
Steven Arndt	RES/DET/ERAB
Charles Ader	RES/DRAA
WILLIAM BECKNER	NRR
RICHARD BARRETT	NRC/RES/DET
Yuri Orechwa	NRR/DSSA
FAROUK ELTAWILA	NRC/RES
JOSEPH MUSCARA	NRC/RES
M. K. BAGCHI	NRC/NRR/I
Kent B. Welter	NRC/RES
Sud Basu	NRC/RES
[illegible]	NRC/RES
Don Carlson	RES
Stuart Rubin	RES
Isabelle Schoenfeld	OE
J PERSENSKY	RES

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
522nd FULL COMMITTEE MEETING

May 5-6, 2005

TODAY'S DATE: May 5, 2005

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

NAME	NRC ORGANIZATION
Bill Rogers	NRR/DIPM
Rebecca Neese	RIV
Jim DRAKE	RIV
LINDA SMITH	RIV
MIKE STROUD	ENTERGY
Roger B Ruckar	Entergy
REZA AHRABLD	ENTERGY
Mark Rinckel	AREVA - FAND
David J. Lach	Entergy
Ted Ivy	Entergy
MICHAEL PRIBISH	RII
Kurt Cozens	NRR/DRIP
MICHAEL MORGAN	DRIP/RLEP
Johnny Eads	DRIP/RLEP
RICHARD DIPERT	NRR/DSSA/SPLB
GREG CRANSTON	NRR/DRIP/RLEP
DAN MERZKE	NRR/DRIP/RLEP
P T KUO	NRR/DRIP/RLEP
K R HSU	NRR/DRIP/RLEP
L. Tran	NRR/DRIP/RLEP

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
522nd FULL COMMITTEE MEETING

May 5-6, 2005

TODAY'S DATE: May 5, 2005

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

NAME	NRC ORGANIZATION
JAMES MEDOFF	NRR/DE/EMCB
SAMSON LEE	NRR/DRIP/RLEP
Arnold Lee	NRR/DE/EMEB
Matthew A. Mitchell	NRR/DE/EMCB
JOHN G. LAMB	NRR/DE/EEEB
Richard McNally	NRR/DE/EMEB
Ram Subbaraman	NRR/RLEP
Yoira Diaz	NRR/RLEP
JOHN TSAO	NRR/DE/EMCB
George Georgiev	NRR/DE/EMCB
Chang-Yang Li	NRR/DSSA/SPLB
MARK HARTZMAN	NRR/DE/EMEB
Hans Ashar	NRR/DE/EMEB
Jin-Sien Guo	NRR/DSSA/SPLB
John S Ma	NRR/DE/EMEB
Jonathan Rowley	NRR/DRIP/RLEP
S.K. MITRA	NRR/DRIP/RLEP
Matthew McConnell	NRR/DE/EEIB
Greg Galletti	NRR/DIPM/IPSB
Autumn Szabo	RES/DRAA/PRAB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
522nd FULL COMMITTEE MEETING
May 5-6, 2005

May 5, 2005
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
GARRY G. YOUNG	ENTERGY
MIKE FALLIN	CONSTELLATION ENERGY
BOB KALINOWSKI	AEP
Richard GRUMBIR	AEP
NATALIE Mosher	Entergy
SPYROS TRAIFOROS	LINK
STEVE POPE	ISL
Chris Plott	MA&D
BILL SHERMAN	State of Vermont
Charles Brinkman	Westinghouse
A. David Henderson	Dept. of Energy
Paul Prichard	Scandia
Rob Verskis	DOE-NE
~~M. A. BAGCHI~~	~~NRC NRR~~
Jenny Weil	McGraw-Hill

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
522nd FULL COMMITTEE MEETING

May 5-6, 2005

TODAY'S DATE: May 6, 2005

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

NAME	NRC ORGANIZATION
James Davis	RES/DET/MEB
ANDREA LEE	RES/DET/MEB
RICK CROTSAN	RES/DET/MEB
Todd Mintz	RES/DET/MEB
JOE MUSCARA	RES/DET/MEB
RICH BARRETT	RES/DET
Ken Karwoski	NRR/DE/EMCB
Mike Waterman	RES/DET/ERAB
Louise Lund	NRR/DE/EMCB
JOHN G. LAMB	NRR/DE/EEIB
Michael Mayfield	NRR/DE
WILLIAM E. KEMPER	RES/DET/ERAB/IAGS
STEVEN ARNDT	RES/DET/ERAB
Tekia Govan	RES/DET/ERAB
George Tartal	RES/DET/ERAB
Todd Hilsmeier	RES/DRAA/PRAB
HOSSEIN HAMZEHEE	RES/DRAA
Michele Evans	RES/ERAB/DET
Paul Loeser	NRR/DE/EEIB
Christina Antonescu	RES/DE/ERAB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
522nd FULL COMMITTEE MEETING
May 5-6, 2005

May 6, 2005
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
SPYROS TRAIFOROS	LINK
Steven Dottey	Inside NRC

ITEMS OF INTEREST

522nd ACRS MEETING

MAY 5, 2005

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
522nd MEETING
April 7-8, 2005

Page

STAFF REQUIREMENTS

ORDERS

SPEECHES

CONGRESSIONAL TESTIMONY

SIGNIFICANT ENFORCEMENT ACTIONS:

INSIDE NRC

IN RESPONSE, PLEASE
REFER TO: M050329A

April 27, 2005

MEMORANDUM FOR: John T. Larkins
Executive Director, ACRS/ACNW

FROM: Annette L. Vietti-Cook, Secretary ***/RA by Andrew L. Bates Acting For/***

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON NSIR PROGRAMS, PERFORMANCE, AND PLANS, 9:30 A.M., TUESDAY, MARCH 29, 2005, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff on the Office of Nuclear Security and Incident Response programs, performance, and plans. The staff should provide to the Commission recommendations for when to use each of the vehicles used to communicate security matters with licensees (i.e., security advisories, regulatory issue summaries, information notices, etc.). To the extent practicable, the staff should strive to issue a publicly releasable summary of the action taken as soon as possible.

The staff should provide to the Commission offices, within a month of the date of this staff requirements memorandum, a timetable for completing this task.

(EDO) (SECY Suspense: 5/27/05)

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

IN RESPONSE, PLEASE
REFER TO: M050407

April 25, 2005

MEMORANDUM FOR: John T. Larkins
Executive Director, ACRS/ACNW

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

SUBJECT: STAFF REQUIREMENTS - MEETING WITH THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), 1:30 P.M., THURSDAY, APRIL 7, 2005, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission met with the ACRS to discuss several topics of mutual interest. The Committee should provide the Commission a list of research projects it intends to review in the short term as part of its assessment of research quality, with an indication of the methodology the Committee will use for the reviews. The Commission appreciates the Committee's undertaking of this effort in addition to providing a biennial report to the Commission on the NRC safety research program.

The Committee should review the staff's Regulatory Guide that will implement the revised 10 CFR 50.46. In addition, the Committee should consider reviewing upcoming revisions to NUREG-0800 sections that involve significant changes.

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

IN RESPONSE, PLEASE
REFER TO: M050316B

April 5, 2005

MEMORANDUM FOR: Michael T. Ryan, Chairman
Advisory Committee on Nuclear Waste

John T. Larkins
Executive Director, ACRS/ACNW

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

SUBJECT: STAFF REQUIREMENTS - MEETING WITH ADVISORY COMMITTEE ON NUCLEAR WASTE (ACNW), 9:30 A.M., WEDNESDAY, MARCH 16, 2005, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the Advisory Committee on Nuclear Waste (ACNW). The Commission appreciates the expert technical advice the Committee provides. The Committee should continue to follow closely the revision of the International Commission on Radiological Protection's (ICRP's) standards. The Commission anticipates ACNW's review of NRC's proposed rulemaking on the control of solid materials. Additionally, the Commission looks forward to the Committee's white paper on its approach to addressing low-level radioactive waste issues. The Committee should consider, moving low-level waste issues to a tier one priority if a specific need arises requiring Agency action. ACNW should provide feedback to the Commission, as appropriate, on these and other salient issues, including waste incidental to reprocessing.

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

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED 04/21/05

SERVED 04/21/05

COMMISSIONERS:

Nils J. Diaz, Chairman
Edward McGaffigan, Jr.
Jeffrey S. Merrifield
Gregory B. Jaczko
Peter Lyons

In the Matter of DUKE ENERGY CORPORATION (Catawba Nuclear Station, Units 1 and 2)	)	Docket Nos. 50-413-OLA, 50-414-OLA

CLI-05-10

MEMORANDUM AND ORDER

This proceeding arises from Duke Energy Corporation's application for a license amendment to authorize the use of four lead test assemblies of mixed oxide (MOX) fuel in one of its Catawba nuclear reactors. On March 10, 2005, the Licensing Board issued its final partial initial decision[1] on the security contention brought by the Blue Ridge Environmental Defense League ("BREDL") to challenge certain exemptions Duke Energy Corporation sought for its Catawba facility during testing of MOX assemblies. Because it contains safeguards information, the order has not been made public. The Board did, however, issue a public notice of the decision, indicating that, *subject to certain conditions*, Duke had met its burden to show that its requested exemptions from the requirements of 10 C.F.R. Parts 11 and 73 are appropriate and that its physical protection system will "provide high assurance that activities involving the MOX fuel will not be inimical to the common defense and security nor constitute an unreasonable risk to the public health and safety."[2]

The March 10 order was the Board's final order in this case, and none of the parties sought review of it. Nevertheless, the Commission has decided to review the Board's order pursuant to 10 C.F.R. § 2.786(a).[3] Before proceeding further, the Commission specifically requests the parties to brief the issue of the necessity of the conditions the Board imposed for purposes of receipt of the MOX lead test assemblies.

The briefs should not exceed 25 pages and should be filed for receipt by the Commission by close of business on May 2, 2005. Parties may file reply briefs, limited to 10 pages and consisting only of rebuttal, for receipt by the Commission by

May 9, 2005. The parties are reminded of the importance of compliance with the procedures regarding submission of safeguards information.

IT IS SO ORDERED.

For the Commission

/RA/

Annette L. Vietti-Cook
Secretary of the Commission

Dated at Rockville, Maryland,
this 21st day of April, 2005.

---

[1]*See Duke Energy Corp.* (Catawba Nuclear Station, Units 1 and 2), unpublished "Final Partial Initial Decision (Issues Relating to BREDL Security Contention 5)" (Mar. 10, 2005).

[2]*See* "Notice of Final Partial Initial Decision (Issues Relating to BREDL Security Contention 5)" (Mar. 10, 2005).

[3]The Commission's new adjudicatory rules do not apply to this case, which began before their promulgation. *See* Final Rule: "Changes to Adjudicatory Process," 69 Fed. Reg. 2182 (Jan. 14, 2004).

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED 04/20/05

SERVED 04/20/05

COMMISSIONERS:

Nils J. Diaz, Chairman

Edward McGaffigan, Jr.
Jeffrey S. Merrifield
Peter B. Lyons
Gregory B. Jaczko

In the Matter of EXELON GENERATION COMPANY, LLC [redacted] Site Permit for Clinton ESP Site)	Docket No. 52-007-ESP
In the Matter of DOMINION NUCLEAR NORTH ANNA, LLC (Early Site Permit for North Anna ESP Site)	Docket No. 52-008-ESP
In the Matter of SYSTEM ENERGY RESOURCES, INC. (Early Site Permit for Grand Gulf ESP Site)	Docket No. 52-009-ESP
In the Matter of LOUISIANA ENERGY SERVICES, L.P. (National Enrichment Facility)	Docket No. 70-3103-ML
In the Matter of [redacted] Inc.	Docket No. 70-7004

(American Centrifuge Plant)

CLI-05-09

MEMORANDUM AND ORDER

On March 18, 2005 the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel issued a Memorandum, LBP-05-07, 61 NRC ___, certifying certain questions to the Commission regarding "mandatory hearing" requirements in NRC enabling legislation and in NRC regulations. The Chief Judge's Memorandum addressed the first four proceedings captioned above. On March 28th, USEC (the applicant in the fifth proceeding) filed with the Commission a motion for leave to submit its views on the certified questions. The Commission hereby grants review of those questions. In doing so, we follow our "customary practice" of accepting Board-certified questions. *See, e.g., Duke Energy Corp.* (Catawba Nuclear Station, Units 1 and 2), CLI-04-11, 59 NRC 203, 209 (2004); *Private Fuel Storage, L.L.C.* (ISFSI), CLI-01-12, 53 NRC 459, 461 (2001).

USEC argues that the certified questions are as relevant to its own application to construct and operate a uranium enrichment facility as they are to the Louisiana Energy Services' pending application (captioned above). According to USEC, both applications were filed under the same statutory and regulatory provisions, both concern the same kind of facility, both are subject to mandatory hearings, and the two proceedings' "Notice[s] of Hearing and Order" are substantially identical.

The Commission agrees that USEC should have the opportunity to present its views on the certified questions. The Commission therefore grants USEC's motion and establishes the following filing schedule for both USEC's brief and any response briefs. No later than 14 days after issuance of this Memorandum and Order, USEC may file with the Commission a brief setting forth its views on the certified questions. USEC's brief may not exceed 20 pages, exclusive of the tables of contents and authorities (both of which we require). No later than 14 days after USEC files its brief, the parties in the remaining four above-captioned proceedings (exclusive of the NRC Staff) and the petitioners to intervene in the *USEC* proceeding may file response briefs with the Commission. Response briefs may address both USEC's brief and the points the Chief Judge raised in LBP-05-07, but need not repeat arguments already raised in the records before the various Boards in these proceedings. Each response brief may not exceed 20 pages, exclusive of the tables of contents and authorities (both of which we require).

For reasons unique to these certified questions, we establish a later filing deadline for the NRC Staff's reply brief. The Chief Judge reviewed, *inter alia*, the agency's hearing notices in the first four above-captioned cases, the Staff's various briefs to the Board regarding the certified questions, and the procedural regulations at issue. But he repeatedly indicated in LBP-05-07 that these various documents, or sets of documents, appear internally inconsistent as to the certified questions. To provide the Staff a sufficient opportunity to address these issues and the certified questions fully and to respond to any suggestions and arguments by other parties, we grant the Staff an additional week -- until 7 days after all other response briefs are filed -- to file its response brief.

The Staff's brief should address LBP-05-07, the certified questions, USEC's brief, and all other response briefs. Because we are establishing a particularly broad scope for the Staff's response brief, we impose upon it no page limit. As with the other parties and participants, we require the Staff to include tables of contents and authorities. Finally, though we are *permitting* all other entities to file their various briefs, we *require* that the Staff file its response brief.

For the Commission

/RA/

Annette L. Vietti-Cook
Secretary of the Commission

Dated at Rockville, MD

this 20th day April, 2005

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ENERGY CHOICES FOR THE SOUTHWEST – THERE'S NO FREE LUNCH

Prepared Remarks by

The Honorable Peter B. Lyons
Commissioner
U. S. Nuclear Regulatory Commission

before the

5th Annual Western Energy Summit
Scottsdale, AZ
March 31, 2005

I grew up in southern Nevada and attended the University of Arizona. Thus, returning here tonight is something of deja vu. I recall family trips to Arizona, as well as frequent drives through your Valley of the Sun to the University.

This isn't the same Valley that I recall from years ago. It won't surprise you to know that there are a whole lot more folks here than when I first saw it. I traveled through a Phoenix of 17 square miles, today, the city exceeds 430 square miles. I remember Las Vegas with less than 50,000 people, not the two million there today.

I've watched the transformation of these desert areas into oases with air-conditioned houses, green lawns, and many with swimming pools, to say nothing of endless golf courses.

Your Summit focuses on Western energy needs, and you've picked a topic of immense importance given the spectacular growth in this region. While the annual growth in our country's electricity consumption is forecast to be 1.8%, Arizona's gross state product has grown by almost 7% annually. Your growth rate in electricity is twice the national average, and you've been increasing your use of natural gas by 7% annually.

The water demands of the Southwest are alarming, especially with the recent drought years. Lake Mead has fallen to record low levels, with an 85-foot drop in the last four years. The artesian wells in Las Vegas dried up decades ago, and the translation of the city's name, "the meadows," is probably lost on newcomers today. Short of massive desalination efforts, which require still more energy, it's not clear where the water is going to come from for the next golf course.

The beautiful open vistas of Arizona also mean that you drive a long way between cities. Thus, the five million cars in Arizona require lots of gasoline. But as I wrote this speech, oil was selling at $56 a barrel and gasoline prices were at an all time high. To meet our nation's transportation demands for the next 20 years, our petroleum consumption will increase by 33%, requiring an increase in oil imports of 60%.

Sustainable Electricity Sources

The Senate conducted a hearing last year devoted to discussion of sustainable electricity sources, those sources from which we can reliably forecast new generation for the next century. Witnesses agreed that only coal, renewables, and nuclear energy were candidates. Oil and natural gas were not part of that Senate hearing. While experts argue whether the peak in global oil production is occurring now, or may occur in 20 years, oil is a finite resource.

Supplies of natural gas are better than oil, but still finite. Hybrid cars should delay the day when oil is too expensive to power our cars, but we'll have to switch to hydrogen-, ethanol-, or electric-powered cars within a few decades. Since we don't pump hydrogen from the ground, it will take more energy to generate that hydrogen. Massive use of ethanol would present some major challenges. And if we use electric cars, we need still more generation capability.

Since nuclear energy is on this list of sustainable sources, and since I'll be speaking about it in this talk, I need to note that I'm not speaking to you as an advocate of nuclear power. That is not the role of the NRC as a regulatory agency or the role of my current position, nor can I speak for the Commission itself. Instead, today I speak to you as an "almost native" of the Southwest, who has spent his professional career in national service, and as an environmentalist who has enjoyed hiking, climbing, and backpacking in our magnificent wilderness and desert areas. I also speak as a student of science who wishes to bring its disciplined approach to the considerable challenges which face us as a nation.

Risk/Benefit Tradeoffs

To sustain your population and economic growth into the future, your choices for new energy sources are going to be increasingly confined to those three sustainable sources. And – to remind you of the title of my talk – there simply is "no free lunch" as you make those choices. You'll be faced with the reality that every source of energy brings with it a set of costs or risks as well as benefits that have to be carefully weighed. Tonight, I'd like to talk a little about the tradeoffs you need to make here in the Southwest, and also note that across our nation, all of our citizens will be facing comparable choices.

To varying degrees, each energy source has costs associated with the extraction of fuel or raw materials from the earth, the refinement of fuel or raw materials, and the use of the fuel to produce energy. Also, each may have risks associated with the generation of wastes, the release or control of those wastes, any adverse exposures to people, and the possibility of accidental releases of hazardous materials.

In discussing risk, we must be mindful that most people tend to perceive risk very personally and not always objectively. But from a scientific perspective, properly defined and quantified risk metrics provide a useful tool for comparisons and informed policy choices. The choice of a risk metric can vary, but in every case where we use such metrics, care is needed to ensure a full understanding of the data and assumptions used, the definition of the metric, and the uncertainties in calculating the metric. With that caution, I'll turn to some details on each of those sustainable sources.

Renewables

Arizona derives about 8% of its electricity from renewable hydropower. That source would be hard to expand and may even contract if the current long-term drought continues. Beyond hydro, your use of renewables is very small. In a quick literature search, I was amazed that I couldn't find an operating commercial wind farm in Arizona. I learned that Arizona Public Service (APS) hopes to have its Eastern Arizona Wind Energy Center operating by the end of this year at St. Johns, but at 15 megawatt capacity, that's not much of a dent in your needs.

The best news for renewables is that the fuel, for example the water, sun, or wind, is typically free. But that doesn't mean the electricity is free. Today, solar electricity systems are not competitive with current electricity costs, while wind is getting closer to economic viability. Construction is required, as is maintenance, along with transmission lines to reach markets.

Calculations of the tons of materials required per megawatt-year of electricity for different energy sources are illuminating. Coal requires about 11 tons (excluding the weight of the coal itself), nuclear about 15, wind about 36, and various solar approaches are from about 140 for photovoltaic to 370 for thermal systems.

Both solar and wind power are very low density energy sources, meaning it takes lots of area to collect much energy. You've got lots of open space here in the West, but the requirements still aren't small. One estimate suggests a wind farm

equivalent in output to a typical nuclear plant would occupy around two thousand square miles. Wind farms have also raised concerns about visual pollution — which is the notion that "don't build it where I can see it" – and damage to bird and bat populations.

Solar and wind power by themselves can't provide baseload power, the steady relatively constant level of electrical power that drives much of our economy, as opposed to peaking power demands at certain times of the day or year. Both sources only generate about 30% of their capacity just because of their intermittent nature. That intermittence is a real issue, as is the fact that they simply may not be available for days on end depending on weather patterns. Thus, if they become a major supplier of the grid, then backup power sources must also be built and kept ready for operation.

Alternatively, maybe we can devise systems for storing large amounts of energy, then extra renewable capacity could provide energy for storage during their operational times. That's done in some parts of the world now. For example, wind power pumps water into reservoirs in some parts of Scandinavia. In the future, solar or wind power might be used for production of hydrogen, another way of storing energy. But all that will add cost.

Coal

Arizona gets 40% of its electricity from coal plants. While coal plants are becoming cleaner with installation of scrubbers, they still release particulates, sulfur and nitrogen oxides, heavy metals, and radioactive materials. Due to its sulfur oxide emissions, coal is the largest contributor to acid rain in the country. And it may surprise you to know that, for a plant of comparable capacity, a coal plant releases about 100 times more radioactivity into the environment than a nuclear plant, even though both amounts are very low.

A big plus of coal plants is their low cost of electricity generation, just slightly more than the nuclear power plants. But future costs will increase if emissions are fully captured. No serious attempt has yet been made to fully collect carbon dioxide, the primary gas of concern for global warming.

The air pollution from coal plants can be a real issue; a recent study suggests 15,000 premature deaths annually in the U.S. due to airborne emissions from coal plants. It also leads to degraded visibility. This visual pollution, largely from fossil fuels like coal, is of concern in many areas of the country, including your Grand Canyon. Furthermore, coal mining is hazardous and substantial transportation resources are needed to move coal.

As one measure of risk, several groups have studied the life-cycle emissions of various energy sources. In these studies, their conclusions may, at first, seem surprising. For one example, even though reactors don't emit carbon dioxide, mining operations do; thus, on a life-cycle basis, nuclear power is not free of such emissions. Furthermore, these calculations have significant uncertainties and depend on the detailed assumptions made, but they still are useful for general guidance.

One study from the University of Wisconsin shows coal producing more than 1000 tons of carbon dioxide for each gigawatt-hour of electricity, with nuclear at 17, wind at 14, and solar photovoltaics at 39.

A similar Japanese study shows carbon dioxide emissions from wind power to be about 50% more than nuclear power, with nuclear about 50 times less than coal.

Another study from the International Energy Agency shows particulate emissions from coal plants to be as much 660 kilograms per gigawatt-hour, with nuclear a factor of 300 lower and with hydropower and wind a factor of about 120 lower.

Advanced coal plants currently on the drawing board will completely capture all emissions. This effort is led by the DOE FutureGen plant project, but it's far too early to predict the costs or even the feasibility of a plant of this type.

Nuclear Power

Arizona derives 30% of its electricity from nuclear power, and nationwide about 20% of our electricity is from this source. Nuclear power from existing plants is cheaper than from coal plants, but new nuclear plants will be costly. Even though nuclear energy is very capital intensive, several studies show that new plants may be competitive with other energy options.

Nuclear energy, both from our civilian plants and our nuclear navy, has demonstrated a superb safety record. But the public must be confident of their continued safety. Furthermore, with today's fear of terrorism, the public must be satisfied that adequate security is in place at each of our plants. I'll discuss later that my job at the NRC is to focus on safety and security

of our plants.

Nuclear power releases minimal airborne emissions, while collecting all of its nuclear wastes for later disposition. Relative to coal wastes are very small in volume, well contained, and well controlled. But spent fuel is hazardous and remains so for many years. We currently store spent fuel at reactor sites. While that storage is safe, space for future storage may be limited at some sites. Progress toward an underground repository for spent fuel has been glacial and far behind predictions.

In contrast, some other countries do not plan to dispose of spent fuel, but instead reprocess it to extract and reuse plutonium. That plutonium is an energy resource if it's reused, while if left in the spent fuel and placed in a permanent repository, it is a contributor to long-term health hazards. Reprocessing results in better utilization of the original uranium and less toxicity and volume in the final wastes, but also complicates nuclear non-proliferation concerns. Issues with reprocessing also include cost and environmental impacts.

There are studies underway in several countries to move beyond reprocessing with technologies like transmutation and advanced reactors. That work, if successful and cost effective, would render the final waste products from reactors no more toxic than the original uranium ore after about 300 years and would more efficiently recover the original energy content. But demonstration of these ideas beyond the laboratory stage is well in the future.

It's interesting that concerns over global warming are leading some outspoken environmentalists to recommend further use of nuclear power, in contrast to strong opposition from the "environmental community" just a few years ago. James Lovelock, the creator of the Gaia environmental movement, recently stated: "Now that we have made the Earth sick, it will not be cured by alternative green remedies, like wind turbines and biofuels alone. This is why I recommend instead the appropriate medicine of nuclear energy as part of a sensible portfolio of energy sources."

Radiation Phobia

There's one more risk area, radiation, to which I'd like to devote a little more time. Many people have a strong fear of radiation, maybe because it was first associated with the atomic bomb. They may not be aware that we live in a sea of natural radiation and are exposed to still more radiation through many activities in modern life. Yet, we've clearly adapted to living with this radiation background. For this discussion, I'll be using the term "mrem," a unit of radiation exposure.

In the U.S., the average annual background dose is 300 mrem - not including medical exposures, with very wide variations around the country. If you live at sea level, cosmic rays contribute about 30 mrem each year to your dose, while at higher elevations, like in Leadville Colorado, that dose rises to about 120 mrem. The rocks around you provide more radiation – in the Atlantic coastal plain that only contributes 15-35 mrem per year, while this terrestrial component is about 90 mrem on the Colorado plateau.

There are radioactive materials inside our bodies, and they contribute about 20 mrem annually – which is also the source of some jokes used by nuclear physicists about the dangers of sharing a bed with someone. Naturally occurring radon gas is common around the world. The average annual U.S. dose from radon is about 200 mrem, again with wide variations.

Medical procedures also have historically contributed an average dose of 60 mrem per year, and this dose is estimated to have doubled in recent years in the U.S. Just in the U.S., there are about 180 million diagnostic x-rays annually, along with seven million nuclear medicine procedures.

If we compare background levels around the world, the variation from place to place is more than a factor of 20. Some prominent public places have especially high annual backgrounds, derived from the stone used in their construction. For example, Grand Central Station in New York City at 525 mrem per year is one example. Some hot springs are particularly "hot" in a radioactive sense too – nearby residents experience exposures up to 10,000 mrem annually.

Many activities of daily life also lead to exposures. A cross-country airplane trip adds about 5 mrem, a dental x ray about 6 mrem. Therapeutic medical treatments are typically in the millions of mrem range - but only for the part of the body that is treated.

To compare to a nuclear plant, the average public dose within 50 miles of a nuclear plant is around 0.05 mrem each year. The law requires that someone living at the boundary of any future spent fuel repository receive no more than 15 mrem annually. And the regulatory limit at the boundary of a nuclear power plant is 25 mrem per year. Note that these levels are well within the variability of natural background.

There's no doubt that radiation can be dangerous. Short-term exposures to the whole body of 500,000 mrem are typically fatal. At lower doses, cancer is observed in some cases. Cancer can occur after long latency periods, greatly complicating measurement of cause and effect. Further, about 20% of us will die of various forms of cancer quite independent of any radiation exposures, which makes it very hard to determine the origin of someone's cancer.

Health risks of radiation exposure can only be estimated with reasonable certainty at radiation levels that are far greater than background levels. Health effects have been demonstrated only at doses exceeding 10,000 mrem delivered in a very short time.

Also, there is vast uncertainty in any real effects of low doses of radiation delivered over long times. In discussing the question of radiological risk from low doses, the National Academy of Sciences noted that ". . . the possibility that there may be no risks from exposures comparable to external natural background radiation cannot be ruled out. At such low doses and dose rates, it must be acknowledged that the lower limit of the range of uncertainty in the risk estimates extends to zero."

Conclusion

Let me close by noting that the mission of the Nuclear Regulatory Commission is to develop, oversee, and enforce programs to assure that adequate safety and security is maintained in all operations involving civilian nuclear technologies, and assure the American public that their safety is our main consideration. The actions of the NRC, the responses from industry, and the impact of those actions on the public, will influence the role that nuclear power may play in the energy choices that you will make in future years.

I've provided a brief discussion of some of the tradeoffs that must be made to supply our energy thirst in the future. Energy production, just as life, is not risk-free. Each of us needs to carefully weigh options for our activities and understand the risks and benefits of our actions. In the case of energy, I hope these thoughts will help to inform your choices for the energy portfolio that you want to power the West in years to come.

Above all, remember that energy choices require tradeoffs between risks and benefits. There is no easy answer, and there is no choice that is free of risk. Or, as I noted at the start, there is "no free lunch" in energy choices.

STATEMENT SUBMITTED

BY THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

TO THE

COMMITTEE ON ENERGY AND NATURAL RESOURCES

UNITED STATES SENATE

CONCERNING

NUCLEAR POWER 2010 INITIATIVE -

NEW NUCLEAR POWER GENERATION

IN THE UNITED STATES

PRESENTED BY

DR. NILS J. DIAZ

CHAIRMAN

Submitted: April 26, 2005

Introduction

Thank you, Mr. Chairman and members of the Committee. It is a pleasure to appear before you as you consider "Nuclear Power 2010 - New Nuclear Power Generation in the United States." My testimony today on behalf of the Commission will focus on actions the Commission has taken and is taking to ensure the continued safe and secure uses of nuclear technology and to provide a stable, efficient, and predictable framework for licensing and regulation of the civilian uses of nuclear materials. In particular, I will address actions relating to early site permits, design certification, and combined license applications for new reactors.

The U.S. Nuclear Regulatory Commission (NRC) is dedicated to the mission mandated by Congress - - to ensure adequate protection of public health and safety, common defense and security, and the environment - - in the application of nuclear technology for civilian use. In carrying out this mission, the Commission is mindful of the need to enhance safety, security, and regulatory predictability, when appropriate and justified. We take very seriously our commitment to enable the safe and secure beneficial use of nuclear power.

Regulatory Framework for New Reactor Licensing

The NRC is prepared to discharge its responsibilities regarding licensing of new nuclear power plants, though enhancements and resources are continually being assessed. In 1989, the NRC instituted a new combined construction/operating license process through the promulgation of 10 CFR Part 52, as an alternative to the separate construction and operating licensing steps specified in 10 CFR Part 50. The process was later addressed by Congress in the Energy Policy Act of 1992. The Part 52 licensing process is designed to resolve safety and environmental issues, including emergency preparedness and siting issues, early in the process

and, thus, to provide a more stable, efficient, and predictable regulatory framework for utilities that might wish to pursue a new reactor license.

Part 52 established three new components of our licensing structure - - design certification, early site permit, and combined operating license. First, NRC developed a standard design certification process by which the NRC extensively reviews a proposed reactor design and then, if appropriate, approves the design through public rulemaking. The Commission has already certified three new reactor designs and codified them in the regulations, making them available for new plant orders. The proposed design certification rule for a fourth design was recently published for public comment. The NRC is also prepared to receive a fifth design certification application in the summer of 2005. As a result of experience gained during previous design certification reviews and to promote additional regulatory effectiveness, the NRC encourages early communication with potential applicants to identify unique design features or challenging licensing issues through the pre-application process. Currently, the NRC is engaged in conducting pre-design review or preliminary review discussions on six additional reactor designs, so we could receive several more design certification applications in the near future. I cannot stress enough the need for applicants to provide complete and high quality technical information.

The NRC also established a process for obtaining an early site permit, which allows applicants to seek approval of sites for new reactor units separate from an application for a construction permit or combined construction/operating license. By obtaining an early site permit, applicants can resolve site-related issues, including certain environmental issues, before the early site permit is issued. The NRC received three early site permit applications in late 2003 for sites at which operating reactors already exist in Virginia, Illinois, and Mississippi.

Schedules are in place to complete the safety reviews and environmental impact statements in approximately two years from the date of an application. In fact, the NRC staff has already issued draft safety evaluation reports on all three early site permit applications. Also, draft environmental impact statements for two of the three early site permit applications have been issued for public comment. The NRC staff is currently reviewing the public comments received on these documents. The mandatory adjudicatory hearings associated with the early site permits are currently ongoing; conclusion of these hearings is, in part, dependent upon completion of all associated staff reviews. While I am pleased to be able to provide this information on the status of the reviews of the three early site permit applications, the Commission serves in an adjudicatory capacity in reviews of our Licensing Board's decisions and, thus, it would be inappropriate for me to address substantive issues associated with the resolution of these early site permit proceedings.

Finally, Part 52 provides for a combined construction/operating license process which allows applicants to seek, in a single application, a license authorizing both construction and operation. This leads to combining adjudication of licensing issues in one hearing, instead of the two hearings that have attended the licensing process utilized previously. Furthermore, the efficiency of NRC's safety-focused reviews would be substantially increased if applicants utilize an early site permit and certified design in their combined license applications. We believe this process will provide the needed stability and predictability in licensing reviews for new nuclear power plants, key components of which have been, or are being, demonstrated by the new reactor design certifications and the ongoing work on the early site permit applications. The NRC is working to clarify and refine the 10 CFR Part 52 licensing process further in order to incorporate recent experience gained from design certification reviews, current early site permit reviews, discussions with nuclear industry representatives, and input from the public.

I am convinced that these measures, individually and in combination, are providing a means to enhance safety for nuclear power generation in the future.

License renewal for existing operating reactors provides another example of how the NRC has sharpened the safety focus of its licensing process. The NRC has received license renewal applications for 48 reactor units and has approved 20-year extensions for 30 reactor units; an additional application covering two reactor units was recently returned to a licensee as unacceptable for docketing. These reviews have been consistently completed in a timely fashion, meeting the NRC's schedule of 22 months for completing a review without a hearing request and 30 months when a hearing is requested. NRC is using experience gained from the license renewal process to improve the efficiency of Part 52 combined license application reviews. The agency is committed to a continuing holistic improvement of our regulatory review processes, with a sharpened focus on matters important to safety. This has been well demonstrated by the use of disciplined review processes in many licensing activities, including the review of applications for license renewals and for power uprates. Our experience to date is that an application that is complete, of high technical quality, and responsive to staff questions has a direct impact on the NRC's ability to make the appropriate safety determinations, meet our review schedules, and stay within resource estimates.

The Commission has also worked actively to ensure that its adjudicatory proceedings are conducted in a fair, effective, and disciplined manner, now and in the future. For example, the Commission revised its rules of practice for agency adjudication early last year and has just published a final rule that adopts model milestones for presiding officers to use in scheduling and managing hearings. The Commission continues to exercise oversight of the adjudicatory process.

New Reactor Construction

Licensing of new reactors requires a revised approach for inspecting new reactors during construction and pre-operational testing. Key challenges include establishing a state-of-the-art construction inspection framework; ensuring that safety is built into each phase, whether it be design, construction, or operation; ensuring the availability of an adequate number of qualified inspection personnel; ensuring that appropriate information systems are in place to efficiently and effectively perform the necessary inspections, tests, analyses, and acceptance criteria verifications; and responding to the anticipated use of multi-national modular construction techniques.

The industry is presently considering the construction of new plants in a modular fashion, with many of the modules fabricated at locations away from the plant site, including facilities located abroad. The industry's estimate for completing construction varies by plant design, but has been in the range of about 60 months and could be decreasing as new modular techniques are added.

The NRC is paying special attention to human resource requirements, especially the need for the construction inspection staff to have the requisite combination of construction knowledge and inspection skills. The NRC is utilizing the know-how of our senior inspectors with construction experience and incorporating their insights and lessons learned into the revised construction inspection program, procedures, and training. The NRC is actively revising its construction inspection program to provide an enhanced safety focus and ensure timely support to all phases of the license application and construction processes. We are working with industry and public stakeholders as we go through this revision process and are confident

that our revised program will be well established and in place before new construction would begin.

Resources for the Expected Demand for New Reactor Licensing

The FY 2006 President's budget request includes $37 million for the NRC's continuing work on new reactor licensing, including review of the three early site permit applications, review of two standard design certification applications, and development and updating of the agency's regulatory structure to accommodate new, advanced reactor designs. The demand for new reactor licensing is now expected to grow more rapidly than previously anticipated and budgeted. These demands have been identified in response to the Department of Energy's Nuclear Power 2010 Program solicitations, industry letters, and press releases.

Although specific plans are not yet available from the industry, the NRC may be faced with a significant increase in its workload for new reactor licensing, including receipt of up to five combined license applications beginning in 2007-2008. To meet this expected increased demand, NRC would need to begin preparatory activities soon to accommodate such large growth. This includes ensuring a state-of-the-art regulatory framework and conducting associated technical activities, obtaining sufficient NRC staff and contractors in the relevant disciplines, securing space, developing and conducting training, and putting in place the appropriate organizational structure that would allow timely completion of the newly anticipated work. The NRC will also have to assess how to manage such a workload in light of other high priority activities, such as security and fuel cycle work. In short, NRC must determine the additional substantial resources for nuclear reactor licensing that will be needed to fully support the Nuclear Power 2010 initiative.

Summary

The Commission is dedicated to enabling the safe and secure use and management of radioactive materials and nuclear technology for beneficial civilian purposes. To that end, the Commission is fully committed to making sure that our agency is ready to meet the expected demand for new reactor licensing. The Commission believes the agency is prepared to accept and process applications in accordance with the applicable laws and regulations, continuing to focus on safety and utilizing risk-informed and performance-based regulation as appropriate. The NRC's Part 52 processes are safety-focused and should be stable, efficient, and predictable. We are also addressing our challenges. These include ensuring a strong regulatory and oversight framework; meeting the NRC's resource needs associated with the potential for receiving multiple combined license applications; establishing our technical and legal staff and contractor requirements early; and seeking additional funding as needed. We will continue to work with stakeholders to address issues associated with implementation of our licensing process. The Commission has benefitted from strong Congressional oversight, and we will continue to keep Congress informed about the impact of new reactor activities on the NRC.

I appreciate the opportunity to appear before you today, and I welcome your comments and questions.

EA-05-071 - Davis-Besse (FirstEnergy Nuclear Operating Company)

April 21, 2005

EA-03-025; EA-05-066; EA-05-067; EA-05-068;
EA-05-069; EA-05-070; EA-05-071; EA-05-072

Mr. Gary Leidich, President
FirstEnergy Nuclear Operating Company
76 S. Main St.
Akron, OH 44308

SUBJECT: NOTICE OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTIES - $5,450,000; (NRC OFFICE OF INVESTIGATIONS REPORT NO. 3-2002-006; NRC SPECIAL INSPECTION REPORT NO. 50-346/2002-08 (DRS)); DAVIS-BESSE NUCLEAR POWER STATION

Dear Mr. Leidich:

This refers to the U.S. Nuclear Regulatory Commission's (NRC) inspections and investigations relative to the significant degradation of the reactor pressure vessel head identified at the FirstEnergy Nuclear Operating Company's (FENOC) Davis-Besse Nuclear Power Station in February and March 2002. Based upon the discovery of the reactor pressure vessel head degradation, the NRC issued Confirmatory Action Letter Number 3-02-001 to Davis-Besse documenting six commitments required to be accomplished prior to restarting of the reactor. The NRC also chartered an Augmented Inspection Team (AIT) inspection of the reactor pressure vessel degradation, the results of which were documented in Inspection Report No. 50-346/2002-03, issued on May 3, 2002. On October 2, 2002, the NRC issued the AIT Follow-up Special Inspection Report No. 50-346/2002-08, documenting ten apparent violations associated with the reactor pressure vessel degradation.

In a February 25, 2003, letter to FENOC, the NRC documented a performance deficiency associated with the control rod drive penetration cracking and reactor pressure vessel head degradation. The performance deficiency involved FENOC's failure to properly implement its boric acid corrosion control and corrective action programs, which allowed reactor coolant system pressure boundary leakage to occur undetected for a prolonged period of time, resulting in reactor pressure vessel head degradation. The NRC assessed the significance of the performance deficiency using the Significance Determination Process (SDP) and preliminarily concluded that the significance was in the RED range. A RED finding is one with high importance to safety that will result in increased NRC inspection and other NRC action. The NRC offered FENOC an opportunity to request a Regulatory Conference to discuss the preliminary significance determination. In lieu of a Regulatory Conference, FENOC submitted a written response, dated April 24, 2003, in which FENOC acknowledged the performance deficiency and did not contest the RED finding.

In a letter to FENOC, dated May 29, 2003, the NRC documented its conclusions that the significance of the performance deficiency, involving the control rod drive penetration cracking and the reactor pressure vessel head degradation, was appropriately characterized as RED. The NRC noted that the safety signficance of the performance deficiency was one of the inputs into the final characterization and resolution of the apparent violations described in the October 2002 AIT Follow-up Special Inspection Report. The NRC also noted that the results of an ongoing Office of Investigations (OI) investigation into the cause of the apparent violations would be a factor in the final enforcement deliberations. As a result, no Notice of Violation (Notice) was issued concurrent with the May 2003 letter.

Based upon its investigation into the causes of the apparent violations, OI determined that the apparent violations involved the licensee's willful failure to: (1) properly implement the boric acid control program; (2) properly implement the

corrective action program; (3) adequately remove, on several occasions, boric acid and rust deposits from the reactor pressure vessel head; (4) maintain the plant shutdown, i.e., not startup and return the plant to power from the Twelfth Refueling Outage (12RFO), until boric acid deposits were removed and the reactor pressure vessel head was inspected, and (5) maintain and submit to the NRC, complete and accurate information. As a result, the NRC referred the OI report to the U.S. Department of Justice (DOJ) for its review and consideration of criminal prosecution. While the DOJ's review is still ongoing, the NRC has determined that enforcement action should now be taken relative to the apparent violations documented in the AIT Follow-up Special Inspection and the OI Investigation Reports. The NRC does not anticipate taking further enforcement action in this matter, relative to FENOC, absent the DOJ developing new additional information.

Since the licensee's initial discovery of the reactor pressure vessel head degradation and the NRC's issuance of a Confirmatory Action Letter which outlined those actions necessary for the licensee to restart the plant, the NRC has provided extensive oversight of the licensee's evaluation of and corrective actions for the conditions which contributed to the reactor pressure vessel head degradation and the performance deficiency. In a March 8, 2004, letter, the NRC documented its determination that the matters contained in the NRC's Confirmatory Action Letter and Restart Checklist had been adequately resolved and that the NRC had reasonable assurance that the Davis-Besse Station could be restarted and operated safety. Therefore, the NRC has determined that the following results do not represent current licensee performance.

Based on information developed during the AIT Follow-up Special Inspection and OI Investigation, the NRC has determined that nine violations of NRC requirements occurred. The violations are cited in the enclosed Notice and are described in detail in the AIT Follow-up Special Inspection and the OI Investigation Reports. The NRC has determined that all of the violations were associated with the RED finding and the performance deficiency previously communicated to FENOC in our February and May 2003 letters.

Section I of the Notice documents five violations which were considered for civil penalties in accordance with the "General Statement of Policy and Procedures for NRC Enforcement Action," (Enforcement Policy), NUREG-1600. The NRC determined that these violations were of very high safety and regulatory significance because they clearly documented a pattern of willful violations of FENOC's boric acid corrosion control and corrective action programs over a protracted period of time, and a pattern of willful inaccurate or incomplete documentation of information that was required to be maintained or submitted to the NRC. As a direct result of these violations, the NRC determined that FENOC started up and operated the plant for at least the last operating cycle prior to the February 16, 2002, shutdown without: (1) fully understanding or characterizing the condition of the reactor pressure vessel head and the control rod drive penetrations; (2) determining the cause of significant boric acid buildup on the reactor pressure vessel head, the control rod drive penetrations, and several other components in the reactor containment building; (3) properly identifying the presence of ongoing reactor coolant system pressure boundary leakage and taking appropriate corrective actions, and; (4) identifying the very significant ongoing degradation of the reactor pressure vessel head which required a number of years to reach the level of material wastage observed in March 2002. Finally, the NRC determined that FENOC willfully provided incomplete and inaccurate information associated with its responses to the NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," which contributed to continued operation of the plant with ongoing reactor coolant system pressure boundary leakage and the significant degradation of the reactor pressure vessel head. As a result, a civil penalty in the amount of $5,450,000 is proposed as outlined in the following paragraphs and in the enclosed Notice.

Violation I.A of the enclosed Notice concerns a violation of Davis-Besse Technical Specification 3.4.6.2.a which prohibits plant operation in Modes 1 through 4 with any reactor coolant system leakage associated with the reactor coolant system pressure boundary. From at least May 18, 2000, to February 16, 2002, FENOC started up and operated the Davis-Besse Station in Modes 1 through 4 while being aware of the presence of significant boric acid deposits, on the reactor pressure vessel head, which were indicative of reactor coolant system leakage and which could not be justified as being caused by reactor coolant system non-pressure boundary leakage alone. The licensee conducted limited cleaning and inspection of the reactor pressure vessel head during the 12RFO in April-May 2000. However, the limited cleaning and inspection of the reactor pressure vessel head were not sufficient to ensure the integrity of the reactor coolant system pressure boundary.

The NRC determined that the licensee's failure to exercise adequate management oversight and controls, in its assessment of substantial recurring boric acid deposits on the reactor pressure vessel head during 12RFO and the build-up of boric acid deposits on other reactor containment equipment during plant operations, significantly contributed to the length of the Technical Specification violation and the significant reactor pressure vessel head degradation. The licensee's decision to return the unit to power on May 18, 2000, with ongoing reactor coolant system leakage, with significant boric acid deposits on the reactor pressure vessel head, which could not be associated with reactor coolant system non-pressure boundary leakage, and without conducting the reactor pressure vessel head cleaning and inspection required by the boric acid corrosion control procedure, is a serious safety and regulatory concern.

The seriousness of this safety and regulatory concern was exacerbated by FENOC's inaccurate and incomplete response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." The inaccurate and incomplete information provided by FENOC in its responses directly contributed to enabling FENOC to operate the plant beyond the Bulletin 2001-01 recommended shutdown date of December 31, 2001. Had the NRC known that the Davis-Besse Station was being operated with reactor coolant system pressure boundary leakage, the NRC would have taken immediate regulatory action to shut down the plant and to require the licensee to implement appropriate corrective actions. The startup and operation of the Davis-Besse Station, with reactor coolant system pressure boundary leakage, was a continuing violation of Davis-Besse Technical Specification 3.4.6.2.a.

This continuing Technical Specification (TS) violation is associated with a RED finding (EA-03-025) and was evaluated using the Significance Determination Process of the NRC Reactor Oversight Process. While a civil penalty is not usually considered for issues evaluated under the SDP, absent actual consequences (Section VI.C of the Enforcement Policy), the NRC considers a RED SDP finding to be of significant regulatory concern and may issue a civil penalty, up to the statutory maximum civil penalty, for such violations (Section VII.A of the Enforcement Policy).

In consultation with the Commission and because of the safety significance of the violation and the particularly poor performance of FENOC in this matter, the NRC is proposing, in accordance with Section VII.A of the Enforcement Policy, to issue a civil penalty for the Technical Specification violation associated with a RED finding evaluated under the SDP. In determining the proposed civil penalty, the NRC considered the safety significance of the violation, FENOC's multiple opportunities to identify and take corrective action for the violation, prior to and following restart of the plant in May 2000, and the economic benefit FENOC gained by operating the plant with reactor coolant system pressure boundary leakage between May 18, 2000, and February 16, 2002.

The statutory maximum civil penalty for the Technical Specification violation would be $110,000 per day for the period of time prior to and including November 2, 2000, and would be $120,000 per day for the period of time beginning on November 3, 2000, until the plant shut down on February 16, 2002. If the civil penalty for the Technical Specification violation was assessed for the entire operating cycle, the statutory maximum civil penalty would be approximately $75,000,000. However, the NRC's approach in assessing a civil penalty is not punitive, but focuses on deterrence to emphasize the importance of compliance with requirements and to encourage prompt identification and comprehensive corrective actions for violations. In determining the civil penalty, the NRC also noted that the licensee experienced substantial adverse economic impact resulting from the extended outage to replace the reactor vessel head and to make improvements necessary to address NRC requirements and concerns. Therefore, on balance, the NRC determined that a proposed civil penalty of $5,000,000 was appropriate for Violation I.A (EA-05-071).

Violation I.B of the enclosed Notice concerns FENOC willfully maintaining incomplete and inaccurate information in documents required to be maintained by the NRC. The documents indicated that accumulated boric acid deposits were removed from the reactor pressure vessel head and that the entire reactor pressure vessel head was inspected. However, the licensee did not clean or inspect the entire reactor pressure vessel head. The licensee's willful failure to accurately document the condition and cleanliness of the reactor pressure vessel head, including the willful failure to fully describe the accumulated boric acid deposits that remained on the head, is a significant violation that permitted uncorrected reactor coolant system pressure boundary leakage and boric acid corrosion of the reactor pressure vessel head to continue for an extended period of time. Had the NRC known of the reactor coolant system pressure boundary leakage, the NRC would have taken a different regulatory position, including the issuance of an Order. Therefore, this violation is categorized in accordance with the Enforcement Policy at Severity Level I.

In accordance with the Enforcement Policy, a base civil penalty of $110,000 was considered for a Severity Level I violation at the time of occurrence. Because the violation was willful and categorized at Severity Level I, the NRC considered whether credit was warranted for the civil penalty adjustment factors of Identification and Corrective Action. Credit was not warranted for Identification because the NRC identified the violation. Credit was not warranted for Corrective Action because significant intervention was required by the NRC to focus FENOC on the evaluative and corrective action process in order that comprehensive corrective action be taken. While credit was not warranted for the immediate corrective actions, the NRC recognized that your corrective actions ultimately were sufficient to permit restarting the facility. Since credit for Identification and Corrective Action was not warranted, the civil penalty assessment would normally be twice the base civil penalty for a Severity Level I violation or $220,000. However, Section VI.C.2.d of the Enforcement Policy limits the civil penalty to $110,000 per violation, per day. Therefore, a civil penalty of $110,000 is proposed for Violation I.B (EA-05-068).

Violation I.C of the enclosed Notice concerns FENOC willfully failing to ensure that a significant condition adverse to quality, associated with the presence of boric acid on the reactor pressure vessel head, at the end of 12 RFO, on May 18, 2000, was evaluated and corrected prior to restart of the plant. Specifically, the licensee closed at least three condition reports documenting the presence of significant boric acid deposits on the reactor pressure vessel head and associated components

without determining the cause of each condition, i.e., the source of the reactor coolant system leakage, without taking corrective action to address the immediate condition adverse to quality, i.e., the presence of significant deposits of boric acid on the reactor vessel head, and without taking corrective action to prevent recurrence. Therefore, this willful violation is categorized at Severity Level II in accordance with the Enforcement Policy.

In accordance with the Enforcement Policy, a base civil penalty of $88,000 was considered for a Severity Level II violation at the time of occurrence. Because the violation was willful and categorized at Severity Level II, the NRC considered whether credit was warranted for either of the civil penalty adjustment factors of Identification or Corrective Action. As discussed in Violation I.B above, credit was not warranted for either of the civil penalty adjustment factors because the NRC identified the violation, the licensee had multiple opportunities to identify the violation and failed to do so, and significant intervention by the NRC was necessary to focus the licensee on corrective actions and determining the root cause of the violation. Since credit was not warranted for the civil penalty adjustment factors, the civil penalty assessment would normally be twice the base civil penalty for a Severity Level II violation or $176,000. However, the civil penalty is reduced to the statutory maximum of $110,000 per violation (Section VI.C.2.d of the Enforcement Policy). Therefore, a civil penalty of $110,000 is proposed for Violation I.C (EA-05-066).

Violation I.D of the enclosed Notice documented that FENOC, at the end of 12RFO, willfully failed to fully implement the boric acid corrosion control procedure. Specifically, FENOC did not conduct a complete cleaning and inspection of the reactor pressure vessel head as required by the boric acid corrosion control procedure. In addition, FECNOC willfully deferred the implementation of a modification which was a corrective action for previous boric acid corrosion control program implementation non-conformances. As a result, FENOC willfully restarted the plant on May 18, 2000, and operated until February 16, 2002, with visible boric acid deposits on the reactor pressure vessel head and uncharacterized reactor coolant system pressure boundary leakage. Therefore, this willful violation is categorized at Severity Level II in accordance with the Enforcement Policy.

In accordance with the Enforcement Policy, a base civil penalty of $88,000 was considered for a Severity Level II violation at the time occurrence. Because the violation was willful and categorized at Severity Level II, the NRC considered whether credit was warranted for either of the civil penalty adjustment factors of Identification or Corrective Action. As discussed in Violation I.B above, credit was not warranted for either of the civil penalty adjustment factors because the NRC identified the violation, the licensee had multiple opportunities to identify the violation and failed to do so, and significant intervention by the NRC was necessary to focus the licensee on corrective actions and determining the root cause of the violation. Since credit was not warranted for the civil penalty adjustment factors, the civil penalty assessment would normally be twice the base civil penalty for a Severity Level II violation or $176,000. However, the civil penalty is reduced to the statutory maximum of $110,000 per violation (Section VI.C.2.d of the Enforcement Policy). Therefore, a civil penalty of $110,000 is proposed for Violation I.D (EA-05-067).

Violation I.E of the enclosed Notice concerns FENOC willfully providing incomplete and inaccurate information in two responses to the NRC relative to NRC Bulletin 2001-01. The incomplete and inaccurate information was significant because the NRC relied, in part, on the information to assess the adequacy of FENOC's previous implementation of those quality assurance and management controls necessary to ensure a complete understanding of the physical condition of the reactor pressure vessel head, the control rod drive penetrations, and the absence of reactor coolant system pressure boundary leakage. Had the NRC known of the reactor coolant system pressure boundary leakage, the NRC would have taken a different regulatory position, including the issuance of an Order. Therefore, this willful violation is categorized at Severity Level I in accordance with the Enforcement Policy.

In accordance with the Enforcement Policy, a base civil penalty of $120,000 was considered for a Severity Level I violation at the time occurrence. The NRC considered whether credit was warranted for the for either of the civil penalty adjustment factors of Identification or Corrective Action. As discussed in Violation I.B above, credit was not warranted for either of the civil penalty adjustment factors because the NRC identified the violation, the licensee had multiple opportunities to identify the violation and failed to do so, and significant intervention by the NRC was necessary to focus the licensee on corrective actions and determining the root cause of the violation. Since credit was not warranted for the civil penalty adjustment factors, the civil penalty assessment would normally be twice the base civil penalty for a Severity Level I violation or $240,000. However, the civil penalty is reduced to the statutory maximum of $120,000 per violation (Section VI.C.2.d of the Enforcement Policy). Therefore, a civil penalty of $120,000 is proposed for Violation I.E (EA-05-072).

Section II of the enclosed Notice describes other violations of NRC requirements that are associated with the previously issued RED SDP finding (EA-03-025). Also described in Section II of the enclosed Notice are two non-willful violations of 10 CFR 50.9, "Completeness and Accuracy of Information" (Severity Level III violation without civil penalty (EA-04-069) and a Severity Level IV violation). The NRC is exercising discretionary authority under Section VII.B.6 of the Enforcement Policy and is not proposing a civil penalty be issued for the other violations associated with a RED SDP finding and the Severity

Level III violation of 10 CFR 50.9, in part, because of the significant civil penalty proposed in Section I of the Notice.

Therefore, to emphasize the very high safety and regulatory significance of compliance with TSs, FENOC's willful failure to effectively implement its boric acid corrosion control and corrective action programs, and FENOC's willful failure to maintain and provide to the NRC complete and accurate information, and in consultation with the Commission, I am issuing the enclosed Notice with a cumulative civil penalty of $5,450,000.

You are required to respond to this letter within 90 days and should follow the instructions specified in the enclosed Notice when preparing your response. However, since the NRC enforcement action is being proposed prior to any final action by the U.S. Department of Justice, consideration may be given to extending the response time for good cause shown.

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), 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. 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.
Questions concerning this matter should be addressed to Mr. Steven Reynolds, the NRC Senior Manager responsible for the NRC's Manual Chapter 0350 oversight activities associated with the Davis-Besse Station. Mr. Reynolds may be reached at (630) 829-9601.

Sincerely,

/RA/

Ellis W. Merschoff
Deputy Executive Director for Reactor Programs
Office of the Executive Director for Operations

Enclosures:
1. Notice of Violation and Proposed Imposition of Civil Penalties
2. NUREG/BR-0254 Payment Methods (Licensee Only)

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

cc w/encl: The Honorable Dennis Kucinich
M. Bezilla, Vice President, Davis-Besse
J. Hagan, Senior Vice President
Engineering and Services, FENOC
L. Myers, Chief Operating Officer, FENOC
Plant Manager
Manager - Regulatory Compliance
D. Jenkins, 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
J. Papcun, President, Ottawa County Board of Commissioners

NOTICE OF VIOLATION
AND
PROPOSED IMPOSITION OF CIVIL PENALTIES

FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station

Docket No. 50-346
License No. NPF-3
EA-03-025; EA-05-066; EA-05-067;
EA-05-068; EA-05-069; EA-05-070;
EA-05-071; EA-05-072

During an NRC inspection conducted from May 15 to August 9, 2002, and an NRC investigation completed on August 22, 2003, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the NRC proposes to impose civil penalties pursuant to Section 234 of the Atomic Energy Act of 1954, as amended (Act), 42 U.S.C. 2282, and 10 CFR 2.205. The particular violations and associated civil penalties are set forth below:

I. Violations Assessed a Civil Penalty

A. Technical Specification 3.4.6.2.a, Amendment 220, dated April 14, 1998, requires, in part, that the licensee shall limit reactor coolant system leakage to "No PRESSURE BOUNDARY LEAKAGE" during Modes 1 through 4.

Contrary to the above, between May 18, 2000, and February 16, 2002, the licensee started up and operated the plant in Modes 1 through 4 with reactor coolant system pressure boundary leakage, i.e. control rod drive penetration leakage. Specifically, the licensee returned the plant to operation following the 2000 refueling outages without fully characterizing and eliminating reactor coolant system pressure boundary leakage on the reactor pressure vessel head as evidenced by significant boric acid deposits on the reactor pressure vessel head at the start and end of the outage and by the development of new and extensive boric acid deposits on reactor containment equipment during the operation cycle.

This is a violation associated with a RED SDP finding.
Civil Penalty - $5,000,000 (EA-05-071)

B. 10 CFR 50.9 requires that information provided to the Commission by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the licensee shall be complete and accurate in all material respects.

10 CFR 50, Appendix B, Criterion XVI requires, in part, that for significant conditions adverse to quality, the cause of the condition and the corrective actions taken to preclude repetition shall be documented.

10 CFR 50, Appendix B, Criterion XVII, requires, in part, that the licensee shall maintain sufficient records to furnish evidence of activities affecting quality and that those records shall include monitoring of work performance.

Condition Report (CR) 2000-1037, closed May 1, 2000, documented corrective actions for the presence of boric acid on the reactor pressure vessel head, a significant condition adverse to quality, that included: "Accumulated boron deposited between the reactor head and the thermal insulation was removed during the cleaning process performed under W.O. [Work Order] 00-001846-000. No boric acid induced damage to the head surface was noted during the subsequent inspection."

Work Order 00-001846-000, "Clean Boron Accumulation from Top of Reactor Head and Top of Insulation," dated April 25, 2000, required the licensee staff to "clean boron accumulation from top of reactor head and on

top of insulation." The Work Order Log, included as Page Four of the completed Work Order, documented that the, "work [was] performed without deviation" and was signed by the System Engineer on April 25, 2000.

Contrary to the above,

1. The information included in CR 2000-1037 relative to the completed corrective actions and the subsequent inspection results were not complete and accurate in all material respects. Specifically, the licensee did not remove the accumulated boron deposits from all areas between the reactor head and the thermal insulation and did not conduct subsequent inspections of the entire reactor head. Instead, the licensee removed accumulated boric acid deposits from a portion of the reactor vessel head and conducted subsequent inspections for those portions of the reactor vessel head where the boric acid deposits had been removed.

2. The Work Order Log, included as Page Four of completed Work Order 00-001846-000, a record required by Commission regulations to furnish evidence of activities affecting quality, contained information that was not accurate in all material respects. Specifically, the Work Order Log indicated that boron accumulation was cleaned from the top of the reactor head and on top of the insulation, without deviation, when, in fact, boric acid deposits were left on the head after the cleaning was completed on April 25, 2000.

This is a Severity Level I violation (Supplement VII).
Civil Penalty $110,000 (EA-05-068)

C. 10 CFR 50, Appendix B, Criterion XVI, requires, in part, that licensees shall establish measures to ensure that conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. For significant conditions adverse to quality, the licensee shall establish measures to ensure that the cause of the condition is determined and that corrective actions are taken to preclude repetition.

Plant Procedure NG-NA-00702, "Corrective Action Program," Revision 3, defined a significant condition adverse to quality to be a condition, which, if left uncorrected, could have an undesirable effect on plant safety, personal safety, regulatory position, financial liability, or environmental impact.

Contrary to the above, the licensee did not determine the cause of the condition and did not implement corrective actions to preclude repetition of the condition associated with the identification and removal of boric acid on the reactor vessel head, a significant condition adverse to quality, prior to closing the associated condition reports.

Specifically:

1. On April 27, 2000, the licensee closed CR 2000-0781, "Leakage from CRD [Control Rod Drive] Structure Blocked Visual Exam of Reactor Vessel Head Studs," issued on April 6, 2000, associated with the accumulation of boric acid deposits on the reactor vessel head studs without determining the cause of the deposits, i.e., identifying the source of the reactor coolant system leakage, and without taking corrective actions to preclude recurrence.

2. On April 27, 2000, the licensee closed CR 2000-0782, "Inspection of Reactor Flange Indicated Boric Acid Leakage From Weep Holes," issued on April 6, 2000, associated with the accumulation of boric acid deposits on the reactor vessel head, without determining the cause of the boric acid deposits, i.e., identifying the source of the reactor coolant system leakage, without removing all of the known boric acid deposits on the reactor pressure vessel head, and without taking corrective actions to prevent recurrence.

3. On May 1, 2000, the licensee closed CR 2000-1037, "Inspection of Reactor Head Indicated Accumulation of Boron in Area of the CRD [Control Rod Drive] Nozzle Penetration," issued on April 17, 2000, associated with the accumulation of boric acid deposits on the reactor vessel head, without determining the cause of the boric acid deposits, i.e., identifying the source of the reactor coolant system leakage, without removing all of the known boric acid deposits on the reactor vessel head, and without taking corrective actions to prevent recurrence.

This is a Severity Level II violation (Supplement I)
Civil Penalty - $110,000 (EA-05-066)

D. 10 CFR 50, Appendix B, Criterion V, requires, in part, that activities affecting quality be accomplished in accordance with written procedures.

Davis-Besse Station Procedure NG-EN-00324, "Boric Acid Corrosion Control Program," Revisions 1/C1 and 2, Step 6.3.1, required, in part, that an initial inspection of boric acid buildup shall be performed to determine the "as found" conditions and to document the inspection results. The procedure also required, in Attachment 3, that insulation and other hindrances to direct visual [inspection] be removed as needed to allow detailed inspections of components suspected of leakage.

Potential Condition Adverse to Quality (PCAQ) 96-0551, initiated on April 21, 1996, documented the licensee's inability to comply with some inspections of the reactor pressure vessel head, as required by Procedure NG-EN-00324, and an inability to accurately determine the reactor pressure vessel head "as found" conditions, associated with boric acid deposits on the reactor pressure vessel head, due to the restrictions resulting from the location and size of the inspection ports, "mouse holes." The PCAQ further documented that only 50 to 60 percent of the reactor pressure vessel head could be inspected using the current inspection ports.

Modification 94-0025, initiated on May 27, 1994, and referenced as corrective action for PCAQ 96-0551, directed the completion of modifications to the reactor pressure vessel head service structure inspection ports to permit the inspection and cleaning of 100 percent of the reactor vessel head in accordance with Procedure NG-EN-00324.

Contrary to the above, on May 18, 2000, and at the end of Refueling Outage 12, the licensee failed to remove obstructions, including boric acid deposit buildups, necessary to conduct a detailed inspection of the reactor pressure vessel head and other components that may be suspected of leakage, as required by Plant Procedure NG-EN-00324, "Boric Acid Corrosion Control Program." The licensee's ability to conduct the inspections was significantly limited as a result of its concurrent deferral of the installation of Modification 94-0025, a corrective action for a significant condition adverse to quality documented in PCAQ 96-0551 and associated with the licensee's failure during previous outages to conduct complete inspections and cleaning of boric acid deposits on the reactor pressure vessel head.

This is a Severity Level II violation (Supplement I)
Civil Penalty $110,000 (EA-05-067)

E. 10 CFR 50.9 requires that information provided to the Commission by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the licensee shall be complete and accurate in all material respects.

NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," required all holders of operating licenses for pressurized water nuclear power reactors to provide information related to the structural integrity of the reactor vessel head penetration (VHP) nozzles for their respective facilities, including the extent of VHP nozzle leakage and cracking that has been found to date, the inspections and repairs that have been undertaken to satisfy applicable regulatory requirements, and the basis for concluding that their plans for future inspections will ensure compliance with applicable regulatory requirements.

Contrary to the above, the licensee, a holder of an operating license for a pressurized water nuclear power reactor, the Davis-Besse Station, provided the Commission responses to Bulletin 2001-01 which included materially inaccurate and incomplete information as follows:

1. In a September 4, 2001, response to the Bulletin entitled, "Response to Bulletin 2001-01," Serial 2731, the licensee made the following four materially inaccurate and incomplete statements:

   (a) The licensee's response to Bulletin Item 1.c, on page 2 of 19, stated: "the minimum gap being at the dome center of the RPV [reactor pressure vessel] head where it is approximately 2 inches, and does not impede a qualified visual inspection."

   The licensee's response was materially inaccurate, in that, the statement contradicted statements in the licensee's documents identified as PCAQR 94-0295 and 96-0551, which clearly stated that

inspection capability at the top of the reactor vessel head was limited. The limitation was stated to be caused by the restricted access to the area through the service structure "weep holes", the curvature of the reactor pressure vessel head, and by the limited space to manipulate a camera due to the insulation that creates the two inch gap.

(b) The licensee's response to Bulletin Item 1.d, which requested inclusion of a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the reactor pressure vessel head for visual examinations, did not include a description of any limitations.

The licensee's response was materially incomplete in that the response did not mention that accessibility to the bare metal of the reactor pressure vessel head was impeded, during the Eleventh (1998) and the Twelfth (2000) Refueling Outages, by the presence of significant accumulations of boric acid deposits.

(c) The licensee's response to Bulletin Item 1.d, which also requested a discussion of the findings of reactor pressure vessel head inspections, stated that for the Twelfth Refueling Outage (2000), the inspection of the reactor pressure vessel head/nozzles indicated some accumulation of boric acid deposits.

The licensee's response was materially incomplete and inaccurate in that it mischaracterized the accumulation of boric acid on the reactor pressure vessel head and did not mention the evidence of corrosion that was evidenced by the pictures and the video examination of reactor pressure vessel head conditions documented at the beginning and ending of the Twelfth Refueling Outage (2000).

(d) The licensee's response to the Bulletin, on Page 3, stated: "The boric acid deposits were located beneath the leaking flanges with clear evidence of downward flow. No visible evidence of nozzle leakage was detected."

The licensee's response was materially inaccurate in that the boric acid deposits were not all located under leaking flanges and the licensee lacked clear evidence of the absence of downward flow for all nozzles. Specifically, the presence of boric acid deposits was not limited only to the areas beneath the flanges, as implied by that statement. The build-up of boric acid deposits was so significant that the licensee could not inspect all of the nozzles. As a result, the licensee also did not have a basis for stating that no visible evidence of nozzle leakage was detected.

2. In an October 17, 2001, response to the Bulletin entitled, "Supplemental Response to Bulletin 2001-01," Serial 2735, the licensee stated: "In May 1996, during a refueling outage, the RPV [reactor pressure vessel] head was inspected. No leakage was identified, and these results have been recently verified by a re-review of the video tapes obtained from that inspection. The RPV head was mechanically cleaned at the end of the outage. Subsequent inspections of the RPV head in the next two refueling outages (1998 and 2000), also did not identify any leakage in the CRDM [control rod drive mechanism] nozzle-to-head areas that could be inspected. Video tapes taken during these inspections have also been re-reviewed."

The licensee's response was materially inaccurate, in that: (1) each reactor pressure vessel head control rod drive penetration was not inspected in May 1996, as documented in PCAQR 96-0551, and; (2) the reactor pressure vessel head, including the area around each control rod drive penetration, was not completely cleaned, as noted in PCAQR 98-0649, which was prepared at the start of the Eleventh Refueling Outage (1998), which stated that there were old boric acid deposits on the head.

This is a Severity Level I violation (Supplement VII)
Civil Penalty $120,000 (EA-05-072)

II. Violations Not Assessed a Civil Penalty

A. 10 CFR Part 50, Appendix B, Criterion XVI, requires, in part, that the licensee shall establish measures to ensure that conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. Criterion XVI also requires that for significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and that corrective actions are taken to preclude repetition.

Plant Procedure NG-NA-00702, "Corrective Action Program," Revision 3, defined a significant condition adverse to quality to be a condition adverse to quality, which, if left uncorrected, could have an undesirable effect on plant safety, personal safety, regulatory position, financial liability, or environmental impact.

Contrary to the above, the licensee failed to determine the root cause of and take corrective actions to preclude the repetition of:

1. Fouling of containment air cooling fins by boric acid, between June 2000 and February 16, 2002, a significant condition adverse to quality, as documented in:

   Condition Report (CR) 2000-1547, "CAC [containment air cooler] Plenum Pressure Drop Following 12 RFO," dated June 2, 2000;

   CR 2000-4138, "Frequency for Cleaning Boron From CAC Fins Increased to Interval of Approximately 8 Weeks," dated December 21, 2000, and;

   CR 2001-0039, "CAC Plenum Pressure Experienced Step Drop," dated January 4, 2001.

2. Fouling of the containment radiation elements by boric acid and iron oxide, between April 2001, and February 16, 2002, a significant condition adverse to quality, as documented in:

   Condition Report (CR) 99-1300, "Analysis of CTMT [containment] Radiation Monitor Filters.' dated May 13, 1999;

   CR 2001-1110, "Chemistry is Changing Filters on RE4597BA More Frequently," dated April 23, 2001;

   CR 2001-1822, "Frequency of Filter Changes for RE4597BA is Increasing," dated July 23, 2001;

   CR 2001-2795, "RE4597BA Alarmed on Saturation," dated October 22, 2001, and;

   CR 2001-3411, "Received Equipment Fail Alarm for Detector Saturation on RE4597BA," dated December 18, 2001.

3. An increasing trend in unidentified reactor coolant system leakage, between March 2001, and December 2001, a significant condition adverse to quality, as documented in:

   Condition Report (CR) 2001-0890, "Unidentified RCS [reactor coolant system] Leak Rate Varies Daily by as Much as 100 percent of the Value," March 29, 2001;

   CR 2001-1857, "RCS Unidentified Leakage at .125 to .145 gpm [gallons per minute]," July 25, 2001;

   CR 2001-2862, "Calculated Unidentified Leakage for Reactor Coolant System has Indicated Increasing Trend," October 22, 2001, and;

   CR 2001-3025, "Increase in RCS Unidentified Leakage," November 12, 2001.

   This is a violation associated with a RED SDP finding (EA-03-025).

B. 10 CFR Part 50, Appendix B, Criterion V, requires, in part, that 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.

Procedure NG-EN-00324, "Boric Acid Corrosion Control Program," Revisions 0 through 2 (effective date October 1, 1999), were classified as a procedure affecting quality under the licensee's administrative system.

Contrary to the above, between October 1, 1999, and March 6, 2002, Procedure NG-EN-00324, "Boric Acid Corrosion Control Program," Revisions 0 through 2, were not appropriate to the circumstances and contributed to the licensee's failure to detect and address boric acid corrosion of the reactor vessel head, as follows:

1. The procedure inappropriately focused on bolted and flanged connections in the definition of leakage (Sections 4.2 though 4.4), the definition of reactor coolant system pressure boundary components (Section 4.9), and the identification of investigation locations (Section 6.1) at the expense of identifying the potential for through-wall leakage.

2. The procedure did not include adequate guidance, specifications, or threshold levels for initiating a "detailed inspection" in order to ensure consistent implementation of Section 6.3.4 of the procedure.

3. The procedure did not require the identification of and corrective actions to preclude the repetition of boric acid leaks, a significant condition adverse to quality, but instead only required the preparation of a repair tag or work order to facilitate repair of the leak.

4. The procedure did not define the qualifications and training necessary to permit engineering staff to conduct inspections and evaluations in a consistent manner, including the use of proper inspection techniques, observations, recording of results, and evaluations.

5. The procedure inappropriately exempted stainless steel or Inconel components from further examination related to boric acid corrosion, unless the examination was during an ASME Section XI test which might require a bolting examination.

6. The procedure inappropriately did not require the licensee staff to maintain records necessary to demonstrate the proper completion of activities affecting quality.

This is a violation associated with a RED SDP finding (EA-03-025).

C. 10 CFR 50.9 requires that information provided to the Commission by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the licensee shall be complete and accurate in all material respects.

10 CFR Part 50, Appendix B, Criterion XVII, requires, in part, that the licensee shall maintain sufficient records to furnish evidence of activities affecting quality and that those records shall include actions taken to correct any deficient conditions.

Contrary to the above, the following information was not complete or accurate in all material respects for documents required to be maintained or provided to the Commission:

1. Potential Condition Adverse to Quality Report (PCAQR) 98-0649, dated April 18, 1998, contained the following closure statement: "Accumulation of boric acid on the reactor vessel caused by leaking CRDMs [control rod drive mechanisms] has not resulted in any boric acid corrosion. This was identified through inspections following reactor vessel head cleaning in past outages....Additionally, B&W [Babcock & Wilcox] documentation discussing CRDM nozzle cracking further stated that boric acid deposits on the head caused by leaking CRDM flanges would not result in head corrosion." However, the quoted statements were not accurate in all material respects in that the licensee had previously not cleaned all areas of the reactor head of boric acid deposits, had not inspected the base metal under all the deposits to determine whether corrosion was present, and no B&W documentation was available to support the claim that boric acid would not result in head corrosion.

2. Potential Condition Adverse to Quality Report (PCAQR) 98-0767, dated April 25, 1998, Section 4A, Item F, included the following closure justification, "The boric acid deposits were removed from the head." However, the quoted statement was not accurate in all material respects in that the licensee had not removed all of the boric acid deposits from the head as of the end of the eleventh refueling outage.

This is a Severity Level III violation (Supplement VII) (EA-05-069)

D. 10 CFR 50.9 requires that information provided to the Commission by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the licensee shall be complete and accurate in all material respects.

10 CFR Part 50, Appendix B, Criterion XVII, requires, in part, that the licensee shall maintain sufficient records

to furnish evidence of activities affecting quality and that those records shall include audits and those actions taken to correct any deficient conditions.

Contrary to the above, the following information was not complete or accurate in all material respects for documents required to be maintained or provided to the Commission:

1. On September 23, 1993, the licensee processed a "Document Void Request" to cancel Modification 90-012 which stated, "Current inspection techniques using high-powered cameras preclude the need for inspection ports, additionally, cleaning of the reactor vessel head during last three outages was completed successfully without requiring access ports." However, the quoted statement was not accurate in all material respects, in that, the licensee left boric acid deposits on the reactor vessel head at the end of both the seventh and eighth refueling outages, the two outages preceding this statement.

2. Quality Assurance Audit Report AR-00-OUTAG-01, dated July 7, 2000, stated, in part, "Boric Acid Corrosion Control Checklists and Condition Reports were initiated by inspectors when prudent to document and evaluate boric acid accumulation and leaks. Boric acid leakage was adequately classified and corrected when appropriate. Engineering displayed noteworthy persistence in ensuring boric acid accumulation from the reactor head was thoroughly cleaned." However, the audit report was not accurate in all material respects in that the licensee did not: 1) thoroughly clean the reactor head during the outage; 2) did not prepare a boric acid corrosion control checklist for the boric acid left on the head after the cleaning attempt; and 3) identify, properly classify, or correct the boric acid accumulation and leaks.

This is a Severity Level IV violation (Supplement VII) (EA-05-070)

Pursuant to the provisions of 10 CFR 2.201, FirstEnergy Nuclear Operating Company (Licensee) is hereby required to submit a written statement or explanation to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, in response to this Notice of Violation and Proposed Imposition of Civil Penalties (Notice) within 90 days of the date of this letter. However, since this enforcement action is being proposed prior to any final action by the U.S. Department of Justice, consideration may be given to extending the response time for good cause shown. The reply should be clearly marked as a "Reply to a Notice of Violation: EA-03-025; EA-05-066; EA-05-067; EA-05-068; EA-05-069; EA-05-070; EA-05-071 and EA-05-072" and should include for each alleged violation: (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 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. Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Within the same time as provided for the response required above under 10 CFR 2.201, the Licensee may pay the civil penalties proposed above or the cumulative amount of the civil penalties, if more than one civil penalty is proposed, in accordance with NUREG/BR-0254 and by submitting to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555, a statement indicating when and by what method payment was made, or may protest imposition of the civil penalties in whole or in part, by a written answer addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission. Should the Licensee fail to answer within the time specified, an order imposing the civil penalties will be issued. Should the Licensee elect to file an answer in accordance with 10 CFR 2.205 protesting the civil penalties, in whole or in part, such answer should be clearly marked as an "Answer to a Notice of Violation" and may: (1) deny the violations listed in this Notice, in whole or in part, (2) demonstrate extenuating circumstances, (3) show error in this Notice, or (4) show other reasons why the penalties should not be imposed. In addition to protesting the civil penalties, in whole or in part, such answer may request remission or mitigation of the penalties.

In requesting mitigation of the proposed penalties, the factors addressed in Section VI.C.2 of the Enforcement Policy should be addressed. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanation in reply pursuant to 10 CFR 2.201, but may incorporate parts of the 10 CFR 2.201 reply by specific reference (e.g., citing page and paragraph numbers) to avoid repetition. The attention of the Licensee is directed to the other provisions of 10 CFR 2.205, regarding the procedure for imposing civil penalties.

Upon failure to pay any civil penalties due which subsequently has been determined in accordance with the applicable

provisions of 10 CFR 2.205, this matter may be referred to the Attorney General, and the penalties, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234(c) of the Act, 42 U.S.C. 2282c.

The response noted above (Reply to Notice of Violation, statement as to payment of civil penalties, and Answer to a Notice of Violation) should be addressed to: Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852-2738, with a copy to the Regional Administrator and Enforcement Officer, U.S. Nuclear Regulatory Commission, Region III, and a copy to the NRC Resident Inspector at the Davis-Besse Nuclear Power Station.

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 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

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

Dated this 21st day of April 2005

EA-05-051 - Palo Verde (Arizona Public Service Company)

April 8, 2005

EA-05-051

Gregg R. Overbeck, Senior Vice
President, Nuclear
Arizona Public Service Company
P.O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: NOTICE OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTY - $50,000 (NRC SPECIAL INSPECTION REPORT 2004-014, PALO VERDE NUCLEAR GENERATING STATION)

Dear Mr. Overbeck:

The NRC's January 5, 2005, inspection report described the results of a special inspection that followed up on your discovery in July 2004 that a significant section of containment sump safety injection piping at all three Palo Verde Nuclear Generating Station (PVNGS) units was void of water. The report discussed two findings that were being evaluated for further NRC action under the NRC's Significance Determination Process or NRC Enforcement Policy. This letter provides you the final results of our enforcement deliberations on one of the findings, the apparent violation of 10 CFR 50.59 identified in the special inspection report, and addresses your denial of one example of a violation of 10 CFR 50.59 documented as a non-cited violation (NCV) in the same report. In separate correspondence, we are providing you our evaluation and final significance determination for the second finding, a design control finding that was processed under the NRC's Significance Determination Process.

The apparent violation of 10 CFR 50.59 [1992 version] involved making a change in June 1992 to Surveillance Procedure 41ST-1SI09, "ECCS (emergency core cooling system) Leak Test," which drained, and left empty, a portion of the containment sump safety injection recirculation piping. As described in the inspection report, the apparent violation was based on the NRC's preliminary conclusion that the change modified the facility as described in the Updated Final Safety Analysis Report (UFSAR). The UFSAR states, in part, that safety injection piping will be filled with water. This procedural change also affected the available net positive suction head analysis described in the UFSAR for the containment spray and high pressure safety injection pumps, which assumed that pump suction piping would be filled with water.

NRC's evaluation of this apparent violation considered the fact that Arizona Public Service Company (APS) discovered this condition at PVNGS in July 2004, following notification from another facility where a similar problem had been identified. On July 31, 2004, APS reported this condition to NRC under the provisions of 10 CFR 50.72(b)(3)(v), noting that the voided section of piping had the potential to prevent the fulfillment of the safety function to remove residual heat and mitigate the consequences of a loss-of-coolant accident. In early August, Palo Verde took corrective action to fill the voided piping in all three units, completing those actions by August 4, 2004.

At your request, a Regulatory and Predecisional Enforcement Conference was held on February 17, 2005, to provide APS an opportunity to provide its perspective on this apparent violation before NRC made a final enforcement decision. At the conference, the APS staff contested the apparent violation of 10 CFR 50.59 described above, claiming that the procedural change made in 1992 did not change the facility as described in the UFSAR because the design requirements described in the UFSAR had never been implemented. The APS staff also contended that the change to the surveillance test procedure would not be expected to be evaluated under 10 CFR 50.59 because it resulted in returning the affected section of piping to

its as-found condition by removing demineralized water that was used to perform the ECCS leak test. Following the conference, you submitted letters dated February 24 and 28, 2005, which summarized your views on this and other issues discussed in the inspection report.

Rather than viewing this as a violation of 10 CFR 50.59, the APS staff indicated that the change to the procedure represented a missed opportunity to identify and correct design and licensing basis deficiencies that had existed since plant startup. At the conference and in your February 24, 2005, letter, you made the following points:

1) No design output document was found that specified a design requirement to maintain the emergency core cooling system (ECCS) sump suction lines filled, and no procedural requirement was established to maintain the ECCS sump suction piping filled for system operability purposes.

2) Prior to the 1992 procedural change, the section of piping that was left filled with water at the conclusion of the ECCS leak test was drained of water during quarterly valve stroke testing, supporting the view that system design requirements were not recognized. Interviews of operations personnel indicated a general understanding that the ECCS sump suction lines were maintained empty.

3) The UFSAR actually states "To minimize the potential for water hammer, the safety injection piping will be maintained filled with water." Water hammer is a discharge piping phenomenon.

4) The net positive suction head calculations referenced in the inspection report refer to the expected conditions in the system after fully developed flow from the sump is established. No statements in the calculations define the initial conditions.

Based on the information developed during the special inspection, and the information that APS provided during and subsequent to the conference, the NRC has determined that the failure to perform a safety evaluation and receive prior NRC approval of the change to Procedure 41ST-1SI09 was a violation of 10 CFR 50.59. We concluded that the procedure change should have been subjected to an evaluation in accordance with the requirements of 10 CFR 50.59 in existence in 1992, and that had an evaluation been performed, it would have led to a conclusion that the change was contrary to the description of the facility in the UFSAR and that it created an unreviewed safety question. The primary bases for the NRC's decision follow:

1) Whether or not a specific procedure existed to implement the design basis, the fact is that prior to 1992, the ECCS leak-test procedure resulted in leaving the affected section of piping filled with water, which was consistent with the description of the system in the UFSAR.

2) The 1992 change to Procedure 41ST-1SI09 resulted in intentionally draining the system piping, a change to the original procedure and a change to the facility as described in the UFSAR.

3) The 1992 procedure change increased the probability of a malfunction of equipment important to safety because it had the potential to affect the ability of the containment spray and high pressure safety injection pumps to perform their intended safety function in the containment sump recirculation mode.

We also considered your statements pertaining to the reference to "water hammer" considerations in the UFSAR, net positive suction head requirements in the UFSAR, and plant staff's knowledge of the design basis requirements for this section of piping. None of these issues affected our determination that a violation of 10 CFR 50.59 occurred. With regard to "water hammer," we conclude that voided conditions in pump suction piping may lead to a water hammer condition in discharge piping and that it is unreasonable to assume that UFSAR statements regarding the filling of piping with water is limited to the discharge side of the pumps. With regard to net positive suction head calculations, we conclude that such calculations are completed to evaluate the range of operating configurations of safety related equipment, including transition periods between suction sources. In this case, the net positive suction head calculations assumed the system was water filled. With regard to plant staff's knowledge of the design basis, we note that plant records of the 1992 procedure change indicate that a shift manager questioned the change because of the potential impact voided suction piping would have on the operability of the ECCS systems. Also, the APS staff acknowledged during the conference that some of the engineering staff understood that the system was to be maintained in a water filled condition.

The NRC also evaluated this violation against the current 10 CFR 50.59 requirements, because NRC policy is to exercise discretion for violations of 10 CFR 50.59 that predate the current rule if the involved circumstances do not indicate that the

current rule would have been violated. The current rule states that a licensee shall obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system or component important to safety previously evaluated in the final safety analysis report (as updated). We conclude that the 1992 change also would have violated the current rule, because changes to Surveillance Procedure 41ST-1SI09, "ECCS Leak Test" resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system or component important to safety.

Based on the analysis described above, the 1992 violation of 10 CFR 50.59 is cited in the enclosed Notice of Violation (Notice), and the circumstances surrounding it were described in more detail in the subject inspection report. The safety significance of this violation is based largely on the risk significance of the associated change to the facility, as discussed in Supplement I of the NRC Enforcement Policy. The policy provides an example of a Severity Level III violation as, "A failure to obtain prior Commission approval required by 10 CFR 50.59 for a change, in which the consequence of the change, is evaluated as having low to moderate, or greater safety significance (i.e., white, yellow, or red) by the SDP." Because the risk significance of the related design control violation was determined to be Yellow, this violation has been classified at Severity Level III, in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600.

In accordance with the Enforcement Policy in existence in 1992, a base civil penalty in the amount of $50,000 is considered for a Severity Level III violation. Because the Palo Verde facility has not been the subject of escalated enforcement action under the Enforcement Policy within the last 2 years, the NRC considered only whether credit was warranted for Corrective Action in accordance with the civil penalty assessment process in Section VI.C.2 of the Enforcement Policy. While you took prompt action to fill the affected portions of the ECCS piping with water following the discovery of this condition in July 2004, and have taken actions to address other deficiencies in your program for implementing 10 CFR 50.59 at your facility, it is not apparent that you have taken actions to address the specific causes or prevent similar violations from occurring. For example, you have not assessed how "non-intent" changes to maintenance and test procedures were screened to assure they do not affect the facility design and may therefore require a 10 CFR 50.59 evaluation, and you have not assessed whether similar changes in the past should have been evaluated under the 10 CFR 50.59 process. APS has not taken corrective actions in response to this violation consistent with NRC requirements for "significant conditions adverse to quality" in 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action. The actions you have taken to address 10 CFR 50.59 problems would not, in our view, preclude the repetition of similar implementation problems. Thus, the NRC has determined that Corrective Action credit is not warranted, resulting in an assessment of a civil penalty at the base value, or $50,000.

Therefore, to emphasize the importance of evaluating changes to the facility that may impact safety, and the importance of taking corrective actions that are comprehensive in correcting significant noncompliances, I have been authorized, after consultation with the Director, Office of Enforcement, to issue the enclosed Notice of Violation and Proposed Imposition of Civil Penalty (Notice) in the base amount of $50,000.

At the conference, the APS staff also contested one example of a violation of 10 CFR 50.59 that was documented in the same inspection report as an NCV. The example involved leaving a 10-20 cubic foot section of piping voided after you identified this condition in July 2004. The NRC's determination in the inspection report was that the decision to leave a portion of piping voided should have been subjected to a 10 CFR 50.59 evaluation. At the conference and in your February 24, 2005 letter, you stated that this was not the final corrective action for this condition, that the final corrective action was to fill the piping with borated water, and that a 10 CFR 50.59 evaluation was not required. The APS staff supported the APS position with references to NRC and industry guidance documents. After evaluating the information in the inspection report, and the information that you provided during and after the conference, the NRC has determined to withdraw this example of the non-cited violation of 10 CFR 50.59 documented in the report.

You are required to respond to the Notice 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 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. 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//

Bruce S. Mallett
Regional Administrator

Docket Nos. 50-528; 50-529; 50-530
License Nos. NPF-41; NPF-51; NPF-74

Enclosures:
1. Notice of Violation and Proposed Imposition of Civil Penalty, EA-05-051
2. NUREG/BR-0254 Payment Methods (Licensee only)

cc w/Enclosure 1 only:

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Austin, TX 78701-3326

John W. Schumann
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Southern California Public Power Authority
P.O. Box 51111, Room 1255-C
Los Angeles, CA 90051-0100

Enclosure 1

NOTICE OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTY

Arizona Public Service Company
Palo Verde Nuclear Generating Station

Docket Nos. 50-528; 50-529; 50-530
License Nos. NPF-41; NPF-51; NPF-74
EA-05-051

During an NRC inspection completed December 8, 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 NRC proposes to impose a civil penalty pursuant to Section 234 of the Atomic Energy Act of 1954, as amended (Act), 42 U.S.C. 2282, and 10 CFR 2.205. The particular violation and associated civil penalty are set forth below:

10 CFR 50.59(a)(1) [1992 version] states, in part, that the holder of a license authorizing operation of a production or utilization facility may: (1) make changes in the facility as described in the safety analysis report, (2) make changes in the procedures as described in the safety analysis report, and (3) conduct tests or experiments not described in the safety analysis report, without prior Commission approval, unless the proposed change, test, or experiment involves a change in the Technical Specifications incorporated in the license or an unreviewed safety question. A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question: (1) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; (2) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (3) if the margin of safety as defined in the basis for any Technical Specification is reduced.

The Palo Verde Nuclear Generating Station Updated Final Safety Analysis Report (UFSAR), Section 6.3, "Emergency Core Cooling System," states, in part, that the safety injection piping will be maintained filled with water, and that during recirculation mode, the available net positive suction head for the containment spray and high pressure safety injection pumps is 25.8 feet and 28.8 feet, respectively (values that assume the pump suction piping is filled with water).

Contrary to the above, on June 22, 1992, the licensee made a procedural change which resulted in a change to the facility as described in the UFSAR that increased the probability of a malfunction of equipment important to safety previously evaluated in the safety analysis report, and the licensee failed to perform a written safety evaluation and obtain Commission approval prior to implementing the change. Specifically, a change was made to Surveillance Procedure 41ST-1SI09, "ECCS Leak Test," which drained, and left empty, a portion of the containment sump safety injection recirculation piping at the conclusion of the leak test. This change also affected the available net positive suction head analysis described in the UFSAR for the containment spray and high pressure safety injection pumps, which are important to safety, since these analyses assumed the pump suction piping would be filled with water.

This is a Severity Level III violation (Supplement I).
Civil Penalty - $50,000

Pursuant to the provisions of 10 CFR 2.201, Arizona Public Service Company (Licensee) 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-05-051" and should include for each alleged violation: (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 previous documented 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. Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Within the same time as provided for the response required above under 10 CFR 2.201, the Licensee may pay the civil penalty proposed above, in accordance with NUREG/BR-0254 and by submitting to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555, a statement indicating when and by what method payment was made, or may protest imposition of the civil penalty in whole or in part, by a written answer addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission. Should the Licensee fail to answer within the time specified, an order imposing the civil penalty will be issued. Should the Licensee elect to file an answer in accordance with 10 CFR 2.205 protesting the civil penalty, in whole or in part, such answer should be clearly marked as an "Answer to a Notice of Violation, EA-05-051" and may: (1) deny the violation listed in this Notice, in whole or in part, (2) demonstrate extenuating circumstances, (3) show error in this Notice, or (4) show other reasons why the penalty should not be imposed. In addition to protesting the civil penalty, in whole or in part, such answer may request remission or mitigation of the penalty.

In requesting mitigation of the proposed penalty, the factors addressed in Section VI.C.2 of the Enforcement Policy should be addressed. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanation in reply pursuant to 10 CFR 2.201, but may incorporate parts of the 10 CFR 2.201 reply by specific reference (e.g., citing page and paragraph numbers) to avoid repetition. The attention of the Licensee is directed to the other provisions of 10 CFR 2.205, regarding the procedure for imposing a civil penalty.

Upon failure to pay any civil penalty due which subsequently has been determined in accordance with the applicable provisions of 10 CFR 2.205, this matter may be referred to the Attorney General, and the penalty, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234c of the Act, 42 U.S.C. 2282c.

The response noted above (Reply to Notice of Violation, statement as to payment of civil penalty, and Answer to a Notice of Violation) should be addressed to: Frank Congel, Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852-2738, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, TX 76011, and a copy to the NRC Resident Inspector at the Palo Verde Nuclear Generating Station.

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 are required to post this Notice within two working days.

Dated this 8th day of April 2005

EA-04-221 - Palo Verde (Arizona Public Service Company)

April 8, 2005

EA-04-221

Gregg R. Overbeck, Senior Vice
President, Nuclear
Arizona Public Service Company
P.O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A YELLOW FINDING AND NOTICE OF VIOLATION - NRC SPECIAL INSPECTION REPORT 2004-014 - PALO VERDE NUCLEAR GENERATING STATION

Dear Mr. Overbeck:

The NRC's January 5, 2005, inspection report described the results of a special inspection that followed up on your discovery in July 2004 that a significant section of containment sump safety injection piping at all three Palo Verde Nuclear Generating Station (PVNGS) units was void of water. The report discussed two findings that were being evaluated for further NRC action under the NRC's Significance Determination Process or NRC Enforcement Policy. This letter provides you the results of our evaluation of one of the findings, the preliminary "Greater than Green" finding involving a failure to maintain portions of the PVNGS emergency core cooling system (ECCS) filled with water in accordance with design control requirements. This finding was processed under the NRC's significance determination process. In separate correspondence, we are providing you the results of our enforcement deliberations on the second finding, an apparent violation of 10 CFR 50.59 that was processed under the NRC's Enforcement Policy.

NRC's evaluation of the design control finding considered the fact that Arizona Public Service Company (APS) discovered this condition at PVNGS in July 2004, following notification from another facility where a similar problem had been identified. On July 31, 2004, APS reported this condition to NRC under the provisions of 10 CFR 50.72(b)(3)(v), noting that the voided section of piping had the potential to prevent the fulfillment of the safety function to remove residual heat and mitigate the consequences of a loss-of-coolant accident. In early August, Palo Verde took corrective action to fill the voided piping in all three units, completing those actions by August 4, 2004.

At your request, a Regulatory and Predecisional Enforcement Conference was held on February 17, 2005, to discuss APS's perspectives on the risk significance of the design control issue. During the meeting the APS staff described their assessment of the significance of the finding, including the results of detailed pump testing APS sponsored to assess the performance of the high pressure safety injection (HPSI) and containment spray (CS) pumps with portions of the ECCS suction piping voided. The APS staff also described corrective actions, including the root cause evaluations for the failure to maintain the design of ECCS suction piping. The APS staff indicated that maintaining voided ECCS suction piping was contrary to the original design intent and was an unanalyzed condition. Your investigation identified possible causes as including: (1) the design requirement was specified, but the end user did not consider the design requirement and incorporate the requirement into procedures; (2) the design requirement was recognized, but there was a breakdown in communicating the design requirement to the end user; and (3) the design requirement was not recognized by the responsible design organization.

The APS staff indicated that the pump testing demonstrated that high pressure safety injection pumps would function for all loss of coolant accidents associated with a pipe break greater than 2.0 inches in diameter. Additionally, the APS staff

indicated that, as a conservative measure during the significance determination, no change was made to your probabilistic safety assessment model to account for small-break loss of coolant accidents between 2.0 and 2.3 inches. The APS staff indicated that the significance of the finding should be characterized as having low to moderate safety significance (White) because the change in core damage frequency from the subject performance deficiency was 7.0 x 10^{-6}.

After considering the information developed during the inspection, the information APS provided at the conference, and the information APS provided in letters dated December 27, 2004, February 10, 2005, February 15, 2005, February 24, 2005, and February 28, 2005, the NRC has concluded that the inspection finding is most appropriately characterized as a Yellow finding, i.e., an issue with substantial importance to safety that will result in additional NRC inspection and potentially other NRC action. While we agreed with many of the assumptions that formed the basis for your risk determination, we identified many uncertainties associated with the conduct of the pump tests. A discussion of these uncertainties, their effect on our significance determination, and the primary basis for the NRC's conclusion, follows.

The NRC's review determined that the pump testing provided useful insights into post-accident high pressure safety injection pump performance. Nevertheless, there were several uncertainties associated with the testing and analysis methodologies that could have an impact on the overall conclusions regarding the availability of ECCS pumps following a loss-of-coolant accident. The significant test method uncertainties were in the areas of: (1) the use of the Froude Correlation and scaling, and (2) the impact of temperature on required net positive suction head. There were also several uncertainties associated with differences between the test configuration and the actual plant configuration. The significant configuration uncertainties were in the areas of: (1) the use of ambient temperature water during testing in lieu of post-accident temperature water, (2) the use of a method of air injection during the full scale testing that did not represent the actual void discovered in the plant, (3) the failure to model the transition between suction sources and the associated impact on check valve and system response, and (4) the failure during testing to account for post-accident conditions affecting the pump discharge.

We evaluated the above test method and test configuration concerns and concluded that they introduced large qualitative uncertainties associated with the selection of the loss-of-coolant accident break spectrum utilized by the APS staff in completing the safety analysis. After accounting for the uncertainties, we concluded that at least some portion of the medium loss-of-coolant accident break spectrum should be included in the significance determination of the failure to maintain the ECCS suction piping filled with water.

Taking into account these uncertainties, we determined that the most appropriate value for the change in core damage frequency lies between 5.7 x 10^{-6}, the result assuming that the performance deficiency only affects system response to small breaks, and 4.6 x 10^{-5}, the result assuming that high pressure safety injection pumps would fail on recirculation during a medium-break LOCA. Given that 89 percent of the range of core damage frequency lies in the Yellow region, as defined by the significance determination process, we have concluded that the most appropriate characterization of the significance of this finding is Yellow. Additional details of our evaluation and basis for arriving at a Yellow significance determination are contained in Enclosure 2.

We will use the NRC Action Matrix to determine the most appropriate NRC response for this issue. We will notify you by separate correspondence of that determination.

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

The NRC also has determined that the failure to maintain portions of the Palo Verde ECCS in accordance with design specifications is a violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control. This violation is cited in the enclosed Notice of Violation (Notice), Enclosure 1. The circumstances surrounding this violation were described in detail in the subject inspection report. In accordance with the NRC Enforcement Policy, the Notice of Violation is considered escalated enforcement action because it is associated with a Yellow finding. You are required to respond to the violation and should follow the instructions specified in the enclosed Notice in preparing your response.

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), 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/

Bruce S. Mallett
Regional Administrator

Docket Nos. 50-528; 50-529; 50-530
License Nos. NPF-41; NPF-51; NPF-74

Enclosures
1. Notice of Violation
2. Final Significance Determination

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Enclosure 1

NOTICE OF VIOLATION

Arizona Public Service Company
Palo Verde Nuclear Generating Station

Docket Nos. 50-528; 50-529; 50-530
License Nos. NPF-41; NPF-51; NPF-74
EA-04-221

During an NRC inspection completed December 8, 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 Part 50, Appendix B, Criterion III, Design Control states, in part, that measures shall be established to assure that the design basis is correctly translated into specifications, procedures, and instructions.
>
> The design basis for the Palo Verde Nuclear Generating Station (PVNGS) is specified, in part, in the plant Updated Final Safety Analysis Report (UFSAR). Section 6.3 of the UFSAR, "Emergency Core Cooling System," states, in part, that the safety injection piping will be maintained filled with water, and that during recirculation mode, the available net positive suction head for the containment spray and high pressure safety injection pumps is 25.8 feet and 28.8 feet, respectively (values that assume the pump suction piping is filled with water.)
>
> Contrary to the above, from initial plant licensing until July 2004, the design control measures established by the licensee were not adequate to assure that the design basis for the PVNGS emergency core cooling system (ECCS) was appropriately translated into specifications, procedures, and instructions. The licensee had no specifications, procedures or instructions in place to assure that the design basis for the ECCS system was maintained. Specifically, except for limited periods of time following ECCS leak testing prior to 1992, the licensee failed to maintain portions of the containment sump safety injection recirculation piping filled with water in accordance with the UFSAR, a nonconformance that affected the available net positive suction head for the containment spray and high pressure safety injection pumps as described in the UFSAR. This condition existed at Units 1, 2 and 3 of the PVNGS facility from initial plant operation (1985, 1986 and 1987, respectively) until August 2004, at which time corrective actions were taken to fill the affected piping.
>
> This violation is associated with a Yellow SDP finding.

Pursuant to the provisions of 10 CFR 2.201, Arizona Public Service Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, TX, 76011, and a copy to the NRC Resident Inspector at the Palo Verde facility, 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-04-221" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852-2738. Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated this 8th day of April 2005

Inside NRC

Volume 27 / Number 9 / May 2, 2005

Bush calls for insurance against new reactor regulatory delays

A federal risk insurance policy that President George W. Bush called for last week would potentially protect companies that invested in the first four new U.S. nuclear reactors against any licensing delays that might arise during NRC ITAAC verification hearings.

Under NRC's new combined construction permit-operating license (COL) process, an application for a new reactor would be subject to a mandatory public hearing before a license is granted and potentially to a second hearing after the unit is constructed. The second hearing would occur after an ITAAC—short for inspections, tests, analyses, and acceptance criteria—review has been conducted to verify the reactor was constructed as licensed. As with the seed money and noncash incentive package the administration and industry also are seeking for new reactors, it is believed that government assistance would only be needed for the first few new plants. Confidence in the regulatory and financial arenas are expected to be well established after that.

In a speech April 27 at a national small business conference, President Bush said he has "asked the Department of Energy to work on changes to existing law that will reduce uncertainty in the nuclear plant licensing process, and also provide federal risk insurance that will protect those

building the first four new nuclear plants against delays that are beyond their control."

DOE, NRC, and industry witnesses at a Senate Energy & Natural Resources Committee on April 26 expressed concern that a potential second hearing could raise the specter of a delay in reactor operations, creating an uncertainty that could deter investment in new reactors. DOE Deputy Secretary Clay Sell pointed to the regulatory turmoil following the 1979 accident at Three Mile Island-2, noting that a lingering apprehension still exists about "the risk of a catastrophic delay due to a problem in the licensing process." Investors in nuclear power should have some certainty that they can benefit from that investment in five years, Sell said. He told committee Chairman Pete Domenici (R-N.M.) that DOE would be willing to work with Congress to address that risk. Domenici, a strong supporter of nuclear power, and other lawmakers want to set the framework to expand nuclear power's share of a diverse U.S. energy mix.

Currently, no other energy technology faces a similar operational uncertainty after a plant has been built, according to a nuclear industry source. The federal insurance Bush is advocating is seen as being separate from the incentive package DOE and the industry are seeking to ease the financial risks the first four new reactors would face (Nucleonics Week, 31 March, 3). Domenici indicated during the hearing last week that he wanted to include provisions in his energy bill to address the regulatory risk.

No information was available at press time on what the administration envisioned would be involved in a federal risk insurance proposal for new reactors that is expected to come out of DOE. But one Washington observer was surprised the administration would create an insurance policy to protect against the government's own actions.

Sen. Chuck Hagel (R-Neb.) introduced a bill April 21 (S. 887) that would direct DOE to encourage the adoption of technologies that reduce greenhouse gas intensity. Among other things, the bill discusses "regulatory failure," which it defines as "a situation in which, because of a breakdown in a regulatory process or an indefinite delay caused by a judicial challenge to the regulatory consideration of a specific eligible project, the federal or state regulatory or licensing process governing the siting, construction, or commissioning of an eligible project does not produce a definitive determination

that the eligible project may go forward or stop within a predetermined and prescribed time period, as determined by the Secretary [of Energy]."

The bill goes on to provide for "standby default coverage" for an eligible project. It defines such coverage this way: "A pledge by the Secretary [of Energy] to pay all or part of the debt obligation issued by an obligor and funded by a lender, plus all or part of obligor equity, if an eligible project fails to receive an operating license in a period of time established by the Secretary because of a regulatory failure or other specific issue identified by the Secretary."

Under NRC's untested COL process, all issues are to be resolved during the first hearing. NRC Chairman Nils Diaz said at the Senate hearing that the second NRC hearing would be held only to answer the question of whether the reactor was constructed as licensed and, if not, whether any differences found were significant. Though Diaz also noted the agency was concerned about the possibility for delay, he said that any issue resolved during the first hearing could not be reopened.

Still, an unresolved issue would not necessarily delay the start of new reactor operations, Diaz said. In response to a question from Sen. Lamar Alexander (R-Tenn.), he told the panel that NRC could let a new reactor begin operations while resolution of an issue was pending, providing the agency didn't think the issue was substantial.

NRC readies for applications

The NRC has a solid process in place to license new reactors but could face challenges in assigning resources to handle the expected influx of license applications, Diaz said. In response to a question from Domenici, Diaz said the agency projects it would need an additional $20-million a year to handle the workload.

"Although specific plans are not yet available from the industry, the NRC may be faced with a significant increase in its workload for new reactor licensing, including receipt of up to five combined license applications beginning in 2007-2008," Diaz told the panel. He added that in order to meet this increased demand, the agency soon would have to begin preparatory activities aimed at accommodating "such large growth."

That includes activities ranging from technical work to acquiring the space and personnel needed, as well as training,

he said. NRC already has begun to hire and train new staff in anticipation of the COL applications, he said. NRC hasn't initiated review of an application for a new U.S. reactor in more than three decades.

But the agency also will be engaged in a balancing act as it manages that workload and other high priority activities, such as security and fuel cycle work, Diaz noted in his testimony. Congressional support for new nuclear generating capacity was evident at the Senate Energy Committee hearing, as well as at hearings later in the week by the House Government Reform Subcommittee on Energy & Resources and the House Science Subcommittee on Energy.

Nuclear power's appeal stems not only from its competitiveness, but also from the reasoning that the construction of new nuclear generating capacity could reduce the cost of natural gas by reducing the demand for that energy source. Most of the new generating plants being built now use natural gas; a reduction in the cost of that commodity could benefit everything from the chemical industry to the cost of farming, which are affected by natural gas prices, DOE's Sell said.

In addition, nuclear and hydro power are the two major emission-free sources of electricity in the U.S. today, Sell said. Additional coal-fired plants, one source of emissions, are the only feasible baseload alternative to new power reactors, he added.

Rep. Darrell Issa (R-Calif.), chairman of the House Government Reform Energy Subcommittee, asserted last week that the U.S. would have been in compliance with the requirements of the Kyoto Protocol today if all of the reactors had been built that were ordered at the time of the 1979 partial core meltdown at Three Mile Island-2. Under the Clinton-era international Kyoto agreement, which the U.S. did not sign under the Bush administration, countries are required to reduce their emission of greenhouse gases to their 1990 levels.

Separately, Sell told the Senate Energy Committee that DOE is committed to a seven-year, $1.1-billion cost-share COL effort with industry. DOE's Nuclear Power 2010 program, which will shepherd the COL applications, is aimed at having at least one advanced LWR on line around 2014. Sell said DOE experienced a delay in initiating separate projects by the Dominion Energy and NuStart Energy Development LLC consortia after Dominion changed its preferred reactor technology last year just prior to the issuance of a cooperative

agreement with DOE in December.

"We believe that Dominion's change from the Atomic Energy of Canada Limited Advanced Candu Reactor, ACR-700, to the General Electric Economic Simplified Boiling Water Reactor (ESBWR), resulted from the longer-thanexpected certification schedule set forth by the Nuclear Regulatory Commission for the AECL reactor," Sell said in written testimony. "The selection of the GE reactor technology affected the issuance of the Dominion cooperative agreement as well as the scope of the NuStart project, because the GE technology is also a part of that project." Sell said DOE asked the two consortia to work out an equitable arrangement and to submit proposals that split the cost of work on the GE technology between the two projects. DOE finalized a cooperative agreement with Dominion March 31, and Dominion has initiated work on the planning phase, which will establish the detailed project schedule and budget, Sell said. The Dominion-led project is to submit a design certification application to the NRC later this fiscal year, which ends Sept. 30, for the ESBWR. The Dominion team is focused on deploying that reactor technology at Dominion's North Anna site.

DOE finalized its cooperative agreement with NuStart on April 26, moving that team into its detailed planning phase. The consortium—made up of nine electric companies and vendors GE and Westinghouse—plans to select two sites for new reactors later this year. It also hopes to accelerate the filing of a COL application to an earlier date in 2008 than originally planned.—***Elaine Hiruo, Washington***

Inside NRC

Volume 27 / Number 9 / May 2, 2005

NMC trying hard to fix problems at two plants

Nuclear Management Co. (NMC) last month presented plans to NRC for dealing with problems that have caused an extended shutdown at Kewaunee and may prolong the current refueling outage at Point Beach-2.

In both cases, NMC is submitting license amendment requests (LARs) and asking NRC to consider them as "exigent" under agency regulations (10 CFR50.91). The Point Beach request deals with NMC's analysis of handling heavy loads—notably the old and new reactor heads—during the unit's vessel head replacement, which is to take place during the ongoing outage.

At an April 27 meeting at NRC headquarters, Jim McCarthy, the Point Beach director of site operations, said the company planned to begin lifting the new vessel head just before midnight on May 9. Regulatory Affairs Manager Aldo Capristo said NMC planned to submit the LAR within days. But that leaves NRC with about a week to review the LAR before the May 9 target date.

Harold Chernoff, NRC project manager for Point Beach, emphasized at the meeting that NMC needed to state very clearly the reasons for considering the request as exigent. Chernoff also asked about short-term contingency plans in case NRC does not approve the LAR; McCarthy said NMC has "not looked at that in detail." The old head already has been removed.

The LAR would amend the Point Beach final safety analysis report (FSAR) to incorporate a head-drop analysis that includes elements such as mitigation strategies and administrative controls, Capristo said. Altering the FSAR in this way would introduce a new accident scenario and therefore wouldn't clear the 10 CFR 50.59 evaluation process for a

type of change, test, or experiment that could be implemented without prior NRC approval, he said.

Chernoff said the issues concerning the adequacy of NMC's current analysis, which dates from 1982, were initially raised by a resident inspector during preparations for the head replacement (INRC, 18 April, 12).

NMC carried out a head replacement at Kewaunee last fall. Because of design differences between that plant and Point Beach, McCarthy said, the head-drop analysis was not an issue at Kewaunee. An NRC staffer said initial indications were that the issue was unique to Point Beach, but he emphasized that more information was needed before the staff draws that conclusion.

Meanwhile, NMC is grappling with a different set of problems at Kewaunee. The plant shut down Feb. 20 after NMC "determined that a high energy line break had the potential to affect the AFW [auxiliary feedwater] pump suction line from the condensate storage tank (CST) due to the inability of the discharge pressure switches to protect the AFW pumps from a loss of suction from the CST," according to the company's report to NRC.

At an April 20 meeting at NRC's Region III headquarters, Lori Armstrong, Kewaunee's site engineering director, said NMC would be submitting two LARs—one dealing with the AFW pumps and the other dealing with the ability of the emergency diesel generator exhaust duct to withstand a tornado-generated missile. NMC spokeswoman Maureen Brown said April 29 the requests were expected to be submitted this week. During the shutdown, NMC has carried out a broad review of the way it runs the plant. At the meeting, Site Vice President Craig Lambert said the company had identified a number of weaknesses in both engineering and operations. But NMC's analysis went beyond those areas to cover issues such as manager and supervisor effectiveness, he said. NMC officials said NMC found that the Kewaunee operators had low standards and that the plant had allowed itself to become isolated from the industry. As part of its improvement effort, they said, NMC has adopted a three-part approach to raising standards. NMC managers will explain a standard, coach the relevant personnel on it, and then demonstrate it, they said. The three-part approach is critical to ensure that plant staff fully absorb and integrate new standards, they said.

The officials said the plant had made progress in addressing the problems. In a March letter to NRC, NMC listed 17

commitments it would meet before restarting the plant. At the meeting, NMC said it had met 10 of them.
The commitments cover five areas: operations leadership, configuration management, engineering effectiveness, corrective action process, and management effectiveness. NMC also has 21 long-term commitments in the same areas.

Kewaunee is in the process of being sold from current owners Wisconsin Public Service Corp. and Wisconsin Power & Light Co. to Dominion. Some of the Dominion managers who will assume roles at Kewaunee attended the meeting. Dominion's Joe Ruttar, the incoming operations manager, said NMC was "aligning" with the operational standards of Dominion and the Institute of Nuclear Power Operations.
After the meeting, Cynthia Pederson, director of the division of reactor safety in NRC's Region III, said the AFW issue that prompted the shutdown had surfaced during an NRC inspection that was completed Feb. 18. The inspection was part of an NRC "temporary instruction" under which one plant in each of NRC's four regions was chosen for a pilot inspection on engineering design.

At an April 28 public meeting on reactor oversight process issues, one NRC staffer called the Kewaunee AFW situation the most significant finding to come out of the special engineering inspections.

Region III chose Kewaunee for that inspection based on past concerns about the plant, Pederson said.
The AFW problem is part of the reactor's original design, but another NRC staffer said there are "always opportunities" for such problems to be discovered if a plant has "strong engineering programs," "alert operators" and "questioning attitudes."—***Daniel Horner, Washington***

Inside NRC

Volume 27 / Number 9 / May 2, 2005

Dyer lays out staff expectation on PRA quality for MSPI launch

A senior NRC official stopped short of saying the agency would prohibit implementation of the mitigating systems performance index (MSPI) in January 2006 if certain probabilistic risk assessment (PRA) quality activities weren't completed by year's end, but he made clear that in order to move ahead, the staff needed to have confidence in the industry's risk data that will be used in the MSPI calculation.

An April 22 letter sent by Office of Nuclear Reactor Regulation Director James Dyer put the industry on notice that the start date for the MSPI, which is to replace the existing safety system unavailability performance indicator, could slip. Dyer's letter was in response to the announcement by Nuclear Energy Institute (NEI) representatives at a routine March meeting with NRC staff that not all licensees would be able to complete a PRA quality checklist before the planned MSPI launch (INRC, 4 April, 9).

Many NRC staffers had viewed the delay of some licensees' PRA quality readiness as a retraction of actions the industry had agreed to at a January meeting.

One of the actions, which had been recommendations by a joint staff-industry MSPI PRA task group, was for licensees to resolve the most significant facts and observations (F&Os) that were identified during industry peer reviews of Level 1 PRAs (for at-power, internal events) before the MSPI implementation. Resolution of the F&Os means they must be closed-out or demonstrated to be "insignificant." The task group also recommended that licensees reconcile findings from a self-assessment against supporting requirements of Regulatory Guide 1.200. The reg guide, which has not yet been finalized, is intended to be used to determine the technical adequacy of PRA results in regulatory decision-making.

The industry says it will "disposition" the most important F&Os but that it cannot meet the self-assessment

requirements before Jan. 1. NEI officials say that part of the reason is that Reg Guide 1.200 is not suitable for MSPI because it is in draft form and subsequently could be changed. They argue that the current state of PRA quality is adequate because licensees adhere to the principles of Regulatory Guide 1.174, which provides an approach for using PRA findings to support changes an individual plant's licensing basis. But the staff counters that Reg Guide 1.174 does not establish a PRA quality standard; it only outlines the scope of what a PRA should contain.

In his letter, Dyer said, "The NRC staff goal with MSPI is that it be implemented successfully based on clear guidance and quality PRA data. Although the NRC staff is anxious to move forward with MSPI, the implementation goal of the beginning of 2006 should be a secondary consideration." Dyer said that the staff wasn't necessarily rejecting the industry's latest proposal to change the activities demonstrating there is a minimum level of quality for MSPI. But, he said, the staff would need time to review any approach different from what had been recommended by the staffindustry task group.

To demonstrate its readiness to start using MSPI, the industry had suggested it would conduct a cross-comparison of licensee PRA results to identify "outliers."

Dyer said the industry would have to detail its "process for identifying PRA model 'outlier Birnbaum importance measures'" before the staff would sign off on the new index. The Birnbaum value is a risk measure that is input to the MSPI program. Dyer said the industry would have to tell the staff how it planned to resolve the outliers, assess the impact of the most risk-significant F&Os that are unresolved at the time of the cross-comparison, and provide a timetable for resolving the F&Os affecting MSPI.

Dyer said "the timing of [the MSPI] implementation should be revisited after the ROP [reactor oversight process] working group has endorsed the industry's alternate approach."

At an April 27 meeting, NEI and industry representatives laid out the first cut of an MSPI cross-comparison process it had developed. Anthony Pietrangelo, senior director of NEI's risk regulation division, said the three owners groups have been discussions to determine the criteria for grouping plants. It may take several revisions to get the correct grouping, but he said the industry expected to have information about potential outliers by a planned June 20 workshop.

The industry did not include anything about the disposition of the F&Os in its cross-comparison chart.

The staff told NEI at the meeting that the industry needed to respond in writing to its plan for addressing PRA quality issues associated with the MSPI implementation.
—***Jenny Weil, Washington***

Inside NRC

Volume 27 / Number 9 / May 2, 2005

NRR regulatory challenges ahead include security and more

NRC's Office of Nuclear Reactor Regulation (NRR) faces many planning, technical and human capital challenges in the coming year, NRR director James Dyer told commissioners at an April 20 briefing. Dyer highlighted issues related to security, power uprates, PWR sump safety, and fire protection, as well as "succession planning and knowledge transfer" to address the agency's "aging workforce."

NRR staff "recognize that we need to improve the agency's integration of security requirements into the licensing process," NRR's Brian Sheron, associate director for project licensing and technical analysis, said at the briefing. A safety security advisory panel and working group was formed in December in conjunction with the Office of Nuclear Security & Incident Response "to ensure safety security interface issues are appropriately considered," he said. The panel will "develop a screening process for use by project managers and technical review staff to identify any aspects of license amendments that could involve security implications," Sheron said. The panel also will review licensing issues that may impact emergency preparedness, said William Kane, deputy executive director for homeland protection and preparedness.

During the question period, Chairman Nils Diaz and Commissioner Peter Lyons focused on power reactor license renewals and power uprates. NRR's decision in March to suspend review of FirstEnergy Nuclear Operating Corp.'s license renewal request for Beaver Valley due to inadequacies in the application should send "a very strong message to industry that these renewal applications need to be treated very carefully, very seriously," Lyons said. Constellation temporarily withdrew its renewal application for Nine Mile Point in March to correct problems identified by NRC (Nucleonics Week, 31 March, 1).

NRC's "ability to effectively manage our reviews in a timely and efficient way is directly proportionate over the

quality of work that's provided by our licensees," Commissioner Jeffrey Merrifield said. "If [licensees] don't do the right job, they should get it sent right back," he said. McGaffigan noted that these and other recent safety initiatives belie claims by NRC critics that the agency is "a lap dog, not a watchdog." NRR staff "drop everything if an important safety issue comes to their attention [and] they find important safety issues," he said, adding that "it's usually folks who can't win a technical argument, or won't even try, [who] question our motives."

Commissioner Gregory Jaczko inquired about the operating experience program. The program has screened more than 200 items since January, with 37 issues opened for resolution and nine closed, Sheron said. Teams with representatives from NRR, the Office of Nuclear Regulatory Research, and regional offices will be formed and meet quarterly to discuss operating experience issues, identify possible trends, and alert NRC management when action should be taken, he said.

A transcript of the briefing is available on NRC's Web site at http://www.nrc.gov/reading-rm/doc-collections/commission/tr/2005/.—***Steven Dolley, Washington***

Entergy

Summary

- Entergy is taking appropriate actions to ensure safe long-term operation of ANO-2
- Entergy's commitment management system will ensure timely implementation of new and enhanced aging management activities for license renewal

Commitment Implementation

Entergy

Commitment Management System

- Over 1,500 commitments implemented at ANO-2 over the past ten years, an average of approximately 150 per year
- 15 new commitments for ANO-2 license renewal over the next 13 years – Slightly more than 1 per year between now and the period of extended operation

Commitment Implementation

Entergy

Scope of license renewal AMP commitments

- 34 aging management program (AMP) commitments are associated with license renewal
  - 19 AMPs are already in place – 56% completed (much >56% of aging management activities needed for license renewal)
  - 15 AMPs to be enhanced or created within next ten years

Plant Improvement Initiatives

Entergy

ANO-2 Completed Improvements

- Steam generator replacement – 2000
- HP / LP turbine upgrade – 2000
- Electrical penetration module replacement – 2000
- FAC piping replacements – 1997 thru 2005

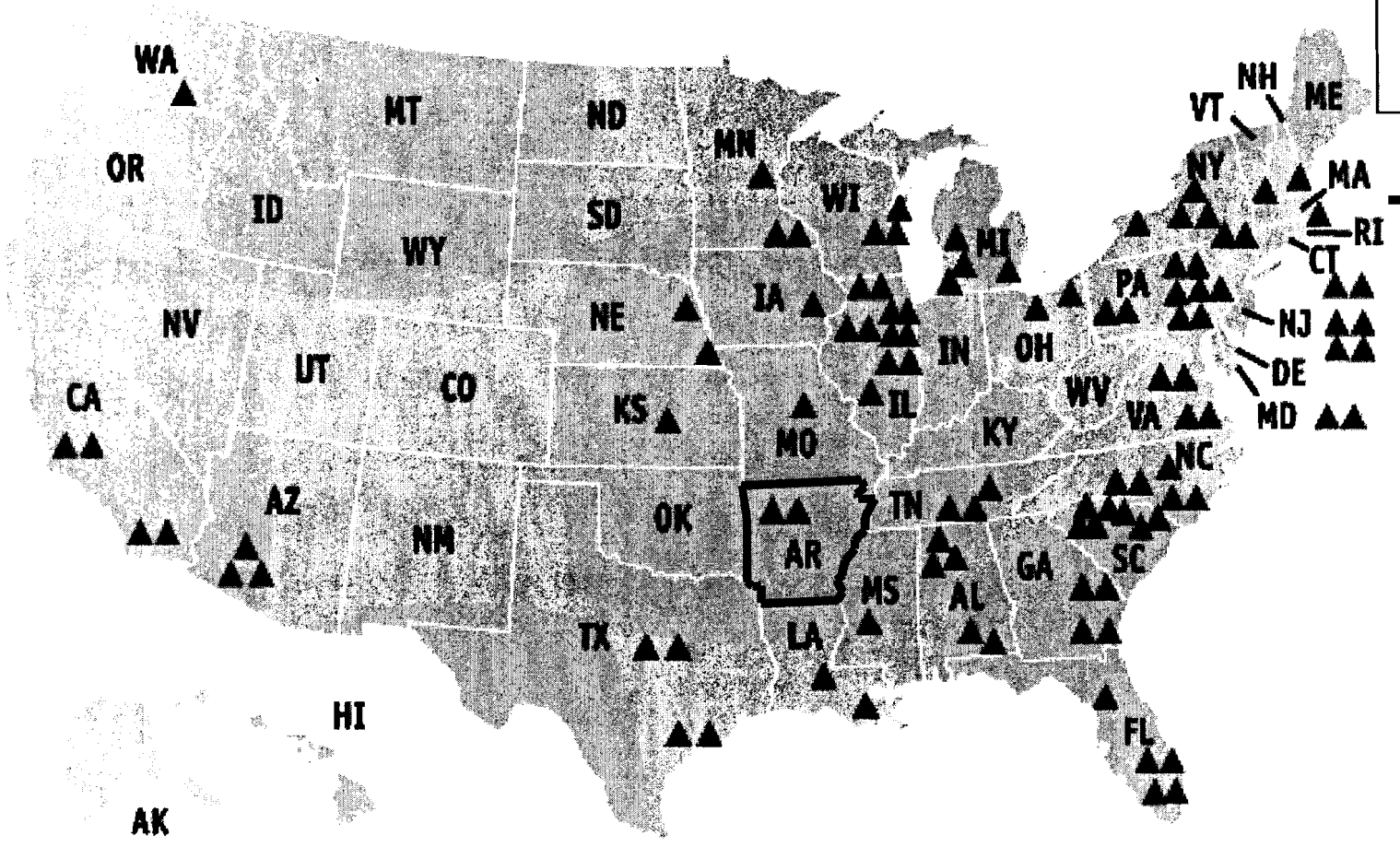
Operating History

Entergy

Performance Trends

- Capability factor increase from 71.5% (1992) to 97.4% (2004)
- ANO-2 dose reduction
  - Outage from 175.35R (1995) to 93.67R (2003)
  - Non-outage from 49.27R (1996) to 8.99R (2004)

U.S. Operating Commercial Nuclear Power Reactors



▲ Licensed to Operate (104)

Source: Nuclear Regulatory Commission

Introduction

Entergy

- Description of ANO-2
- Operating History
- Plant Improvement Initiatives
- Commitment Implementation

Handout for Item #2 (a)

Arkansas Nuclear One – Unit 2 License Renewal Safety Evaluation Report

Staff Presentation to the ACRS Full Committee

Gregory F. Suber, Project Manager
Office of Nuclear Reactor Regulation

May 5, 2005

Highlights of Review

- Items brought into scope and subject to AMR
  - One component added as a result of staff review (power transmission conductors)
  - Two components subject to AMR as a result of staff review (intake canal and feedwater outboard block valve)
  - Several components added as a result of regional inspections (spent fuel pool cooling pumps, switchyard control house, spare valve parts, miscellaneous (a)(2) components)

Regional Inspections

- Scoping and Screening Inspection
  (March 1 through 5, 2004)

- Aging Management Program Inspection
  (November 15 through 19, 2004)

- Optional Inspection
  (February 16 and 17, 2005)

Section 2 Overview

- Applicant's clarification of the scoping and screening methodology and the system walkdowns performed during the regional inspections resulted in the addition of several components
  - Components added
    - Miscellaneous components in support systems - 10 CFR 54.4(a)(2)
    - Spare valve parts - 10 CFR 54.4(a)(3)

Section 3 Overview

- Buried Piping Inspection Program
  - Buried components will be inspected within ten years after entering the period of extended operation. Credit will be taken for any opportunistic inspections occurring within this ten year period

Section 4 Overview

- Time-Limited Aging Analyses (TLAA) for Reactor Vessel USE
  - The staff independently verified the applicant's 48 EFPY calculation and performed an additional calculation using 54 EFPY for the limiting RV beltline materials. (Acceptance Criteria ≥ 50 ft-lb)

Limiting Material Intermediate Axial Weld 2-203A	Applicant USE Value	Staff USE Value	Conclusion
Calculated 48 EFPY USE	54 ft-lb	54 ft-lb	Criterion is met for 48 EFPY
Calculated 54 EFPY USE	na	52.5 ft-lb	Criterion is met for 54 EFPY

Section 4 Overview

- Alloy 600 Nozzle Repair
  - The applicant and the staff determined that the fatigue crack growth analysis remains valid through the period of extended operation.

Performance Summary

- IP 95001 Follow Up Inspection
  - Immediate/Intermediate Corrective Actions Satisfactory
  - Re-analyzing fire protection program
  - Re-evaluating post-fire manual actions
  - Commitments for modification to eliminate complex manual actions

- Consistent with Biennial PI&R Inspection Results (05/03/05)

Item # 3 Handout 1

United States Nuclear Regulatory Commission

ACRS Briefing

Control Room Staffing

James Bongarra, NRR/DIPM/IROB
Autumn Szabo, RES/DRAA/PRAB

May 5, 2005

Meeting Purpose

- Request ACRS endorsement of :
  - Revision to Standard Review Plan Chapter 13.0, Sections 13.1-2 & 13.1-3, "Operating Organization"
  - NUREG - 1791, "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)"

Agenda

- Background
- 10 CFR 50.54 (m)
- Standard Review Plan (SRP) Chapter 13.0, "Conduct of Operations, " Sections 13.1-2 & 13.1-3, "Operating Organization"
- NUREG -1791, "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)"

Background

- Staffing - Roles, responsibilities, qualifications, composition, and size of the crews required to control plant operations
- Current regulation – 10 CFR 50.54 (m)
  - Prescriptive approach
  - Prescribed numbers and qualifications of staff
  - Based on concept of operations for current light water reactors

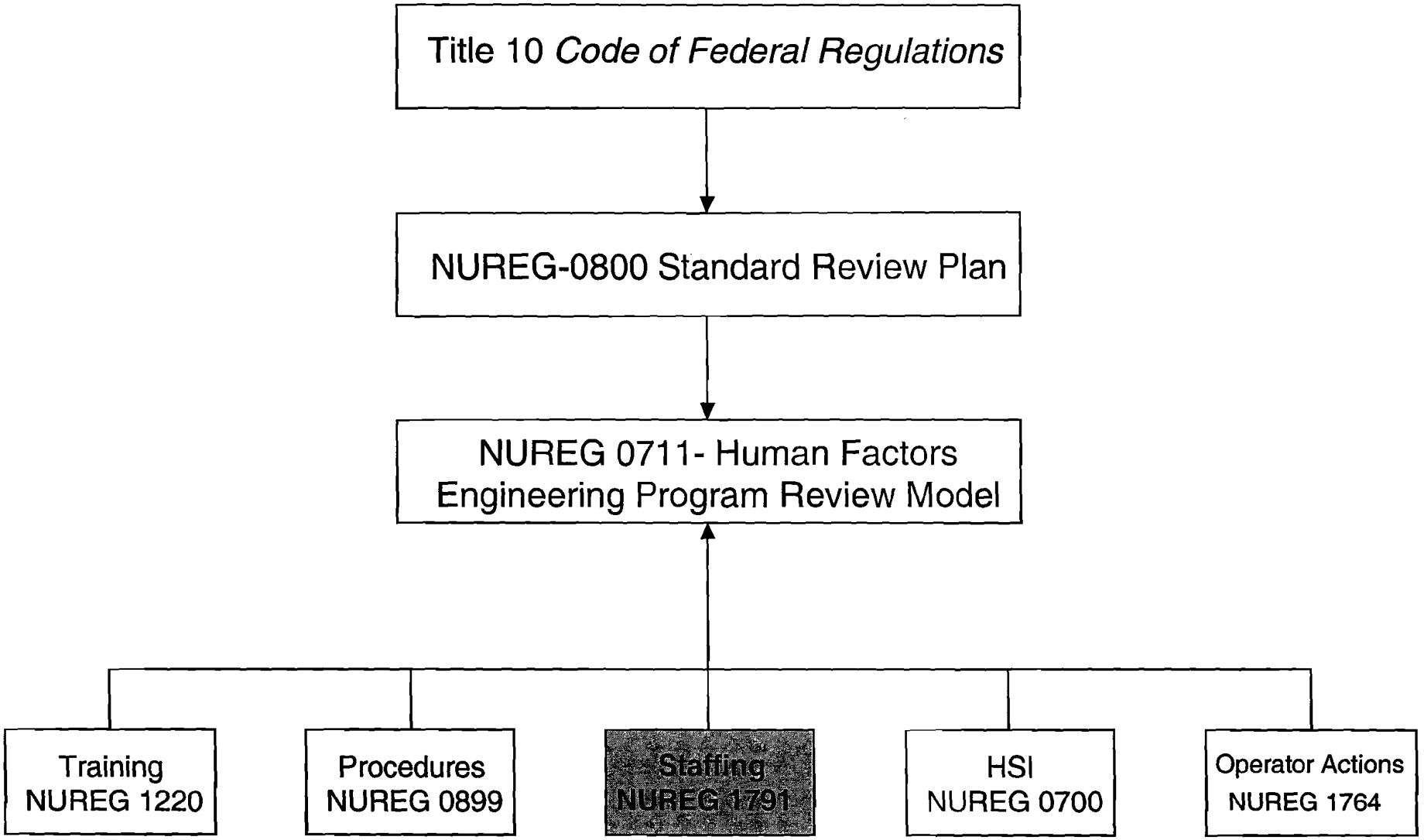
Background (con't)

- Impact of New Technologies on the Roles and Responsibilities of Licensed Personnel
  - Passive safety features
  - Simplified designs and operations
  - Multiple modular reactors per control room
  - New Human System Interface (HSI) technologies
- Implications for the Review of Exemption Requests
  - Flexible approach needed based on:
    - Change in the operator's role and qualifications
    - Staffing reductions proposed

Staffing Guidance Documents

- **NUREG-0800**, "Standard Review Plan", Chapter 13 "Conduct of Operations", Sections 13.1.2 & 13.1.3 "Operating Organization" updates
- **NUREG-1791**, "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)"
- **NUREG/CR-6838**, "Technical Basis for Regulatory Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)"
- **NUREG/IA-0137**, "A Study of Control Room Staffing Levels for Advanced Reactors"

Where Staffing Guidance Fits in the Overall Regulatory Framework



Standard Review Plan
Chapter 13.0, Sections 13.1-2 & 13.1-3

- Public Comment – November, 2004
- Minor edits to Standard Review Plan (SRP) Sections
  - incorporate new references
  - enhancement of SRP to refer to NUREG -1791, e.g., "Any requests for exemptions from the requirements of 10 CFR 50.54(m) concerning the number of licensed personnel should be justified and reviewed using the NRC's "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)" (NUREG-1791)."

NUREG-1791 Process Overview

1. Review the Exemption Request
2. Review the Concept of Operations
3. Review the Operational Conditions
4. Review Operating Experience
5. Review the Functional Requirements Analysis and Function Allocation
6. Review the Task Analysis
7. Review the Job Definitions
8. Review the Staffing Plan
9. Review of Additional Data and Analyses
10. Review the Staffing Plan Validation
11. Determine Acceptability of the Exemption Request

Each review step includes:

- **Discussion** of the review step and why it needs to be addressed
- **Data and information required** to support the review step
- **Review criteria** for evaluating the submittals
- **Additional information** that may be useful in performing the review.

Selected Steps - NUREG-1791 (1)

- Step 2: Review of the Concept of Operations
  - Understand Role of Control Personnel in plant operations
- Step 5: Review the Functional Requirements Analysis and Function Allocation
  - Defines and evaluates scenarios impacted by exemption request
  - Allocates tasks appropriately

Selected Steps - NUREG-1791 (2)

- Step 10: Review the Staffing Plan Validation
  - Appropriate considerations given to the dynamic interactions between staff, plant, and other systems

NUREG-1791 – Public Comment

- Requested Clarification on Terminology
- Clarification on Intent
- Concerns about:
  - Exemption Request Process (e.g. 10 CFR 50.12)
  - Potential Issues given failure of proposal / exemption

Summary/Conclusions

- Minor changes to SRP
- Few changes to NUREG-1791 from public comment
- NUREG-1791 – Provides regulatory staff with guidance to review exemption requests to 10 CFR 50.54 (m) staffing requirements

Item #4 Handout 1

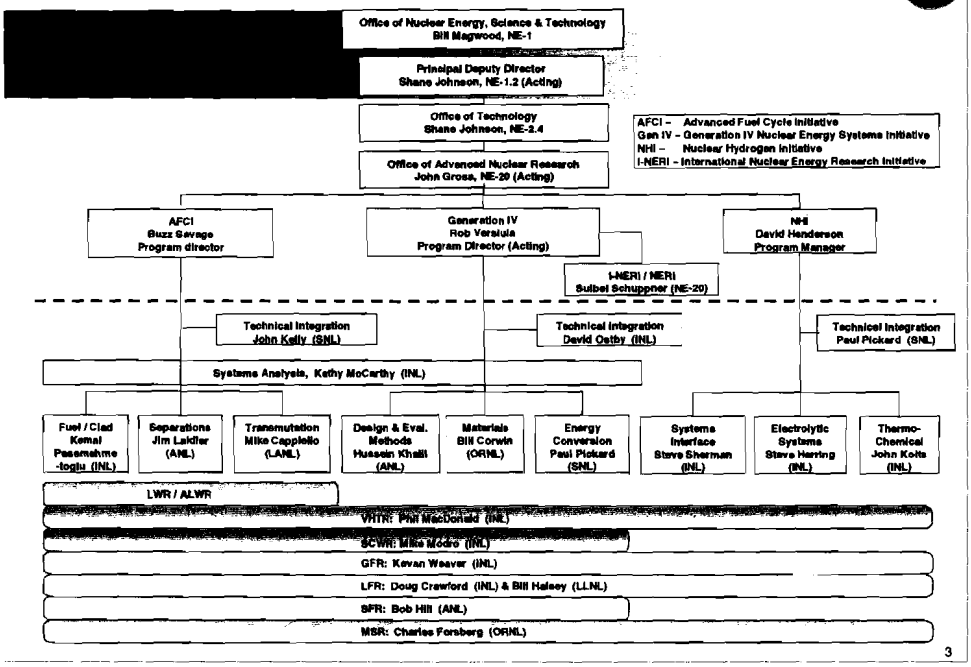
Hydrogen Production Using Nuclear Energy

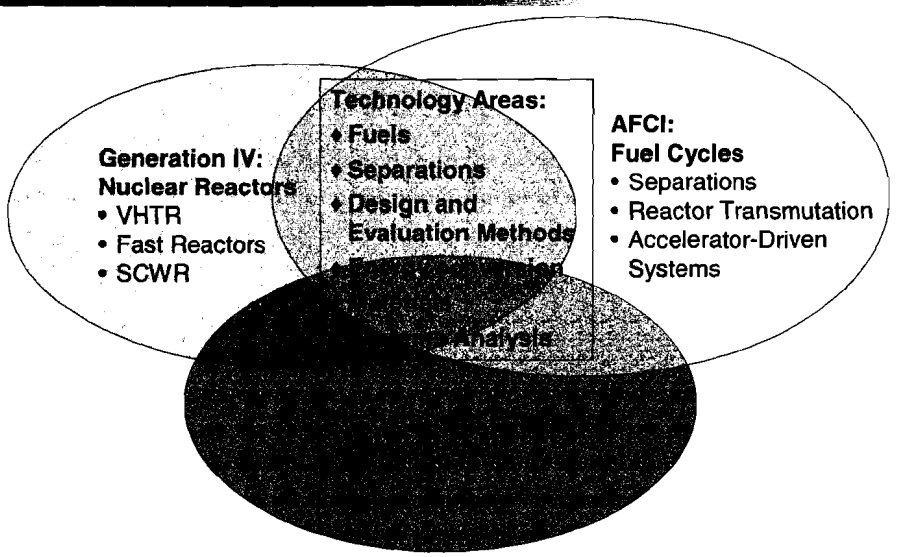
John F. Gross
Robert M. Versluis
A. David Henderson
Office of Nuclear Energy, Science and Technology

Paul S. Pickard
Sandia National Laboratories

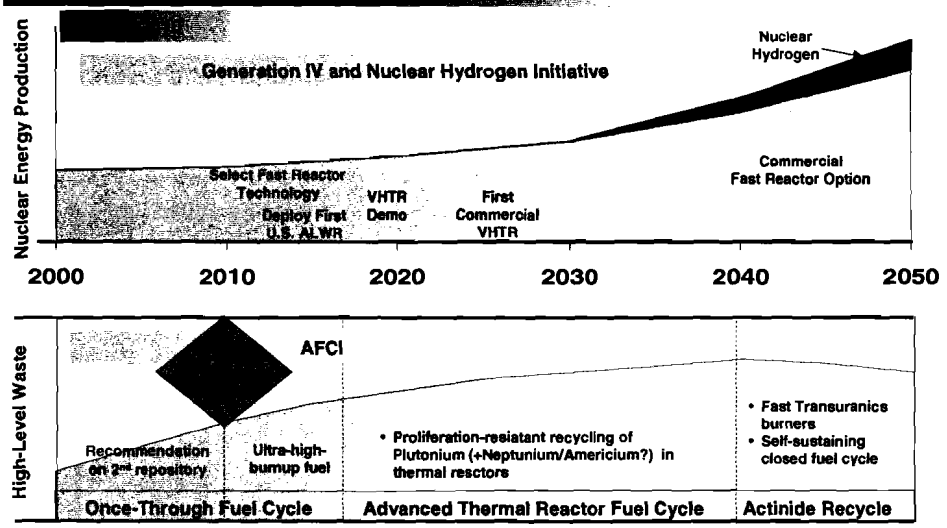
Rockville, MD
May 5, 2005

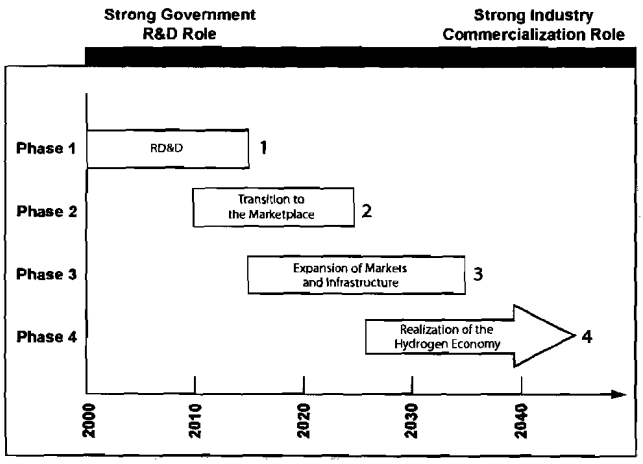
- Program and personnel
- Hydrogen production technologies
- Coupling to a nuclear energy system
- Discussion


Office of Nuclear Energy, Science & Technology
Bill Magwood, NE-1
Principal Deputy Director
Shane Johnson, NE-1.2 (Acting)
Office of Technology
Shane Johnson, NE-2.4
Office of Advanced Nuclear Research
John Gross, NE-20 (Acting)
AFCI – Advanced Fuel Cycle Initiative
Gen IV – Generation IV Nuclear Energy Systems Initiative
NHI – Nuclear Hydrogen Initiative
I-NERI – International Nuclear Energy Research Initiative
AFCI
Buzz Savage
Program director
Generation IV
Rob Versluis
Program Director (Acting)
NHI
David Henderson
Program Manager
I-NERI / NERI
Suibel Schuppner (NE-20)
Technical Integration
John Kelly (SNL)
Technical Integration
David Ostby (INL)
Technical Integration
Paul Pickard (SNL)
Systems Analysis, Kathy McCarthy (INL)
Fuel / Clad
Kemal Pasamehme -toglu (INL)
Separations
Jim Laidler (ANL)
Transmutation
Mike Cappiello (LANL)
Design & Eval. Methods
Hussein Khalil (ANL)
Materials
Bill Corwin (ORNL)
Energy Conversion
Paul Pickard (SNL)
Systems Interface
Steve Sherman (INL)
Electrolytic Systems
Steve Herring (INL)
Thermo-Chemical
John Kotts (INL)
LWR / ALWR
VHTR: Phil MacDonald (INL)
SCWR: Mike Moore (INL)
GFR: Kevan Weaver (INL)
LFR: Doug Crawford (INL) & Bill Halsey (LLNL)
SFR: Bob Hill (ANL)
MSR: Charles Forsberg (ORNL)


Technology Areas:
• Fuels
• Separations
• Design and Evaluation Methods
Generation IV:
Nuclear Reactors
• VHTR
• Fast Reactors
• SCWR
AFCI:
Fuel Cycles
• Separations
• Reactor Transmutation
• Accelerator-Driven Systems

NHI Objectives





Transitional Phases

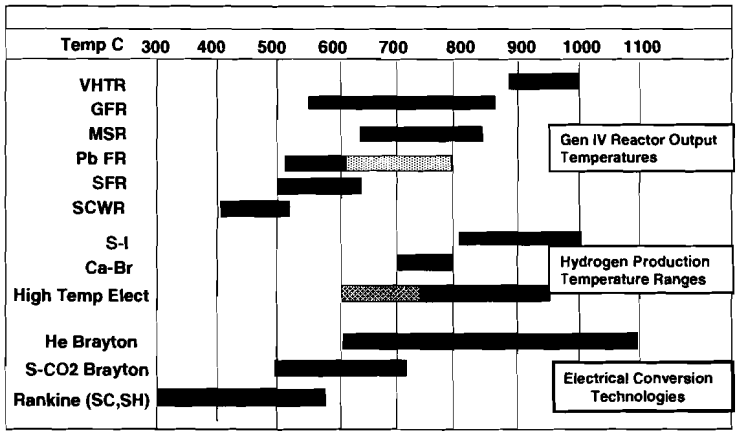
1. **Technology Development Phase**
   H_2 power and transport systems available in select locations; limited infrastructure
2. **Initial Market Penetraion Phase**
   H_2 power and transport systems begin commercielization; infrastructure investment begins with governmental policies
3. **Infrastructure Investment Phase**
   H_2 power and transport systems commercially available; infrastructure business case realized
4. **Fully Developed Market and Infrastructure Phase**
   H_2 power and transport systems commercially available in all regions, national infrastructure

03-GA51038-14

Hydrogen Production Technologies

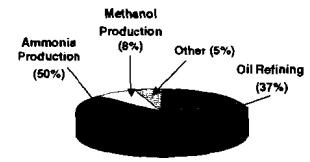
Temperatures

Potential for more efficient electricity generation and efficient hydrogen production methods

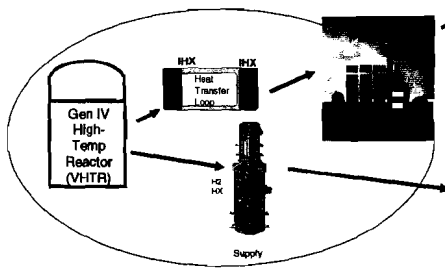


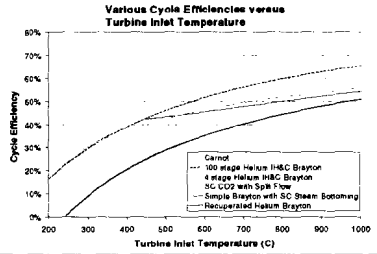
H_2 demand independent of H_2 economy and infrastructure – petrochemical industry – refining, ammonia

- **World H_2 demand ~50 Million Tons/yr**
- **US demand ~10 million tons/yr**



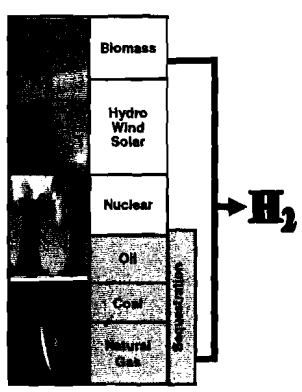
Fuel efficiency	~ 0.02 kg H_2/mile (~ 60 mpg)
H_2 – low T electrolysis	~ 1.0 kWh/mile
Miles driven (U.S.- 1997)	2.6 x 10^{12} miles
Potential Req's (US)	~ 5 x 10^7 T H_2 – – 300 GWe





Reactors

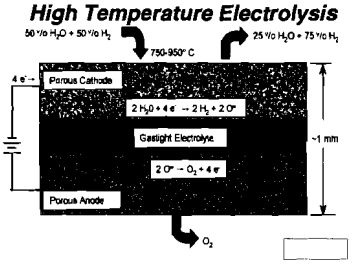
- DOE is examining all possible approaches for H_2 production – nuclear power is one of the options for large scale production.
- Nuclear could support conventional H_2 production (electrolysis, SMR)
- NE R&D focuses on large-scale, non-emitting technologies



- **Current Methods**
  - **Conventional Electrolysis**
  - **Steam Methane Reforming (SMR)**
- **Advanced Methods**
  - **Thermochemical cycles**
  - **High temperature electrolysis**

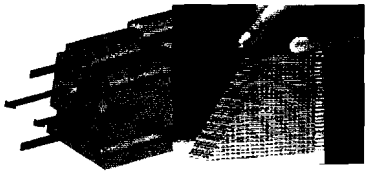
[redacted]

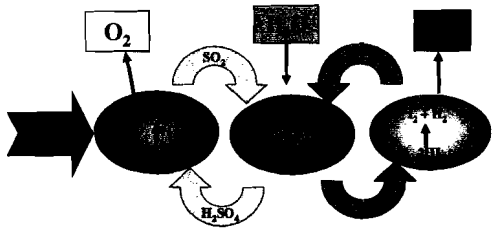
High Temperature Electrolysis



- Thermochemical Cycles *(Scaling, efficiency)*
- High Temperature Electrolysis *(modular scaling, efficiency)*
- System Interface *(High temperature materials and HX design*

System Interface

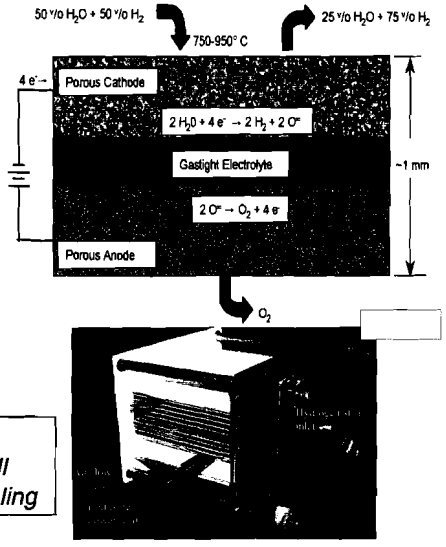




Thermochemical Cycles

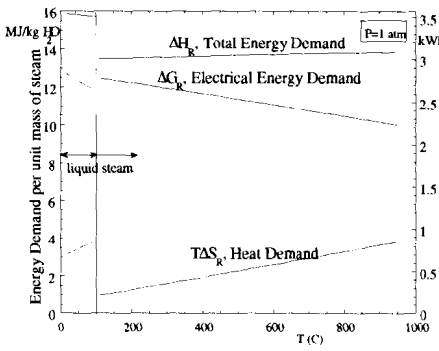
[redacted]ential for Higher Efficiency

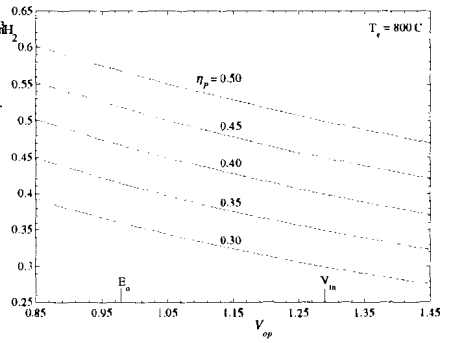
- Potential for higher efficiency than conventional electrolysis (thermal energy fraction, lower cell losses)
- Leverages fuel cell development
  - Solid oxide electrolytes
- Technology demonstrated
  - Engineering /cost issues similar to fuel cell development
- Wider range of source temperatures possible
- No hazardous industrial chemicals



R&D focuses on sealing technology, materials, cell engineering, modular scaling

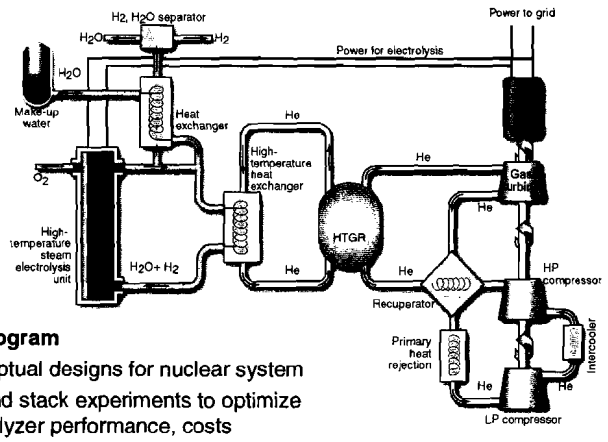
Power Electrical Demand





Electrical energy requirement decreases with increasing temperature

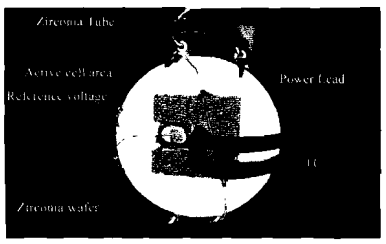
Efficiency as a function of electrical efficiency and electrolyzer cell operating voltage



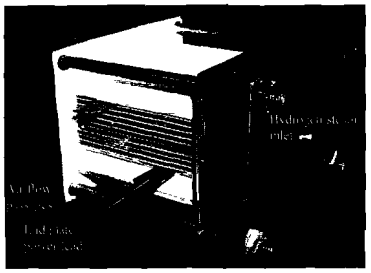
HTE Program

- Conceptual designs for nuclear system
- Cell and stack experiments to optimize electrolyzer performance, costs

- Anode: Nickel zirconia cermet (cathode in electrolysis mode)
- Cathode: Strontium-doped lanthanum manganite (anode)
- Electrolyte: YSZ, 175 μm thickness
- Active cell area: ~ 3.2 cm^2



"Button Cell" (Single-Cell), and Stack Testing Apparatus



FY2005

Improve stack performance. Perform series of stack/cell tests to develop:

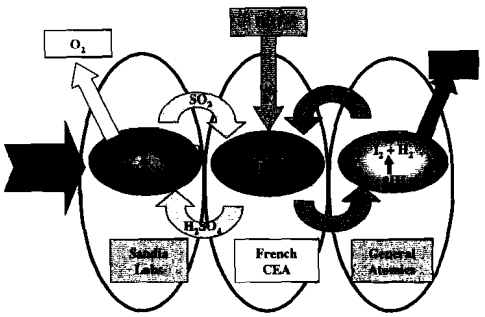
- Seals & interconnects
- Longer duration testing

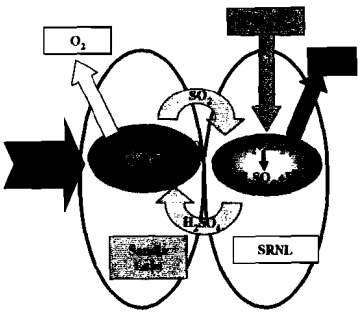
Heat Temperatures

- Thermochemical (TC) cycles produce H_2 and O_2 from water through a series of chemical reactions at lower temperatures than thermal dissociation
- Potential for high efficiency (~50%) and scaling to large sizes
- Over 100 cycles identified, only a few have been demonstrated in integrated experiments, all at lab scale and glassware
- Most cycles involve corrosive species at elevated temperatures
- NHI baseline TC cycles – Sulfur-Iodine, Hybrid Sulfur
- Alternative cycles also identified for evaluation

- DOE Program Approach: flowsheet analysis, lab scale experiments (technical feasibility), pilot scale experiments (engineering)

TC cycles require high temperatures, extensive thermal management, and high temperature, corrosion resistant materials





Sulfur-Iodine (S-I)

(1) $H_2SO_4 \rightarrow H_2O + SO_2 + 1/2O_2$

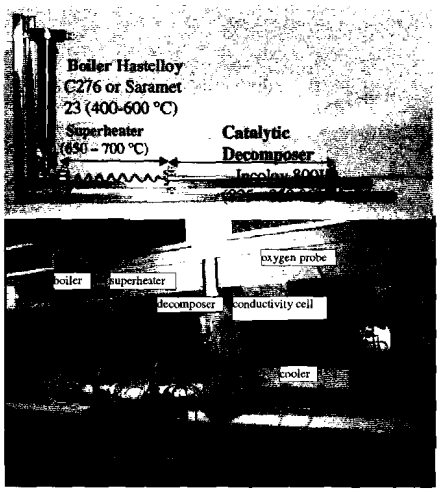
(2) $2HI \rightarrow I_2 + H_2$

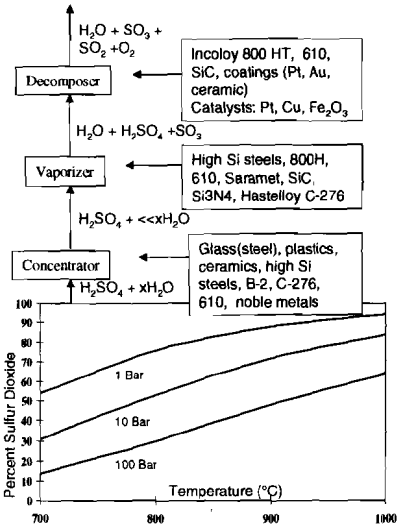
(3) $2H_2O + SO_2 + I_2 \rightarrow H_2SO_4 + 2HI$

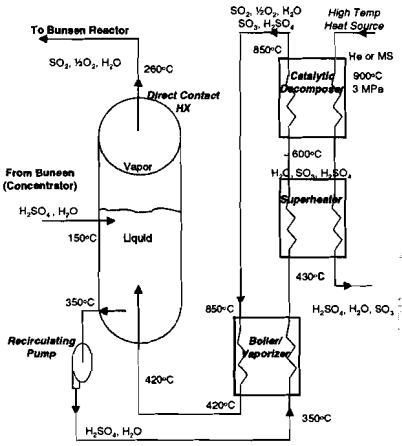
Hybrid Sulfur

(1) $H_2SO_4 \rightarrow H_2O + SO_2 + 1/2O_2$

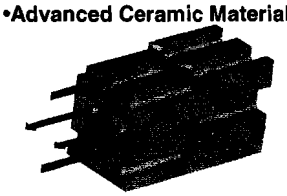
(2) $2H_2O + SO_2 \rightarrow H_2SO_4 + H_2$



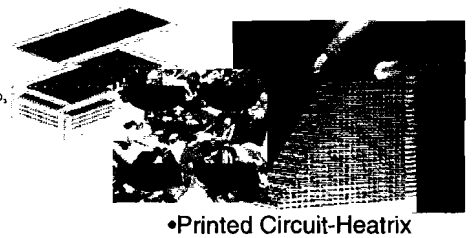




•Advanced Ceramic Materials (SiC, C-SiC)



Offset-fin C-SIC Composite plate heat exchanger



•Printed Circuit-Heatrix

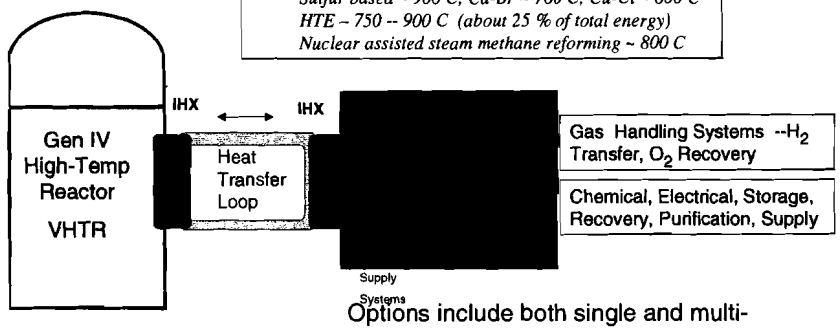
Potential for lower operating temperature, higher efficiency or less corrosive species

Candidate Alternative Cycles

	Peak Temp (°C)	Number of Rxn's	Flowsheet Efficiency % (LHV)*	Demonstration Status	Advantages	Key Issues
Copper Chlorine (GRI H-6)	<550	4	Not known	All reactions	Low peak temperature	Higher efficiency electrolysis
Iron Chlorine (Ispra Mark 9, GRI I-6)	650	3	34-36	All reactions	Low peak temperature	Suppress competing chemical reaction
Iron Chlorine (GRI B-1)	925	4	39-41	All reactions	More mature	Suppress competing chemical reaction
Copper sulfate (GRI H-5)	827-900	5	31	All reactions	Potential for high efficiency (57-61 ideal)	Economics of scaling hybrid processes Higher efficiency electrolysis
Vanadium chlorine	925	5	40.5-42.5 (Basis unknown)	All reactions	Full flowsheet	High peak temperature Conflicting data on one reaction

[redacted]diate Loop

Process temperature requirements
Sulfur based ~ 900 C, Ca-Br ~ 760 C, Cu-Cl ~ 600 C
HTE ~ 750 -- 900 C (about 25 % of total energy)
Nuclear assisted steam methane reforming ~ 800 C

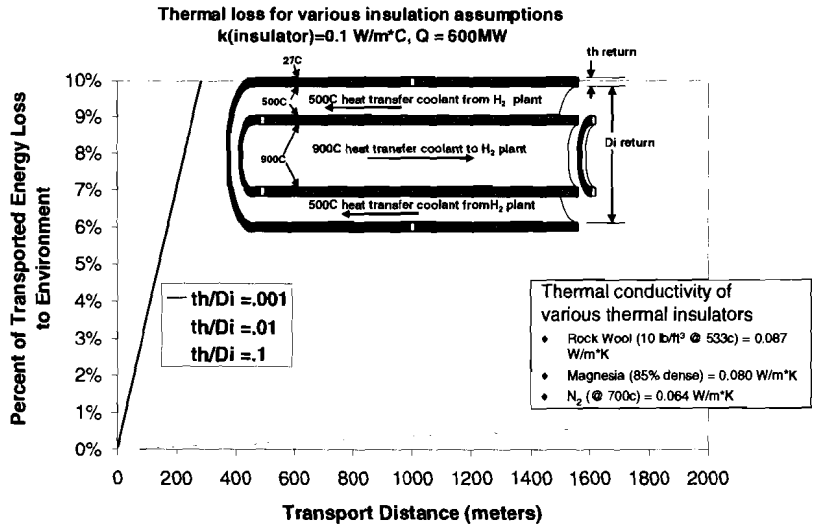


Options include both single and multi-purpose configurations (electricity and H_2).

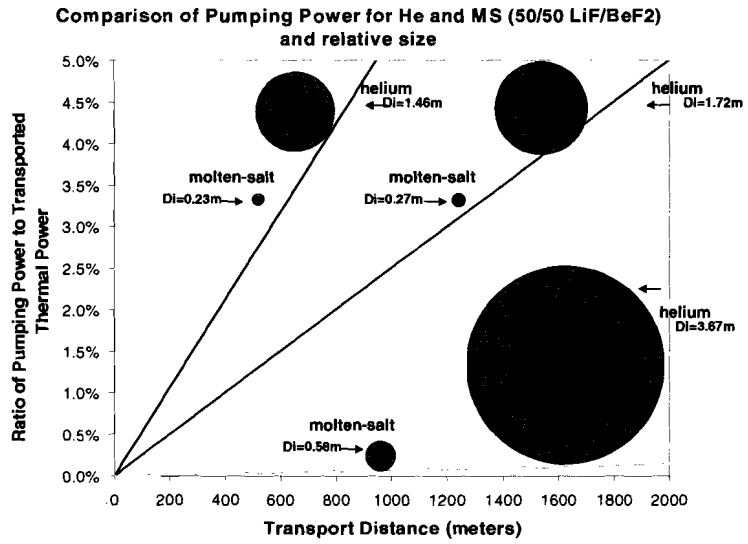
- Safety and regulatory considerations envelope issues for both plants - (radiological, chemical hazards, other industrial hazards) - potentially interactive
- Initial approach - identify chemical plant issues/scenarios that have potential to interact with nuclear plant – understand relevant industry safety experience base, apply PRA and consequence models calibrated to industry data where appropriate
- Identify configurations, separation distances or engineered features which mitigate or eliminate safety interactions of the combined facilities.
- Goal - to enable the facilities to be regulated separately, under the traditional framework for each facility.

- **Technical issues** - efficient thermal coupling, thermal losses and pumping powers as a function of separation distance, structural requirements, high temperature materials
- **Safety issues** – potential impact of chemical or nuclear hazards on combined plant safety.
  - •Hydrogen (detonation, deflagration)
  - •Oxygen (fire, spontaneous combustion)
  - •Hazardous gases (I_2, H_2SO_4, SO_3, SO_2, other cycle species)
  - •Other chemical hazards – corrosive, electrical
  - •Other Industrial hazards – high pressure, temperature components
- **Economic issues** – costs associated with increased isolation, impacts on efficiency, operation
- Candidate process designs are in an early stage – thermochemical configurations will evolve with R&D progress, HTE systems better defined
- Analysis of potential impact of key factors on cost, safety, and performance needed to help prioritize research.
- *NHI assessment is now being initiated*

Manageable



Transport



Sulfur Cycles

S-I Plant at 600 MW_{th}, 60 sec residence time

Chemical Compound	Threshold Planning Quantity (lbs) **	Estimated Inventory (600 MWth) lbs
SO2	500	~ 9000
H2SO4	1000	~ 16000
SO3	100	~ 11000

** TPQ - Threshold Planning Quantities

- **Review chemical industry safety experience base**
  - Refineries, Ammonia, Oxygen, Hydrogen, other relevant plants
- **Develop H_2 plant analysis framework - criteria**
  - NRC, Chemical industry environmental requirements
  - EPA, State regulations, security requirements
  - Codes and standards, insurance
- **Scoping estimates of production plant accident scenarios, probability, consequences**
  - PRA models – informed with industry experience
  - Estimate separation distances and design features needed to mitigate impact
  - Evaluate general cost implications
- **Involve chemical industry in safety strategy - nuclear–chemical knowledge base**
- **Identify security issues, implications**
- **Understand mitigation strategies - industry practice**

- **Review current approach - significant separation distances to mitigate combined plant safety issues**

- Approach – evaluate features that allow the facilities to be regulated separately, under the applicable framework for that facility.
- Studies in FY05, FY06 directed at evaluating separation distance and engineered features
- Incorporate chemical and nuclear perspectives and experience base
- Anticipate review of current approach in FY06, revise approach as NHI R&D progresses

Item # 5 Handout 1

U. S. Nuclear Regulatory Commission
Advisory Committee on Reactor Safeguards

Operating Reactors Summary and Analysis - CY 2003 – 2004

John D. Sieber
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Summary and Conclusions

- Operating plants are safe
- No statistically significant adverse trends are obvious
- Oversight resources are fully utilized
- There are still some problem plants

Oversight Resources

- One plant (DB-1) remains in the 0350 process
- 2 plants are in the multiple degraded cornerstone category
- 21 plants are in the regulatory response category
- 24 % of plants require augmented staff attention

Plant Safety Performance

Inspection findings

- 4 red findings
- 1 yellow finding
- 19 white findings

Performance indicators

- 6 white Performance Indicators

Cornerstone Issues

- Initiating Events – 1 white finding, 2 white PIs
- Mitigating Systems – 16 white findings, 1 yellow finding, 2 red findings, 2 white PIs
- Barrier Integrity – 1 red finding
- Emergency Preparedness – 2 white findings, 1 white PI
- Occupational Radiation Safety - none
- Public Radiation Safety – 2 white findings

Enforcement

- Point Beach – $60,000
- Davis Besse - $5,450,000
- Palo Verde - $50,000
- Perry – $55,000
- 12 plants granted enforcement discretion and not fined for violations

Leading Indicators

- Grid stability
- Events due to aging
- Recurring events
- Events during shutdown
- Increasing trends in number of events and risk severity

Industry Trends

No statistically significant adverse trends have been identified to date, based on level or declining trends in the indicators developed by the Reactor Oversight Program (ROP) and the Accident Sequence Precursor (ASP) program.

Item #5 Handout #2

Operating Experience Briefing 2005-03
April 29, 2005
10:00AM EST, Room O-3B4

Shutdown Operations Concerns & Risk

Marie Pohida DSSA | SPSB
Senior Reliability and Risk Analyst

Presentation Outline

- **Characteristics of Shutdown Operation**
- **Commission Expectations: Staff Monitor Shutdown Risk**
- **Recent Shutdown Observations**
- **Discussion of Significant Issues & Potential Solutions**

Characteristics of Shutdown Operation

- **Few Requirements**
- **Equipment to Mitigate RHR Loss or Interruption NOT Required OPERABLE**
- **Containment & Containment Systems NOT Required OPERABLE**
- **Once RHR Lost, Operator Action Required to Prevent Core Damage**

Proposed Shutdown Rule (SECY 97-168)

- **1997: Staff Requested Commission to Approve Proposed Rule for public comment**
- **SECY 97-168: Staff concluded that existing level of safety is largely dependent on voluntary actions**
- **Commission decided not to authorize Rule based on Industry Performance**

Commission Expectations

- SRM to SECY 97-168: ".. The Commission expects the staff to continue to monitor licensee performance, through inspections and other means to ensure that the current level of safety is maintained..."
- Federal Register states (dated 2/4/99): "...the Commission will continue to monitor industry performance and may take further action if any adverse trends are identified.."

Results: SECY 97-168 Regulatory Analysis

Reg. Case	PWR CDF	BWR CDF	PWR Release/yr	BWR Release/yr
Regulatory Minimum	2E-2	1E-3	2E-2	1E-3
Voluntary Minimum (NUMARC 91-06 & GL-88-17)	8E-5	1E-5	2E-5	4E-6
Voluntary Maximum (NUMARC 91-06 & GL-88-17)	2E-6	6E-7	2E-7	4E-6

BS – What about MaxRule?
→ Does not require with credits

Insights: SECY 97-168

- Base Case only credits W STS & GE-4 STS
- High & Low Industry Risk Bands, NUMARC Guidance Very High Level
  - Exceptions: Containment Closure & BWR automatic RCS low level RHR isolation operability
- Risk Accounts for Frequency & Duration of Cold shutdown
  - Results can be added to full power
- Peak Risk Periods
  - PWRs
    - 1st peak: RCS breached (SGs unavail.)
    - 2nd peak: Enter Midloop Conditions
  - BWRs Cold Shutdown Operation with head on – TS

Results: SECY 97-168

- Results strongly depend on:
  - Operator's sensitivity to plant configuration (especially during high risk periods).
  - Depth of licensee's mitigation capability (NUMARC 91-06 Guidelines)
    - Redundancy & Diversity of standby RCS injection (BWRs & PWRs)
    - Availability of SGs (PWRs)
    - Availability of RCS pressure control (BWRs –head on)
    - AC-Independent Injection (BWRs)
    - Containment Closure (PWRs)
    - Availability of AC power (BWRs & PWRs)

"8.3 hours" – used to develop a risk estimate for a particular scenario/state

used unapproved sealant for penetration

ROP Results: 1999 -2005

- SPSB Evaluated Over 50 Performance Deficiencies
  - WHITE: Containment Closure, Midloop (Oconee 01)
  - SIGNIFICANT ISSUE: Air Entrainment in RHR pumps, Midloop (Palo Verde Spring 04)
  - WHITE: Containment Closure, RCS Vented (Kewaunee October 04)
  - Preliminary WHITE: Containment Closure (Watts Bar April 05)
  - Near Miss: Authorized installation of SG hot leg dam without RCS vent path (Point Beach April 04)
  - Significant Issue: Loss of Inventory (Peach Bottom October 01)
  - Emerging: Air Entrainment at (Waterford April 05)

Briefing 2005-03

Failure to follow – did not follow NUMARC guidelines

Verify of RHR pumps during mid. loop. No measure of amount of gas needed or quantification of air impact on risk. Did not account for air entrainment in risk assessment in '97.

Salem 2 – April 8, 2005

- Reactor vessel drained Below Flange
- 8 minute time-to-boil
- 1 train SW – CCW OOS
- Experiencing problems with Grass Intrusion
- Plant was "yellow" in 6 of 9 ORAM Categories
- When questioned by NRC, licensee response: "compliance with the plants TS" & continuing as planned

Briefing 2005-03 10

Not using NUMARC guidelines

Seabrook Best-up

Senior Resident Inspector Perspectives

- Outage Risk is Significant Compared to Online Risk
  (Data based on Seabrook's last outage using their all mode PRA model)
  - 7 days of outage ~ 1 year online
  - 1 day of draining to midloop ~ the risk of a yellow finding
  - Draining to midloop greater risk than being online with loss of both SI pumps, both RHR pumps, all accumulators, 2 offsite lines, and ½ of all remaining safety components
- Difficult to Inspect
  - Minimal regulation
  - No threshold for acceptable level of risk
  - Limited ability to evaluate/inspect performance
- Variability of Licensee Implementation of Outage Risk Management Guidelines

Briefing 2005-03 11

Significant Issues

- Planned High Risk Outages Not Covered by SDP Process
  - De-Regulated Energy Environment results in shorter outages since 1997
- Performance Deficiencies may be under reported
  - Few Regulations
  - NUMARC 91-06 is vague
- Calculated Risk versus Actual Risk may be Significantly Different (e.g. air entrainment)

Briefing 2005-03 12

Took compensatory action – also took actions that were not wise. Big risk at start of outage – depends on operator mitigation for success. Risk is "considered" + then determined to be reasonably acceptable.

"Performance based rule" Manager use criteria not criteria. Has cannot be mitigate risk by specific and.

Potential Solutions

- Expand MD 8.3 to include shutdown risk
- Amend IMC 71111.20 to include one page checklist to screen high risk outages
- Write GL to understand how NUMARC 91-06 guidance has been recently implemented (identify poor performers)

Briefing 2003-03 13

- Events assessment generate data
- Inspection generate data
- Industry generates data.

Open - What sort of tool needs to be developed for SD risk?

Not sure need GL. Need to get better tools for inspectors.

Need to get back to industry on monitoring SD risk.

BS - Comm told staff to monitor - not write rule.
Cannot inspect against a monitoring program.
If risk has increased, then maybe we should re-visit rule.

C Grover - Is there really a trend to higher risk, or just a few stupid people.
Need a GL to gather data.

BS - Need a reg basis to issue GL.

LT will consider a develop path forward

Item #5 Handout #3

United States Nuclear Regulatory Commission
Advisory Committee on Reactor Safeguards

PREDECISIONAL

Operating Reactors Summary and Analysis for CY 2003 and 2004

May 1, 2005

By: John D. Sieber
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Operating Reactors Summary and Analysis
CY 2003 and 2004

By: John D. Sieber

Summary and Conclusions

An examination of recent operating events at currently operating Nuclear Power Reactors in the United States indicates continued progress in achieving a high level of operating performance and nuclear safety, with a few exceptions. In general, where the data derived form the Reactor Oversight Program and the Accident Sequence Precursor Program might indicate and increase in events or trends, taken in the perspective of the overall operating history of the industry, the occasional declining trend in a few parameters is not statistically significant compared to the overall operating history of the industry. Overall, the industry has achieved s remarkable increase in safety and reliability since the TMI-II accident in 1979.

Notwithstanding this major industry and regulatory achievement, a few operating events and adverse plant conditions continue to occur, albeit at a small rate, which deserve continuing attention. In addition, a few events, which could be classed as major precursors deserving special attention, have occurred in recent years. These events include:

- **Davis-Besse** - Degradation of the Reactor Coolant pressure boundary by corrosion of the reactor vessel head.

- **Perry** – Repetitive failure of the Emergency Service Water pump on two occasions, indicating ineffective corrective actions subsequent to the first failure. In addition, the inadequate venting of the Residual Heat Removal/Low Pressure Coolant Injection System (RHR/LPCI) keep fill system is an indication of a degraded multiple safety cornerstone.

- Point Beach – The licensee identified the potential for a common mode failure of the auxiliary feedwater (AFW) system pumps caused by inadequate operator actions in response to a loss of instrument air. About a year later, the licensee identified the potential for a common mode failure of the AFW system pumps from the plugging by debris of the pressure reduction orifices in the AFW system recirculation lines. In the March 4, 2003, Annual Assessment Letter to the licensee for Point Beach Nuclear Station, the NRC documented this crosscutting issue with Problem Identification and Resolution. . This was based on White Findings for the Safety Injection pump failure and deficiencies involving

Emergency Preparedness. In addition, four Green findings involving the flooding of manholes containing plant equipment, repeat problems with cold weather preparations, delayed maintenance rule action for the G05 gas turbine, and an inadequate extent of condition review when addressing a steam generator narrow range level detector problem were determined to be contributory to the degraded cornerstone assessment.

The Davis-Besse plant is under a special oversight program as part of the Manual Chapter 0350 process. The Perry and the Point Beach units were determined to be in the category of Multiple/Repetitive Degraded Cornerstone Column. In addition, twenty one (21) units are in the Regulatory Response Column, which is the first category of oversight, which requires increased NRC staff oversight.

Overall, slightly fewer than 25 percent of the operating nuclear units in the United States are the subject of increased staff oversight, due to less than desired operating and safety performance.

Sometimes, external conditions beyond the licensee's direct control occur which cause initiating events that place demands on the plant's safety systems. One such situation was the Northeast Blackout of August 14, 2003 that initiated Loss of Offsite Power (LOOP) events at nine (9) U. S. nuclear power plants, as well as some Canadian nuclear units. In this instance, the appropriate safety equipment and systems automatically responded as designed. This event did not introduce any complications into the LOOP recovery and did not reveal any significant flaws in the design or operation of the nuclear unit's emergency electrical equipment design or operation. Nonetheless, the event is a significant initiator that can challenge the safety systems of nuclear power plants. Since all appropriate emergency and safety systems at the effected plants operated as designed, there were no failure data or operating errors generated from this event and therefore, there are no historical data for analysis available from the Reactor Oversight Process.

In addition, there have been instances where non-safety equipment has failed at several nuclear power plants which have caused the plant to trip, but no complications have occurred, which challenged safety systems or revealed safety deficiencies. Such events as transformer failures, circuit card and instrument failures, circuit breaker failures, and similar non-safety equipment malfunctions sometimes occur but do not cause significant increases in risk because of the reliability of safety and mitigating systems. Non-safety related event initiators should logically be expected to increase as plants age, but there is no statistically significant evidence that the fleet of plants has reached the point

where aging of non-safety related equipment is initiating safety challenging events with a significantly greater frequency. However, our insights into the initiating event frequency would benefit from a more in-depth study of this potential causal factor.

Insights from the Reactor Oversight Process

The Reactor Oversight Process uses several data sources to derive a measure of plant safety that allows the NRC staff to allocate its inspection resources to the plants that demonstrate that increased regulatory attention is warranted. This section of this report examines the Inspection Matrix of findings and the associated Significance Determination Process results for those findings, and the performance indicator Data for each operating unit. Finally, the basic inputs form the Inspection and performance data and the significance determination results are evaluated in the ROP Action Matrix Summary. From the Action Matrix Summary, NRC staff management can assess the need for increased regulatory attention in the form of additional inspection hours and management attention and other regulatory responses, as appropriate.

In this analysis, the Inspection matrix, the performance indicator matrix and the action matrix are the principle sources of the data from which the conclusions of this paper are drawn. The use of these three ROP elements are necessarily chosen to represent a comprehensive report of the safety status of the fleet of nuclear power plants and to identify which of these plants deserves our increased attention.

Inspection Matrix

This summary of inspection findings, evaluated by risk significance as indicated by the action color assigned, identifies the most significant inspection findings over the previous 4 quarters of 2004. Plants having no significant inspection findings during this period have been excluded from this table.

Table 1

Inspection Findings Matrix for CY 2004

Plants	Initiating Events	Mitigating Systems	Barrier Integrity	Emergency Preparedness	Occupational Radiation Safety	Public Radiation Safety
Arkansas Nuclear 1		White (1)				
Brunswick		White (1)				

2						
Calvert Cliffs 2		White (1)				
Cooper		White (1)				
D.C. Cook 1						White (1)
D.C. Cook 2						White (1)
Davis-Besse		Yellow (1)				
Hope Creek 1	White (1)	White (1)				
Oyster Creek		White (1)		White (1)		
Oconee 1		White (1)				
Oconee 2		White (1)				
Oconee 3		White (1)				
Perry 1		White (2)				
Point Beach 1						
Point Beach 2						
Salem 1		White (1)				
Sequoyah 1		White (1)				
Surry 1		White (1)				
Surry 2		White (1)				
Vermont Yankee				White (1)		
Waterford 3		White (1)				

Overall, the industry had four (4) red findings, one (1) yellow finding and nineteen (19) white findings. The most significant of the findings are further evaluated in the discussion of the 4Q/2004 Action Matrix below.

Performance Indicator Summary

Performance indicator data are collected and tabulated on a quarterly basis for the following categories of performance, which are based on the seven ROP Cornerstones of Safety:

Initiating Events
Unplanned Scrams per 7000 Critical Hours (IE01)
Scrams with loss of Normal Heat Removal (IE02)
Unplanned Power Changes (IE03)

Mitigating Systems
Emergency AC Power System (MS01)
High Pressure Injection System (MS02)
Heat Removal System (MS03)
Residual Heat Removal System (MS04)
Safety System Functional Failures (MS05)

Boundary Integrity
Reactor Coolant System Specific Activity (BI01)
Reactor Coolant System Leakage (BI02)

Emergency Planning
Drill/Exercise Performance (EP01)
ERO Drill Participation (EP02)
Alert and Notification System (EP03)

Occupational Radiation Exposure Control
Occupational Exposure Control Effectiveness (OR01)

Public Radiation Control
RETS/ODCM Radiological Effluents (PE01)

Physical Protection
None

Table 2

4Q/2004 ROP Performance Indicators Summary

Plants	IE 01	IE 02	IE 03	MS 01	MS 02	MS 03	MS 04	MS 05	BI 01	BI 02	EP 01	EP 02	EP 03	OR 01	PR 01
D.C. Cook 2		W													
Davis-Besse													W		

Fermi 2				W											
Fort Calhoun				W											
Robinson 2										W					
San Onofre 2		W													

In the above table 3 summary, we note that Scrams with Loss of Normal Heat Sink (e.g. loss of condenser vacuum) occurred twice, loss of the high pressure injection system occurred twice, increased Reactor Coolant System Leakage occurred once, and inoperability or malfunction of the EP Alert and Notification System occurred once.

4Q/2004 ROP Action Matrix Summary

The assessment program collects information from inspections and performance indicators (PIs) in order to enable the agency to arrive at objective conclusions about the licensee's safety performance. Based on this assessment information, the NRC determines the appropriate level of agency response, including supplemental inspection and pertinent regulatory actions ranging from management meetings up to and including orders for plant shutdown. The Action Matrix Summary listed below reflects overall plant performance and is updated regularly to reflect inputs from the most recent performance indicators and inspection findings. Notes have been added to some plants to explain the reasons that these plants have changed Action Matrix columns from the previous quarter. This page will be updated as necessary to reflect changes in licensee performance.

Table 3

Action Matrix Summary 4Q/2004

Regulatory Response Column	Multiple/Repetitive Degraded Cornerstone Column
Arkansas Nuclear 1[1]	Perry 1[2]
Brunswick 2[3]	Point Beach 1[4]

Calvert Cliffs 2[5]	Point Beach 2[6]
Cooper[7]	
D.C. Cook 1[8]	
D.C. Cook 2[9]	
Fermi 2[10]	
Fort Calhoun[11]	
Hope Creek 1[12]	
Oconee 1[13]	
Oconee 2[14]	
Oconee 3[15]	
Oyster Creek[16]	
Robinson 2[17]	
Salem 1[18]	
San Onofre 2[19]	
Sequoyah 1[20]	
Surry 1[21]	
Surry 2[22]	
Vermont Yankee[23]	
Waterford 3[24]	

Note 1:	**ANO 1 is in the regulatory response column due to one white finding in the mitigating systems cornerstone originating in 2Q/2004.**
Note 2:	**Perry unit 1 is in the multiple/repetitive degraded cornerstone column due to the mitigating systems cornerstone being degraded with multiple white findings for greater than 4 consecutive quarters. In particular, the ESW pump failure finding from 3Q/2003 was held open in accordance with IMC 0305 for greater than 4 quarters because corrective actions were ineffective and the pump failed again in May 2004. This finding, in conjunction with the 4Q/2003 finding involving inadequate venting of the RHR/LPCI keep fill system, which is also being held open in accordance with IMC 0305 for greater**

	than 4 quarters pending the implementation of effective corrective actions to address performance deficiencies, resulted in greater than 4 consecutive quarters in the degraded cornerstone column and placed the plant in the multiple/repetitive degraded cornerstone column.
Note 3:	Brunswick unit 2 is in the regulatory response column due to one white finding in the mitigating systems cornerstone originating in 1Q/2004.
Note 4:	Point Beach unit 1 is in multiple/repetitive degraded cornerstone column due to one red finding and one yellow finding in the mitigating systems cornerstone originating in 1Q/2002 and 1Q/2003, respectively. Both findings are being held open for greater than four quarters in accordance with IMC 0305 pending the implementation of effective corrective actions to address performance deficiencies.
Note 5:	Calvert Cliffs unit 2 is in the regulatory response column due to one white finding in the mitigating systems cornerstone originating in 2Q/2004.
Note 6:	Point Beach unit 2 is in the multiple/repetitive degraded cornerstone column due to two red findings in the mitigating systems originating in 1Q/2002 and 1Q/2003, respectively. Both findings are being held open for greater than four quarters in accordance with IMC 0305 pending implementation of effective corrective actions to address performance deficiencies.
Note 7:	Cooper Nuclear Station is in the regulatory response column due to one white finding in the mitigating systems cornerstone originating in 1Q/2004.
Note 8:	DC Cook unit 1 is in the regulatory response column due to one white finding in the public radiation safety cornerstone originating in 1Q/2004.
Note 9:	DC Cook unit 2 is in the regulatory response column due to one white performance indicator in the initiating events cornerstone originating in 3Q/2002 and one white finding in the public radiation safety cornerstone originating in 1Q/2004.
Note 10:	Fermi unit 2 is in the regulatory response column due to one white performance indicator in the mitigating systems cornerstone originating in 3Q/2003.
Note 11:	Fort Calhoun is in the regulatory response column due to one white performance indicator in the mitigating systems cornerstone originating in 3Q/2004.
Note 12:	Hope Creek is in the regulatory response column due to one white finding in the mitigating systems cornerstone originating

	in 1Q/2004.
Note 13:	Oconee unit 1 is in the regulatory response column due to one white finding in the mitigating systems cornerstone originating in 3Q/2004.
Note 14:	Oconee unit 2 is in the regulatory response column due to one white finding in the mitigating systems cornerstone originating in 3Q/2004.
Note 15:	Oconee unit 3 is in the regulatory response column due to one white finding in the mitigating systems cornerstone originating in 3Q/2004.
Note 16:	Oyster Creek is in the regulatory response column due to one white finding in the mitigating systems cornerstone originating in 1Q/2004.
Note 17:	Robinson unit 2 is in the regulatory response column due to one white performance indicator in the barrier integrity cornerstone originating in 4Q/2004.
Note 18:	Salem unit 1 is in the regulatory response column due to one white finding in the mitigating systems originating in 1Q/2003.
Note 19:	San Onofre unit 2 is in the regulatory response column due to one white performance indicator in the initiating events cornerstone originating in 2Q/2004.
Note 20:	Sequoyah unit 1 is in the regulatory response column due to one finding in the mitigating systems cornerstone originating in 3Q/2004.
Note 21:	Surry unit 1 is in the regulatory response column due to one white finding in the mitigating systems cornerstone originating in 2Q/2004.
Note 22:	Surry unit 2 is in the regulatory response column due to one white finding in the mitigating systems cornerstone originating in 4Q/2004.
Note 23:	Vermont Yankee is in the regulatory response column due to one white finding in the emergency preparedness cornerstone originating in 4Q/2004.
Note 24:	Waterford unit 3 is in the regulatory response column due to one white finding in the mitigating systems cornerstone originating in 2Q/2004.

Last modification: Mar 04, 2005

Multiple/Repetitive Degraded Cornerstone Column

NRC uses the Reactor Oversight Process (ROP) to measure plant performance within the three broad areas of (1) reactor safety, (2) radiation safety, and (3) security. Within these areas, NRC looks at seven cornerstones: (1) Initiating Events, (2) Mitigating Systems, (3) Barrier Integrity, (4) Emergency Preparedness, (5) Occupational Radiation Safety, (6) Public Radiation Safety, and (7) Physical Protection.

If, from an assessment of inspection findings and performance indicators for a nuclear plant, NRC identifies repetitive degradation in a single cornerstone, NRC documents this plant degradation on an ROP matrix in the Multiple/Repetitive Degraded Cornerstone column. As a result of several failures associated in the cornerstone for Mitigating Systems at Point Beach Nuclear Power Station Units 1 and 2, NRC gave this plant such a rating and, therefore, will conduct a diagnostic inspection.

Enforcement Actions

Point Beach Summary

As discussed at the April 2003 Agency Action Review Meeting (AARM), NRC staff decided to conduct an inspection at Point Beach Nuclear Power Station to determine the breadth and depth of the licensee's performance deficiencies. NRC will conduct this inspection in addition to the baseline inspections already scheduled.

First Auxiliary Feedwater Issue

Licensee Report. On November 29, 2001, the licensee reported to the NRC the potential for a common mode failure of the auxiliary feedwater (AFW) system pumps caused by inadequate operator actions in response to a loss of instrument air.

NRC Inspections and Action. For this issue, NRC staff conducted a Special Inspection from December 3, 2001, through February 28, 2002. Inspectors identified that procedures for the reactor operators were inadequate and had been for many years and that the licensee had seven prior opportunities to identify these inadequacies. Failure to provide adequate procedures and failure to take appropriate corrective actions are both violations of NRC regulatory requirements. In accordance with NRC's Significance Determination Process, NRC preliminarily determined that these violations constituted an issue with high safety significance (that is,

a Red finding). The issue has high significance because a common mode failure of AFW system pumps would substantially reduce the operators' capability for safely shutting down the plant in response to certain accidents.

On July 12, 2002, the NRC determined that the potential for a common mode failure of the AFW system pumps caused by a loss of instrument air was a Red finding.

Licensee Corrective Action. The licensee took prompt corrective actions to revise procedures and train operators to address the immediate safety concerns associated with the issue. Additionally, the licensee installed additional equipment to improve the safety of the AFW system design.

Second Auxiliary Feedwater Issue

Licensee Report. On October 29, 2002, the licensee notified the NRC of a potential for a common mode failure of the AFW system pumps from the plugging by debris of the pressure reduction orifices in the AFW system recirculation lines.

NRC Inspections and Action. NRC conducted a second special inspection from October 31, 2002, through March 24, 2003.

During development of modification packages in 1999, the licensee recognized the potential for these orifices to plug. However, because of the lack of full understanding of the AFW system design basis, the licensee installed the orifices. NRC found that in late 2001 and early 2002, the previous AFW system issue, associated with instrument air, presented an opportunity for the licensee to correct this lack of understanding, but no action was taken until the orifice for the "A" motor-driven AFW pump was found partially plugged on October 24, 2002, after post-maintenance testing of the pump. In February 2003, the licensee had an independent laboratory conduct tests that demonstrated that the orifices would quickly plug when subjected to water-borne debris similar to that found in the licensee's service water system.

NRC determined that the finding of the orifice partially plugged for the "A" motor-driven AFW system pump on October 24, 2002, after post-maintenance testing of the pump was a preliminary Red finding which is pending final significance determination.

On December 11, 2003, the NRC determined that the potential for a common mode failure of the AFW system pumps caused by plugging of the orifices was a Red finding.

2002 Performance Assessment Letter

NRC Report. The licensee for the Point Beach Nuclear Station had a substantive crosscutting issue in the area of Problem Identification and Resolution.

NRC Inspections and Action. In the March 4, 2003, Annual Assessment Letter to the licensee for Point Beach Nuclear Station, the NRC documented this crosscutting issue with Problem Identification and Resolution. This was based on the White Findings, (indicated on the plant's Reactor Oversight Process matrix) for the Safety Injection pump failure and deficiencies involving Emergency Preparedness in addition to four Green findings. The four Green findings involved the flooding of manholes containing plant equipment, repeat problems with cold weather preparations, delayed maintenance rule action for the G05 gas turbine, and an inadequate extent of condition review when addressing a steam generator narrow range level detector problem.

As a result of this final Red finding and the discussion at the AARM meeting, NRC will conduct its 95003 supplemental inspection.

95003 Inspection Report

The NRC Inspection Procedure 95003 supplemental inspection was conducted at Point Beach from late-July to mid-December 2003 to review the two AFW issues. It was conducted in three phases: corrective actions, emergency preparedness, and engineering; and involved inspectors from all four NRC Regional Offices and Headquarters. In general, the inspection identified 1) ineffective implementation of the corrective action program, 2) emergency preparedness program weaknesses, 3) engineering design control issues, and 4) operations/engineering interface issues. Specifically, 11 low-level Non-Cited Violations and 1 potential high-level violation were identified. The high-level violation involved unauthorized changes made by the licensee to its emergency preparedness emergency action level scheme.

Predecisional Enforcement Conference

On January 13, 2004, the NRC conducted a predecisional enforcement conference with Point Beach to review the violation identified during the

95003 inspection involving the unauthorized changes to the emergency action level scheme. A summary of that conference was documented in a letter dated January 27, 2004. As a result of NRC deliberations on this issue, the NRC issued a Notice of Violation and proposed the imposition of a $60,000 civil penalty in a letter to the licensee, dated March 17, 2004.

Meeting with the EDO

On February 20, 2004, the NRC's Executive Director for Operations (EDO) and other NRC representatives met with Point Beach management in Manitowoc, Wisconsin to discuss recent performance. A summary of this meeting was documented in a letter dated March 11, 2004.

2003 Performance Assessment Letter

On March 4, 2004, the NRC issued its Annual Assessment Letter to Point Beach. This letter summarized the NRC's assessment of Point Beach performance during 2003. Point Beach remained within the Multiple/Repetitive Degraded cornerstone column of the Action Matrix based on the Red finding for Unit 1 and Unit 2 for the first AFW issue and the Yellow finding for Unit 1 and the Red finding for Unit 2 for the second AFW issue. Additionally, the NRC identified substantive cross-cutting issues in the areas of human performance and problem identification and resolution.

Confirmatory Action Letter

To address the problems identified during the 95003 inspection and problems identified through self-assessments, the licensee committed to the NRC to complete specific individual steps and action plans in its overall performance improvement "Excellence Plan." These commitments were documented in a letter from the licensee to the NRC, dated March 22, 2004. The NRC then incorporated these commitments in a Confirmatory Action Letter that was issued to the licensee on April 21, 2004. Extra inspections and expanded routine baseline inspections will be conducted in 2004 and 2005 as part of the NRC's follow-up on how the licensee meets these commitments. The revised commitments were submitted in an updated Excellence Plan dated April 1, 2004.

Emergency Action Level Civil Penalty

In a letter dated April 8, 2004, the licensee paid the emergency action level $60,000 civil penalty. As of January 16, 2004, the licensee had corrected the unauthorized changes. The licensee is also planning to revise and upgrade, later this year, the existing emergency action levels to a more current, NRC-approved industry scheme.

Davis-Besse

On April 21, 2005, a Notice of Violation and Proposed Imposition of Civil Penalties in the amount of $5,450,000 was issued for multiple violations (some willful) related to the significant degradation of the reactor pressure vessel head identified in February and March 2002. The significant violations included (1) operation with reactor coolant system pressure boundary leakage (associated with a Red SDP finding, $5,000,000), (2) failure to provide complete and accurate information (Severity Level I, $110,000), (3) failure to promptly identify and correct a significant condition adverse to quality (Severity Level II, $110,000), (4) failure to implement procedures (Severity Level II, $110,000), (5) failure to provide complete and accurate information (Severity Level I, $120,000), (6) failure to promptly identify and correct a significant condition adverse to quality (associated with a Red SDP finding), (7) failure to implement procedures (associated with a Red SDP finding), and (8) failure to provide complete and accurate information (Severity Level III).

Watts Bar

On April 11, 2005, a Notice of Violation was issued for a violation associated with a White SDP finding involving the licensee's failure to promptly identify and correct silt blockage of the essential raw cooling water (ERCW) line to the 1A-A centrifugal charging pump (CCP). The violation cited the licensee's failure to establish measures to assure that conditions adverse to quality, such as failures and malfunctions, are promptly identified and corrected, as required in 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions."

Arizona Public Service Company (Palo Verde) EA-05-051

On April 8, 2005, a Notice of Violation and Proposed Imposition of a Civil Penalty in the amount of $50,000 was issued for a Severity Level III violation involving the licensee's failure to perform a written safety evaluation and obtain Commission approval prior to making a procedural change which resulted in a change to the facility as described in the Updated Final Safety Analysis Report that increased the probability of a

malfunction of equipment important to safety previously evaluated in the safety analysis report.

Arizona Public Service Company (Palo Verde) EA-04-221

On April 8, 2005, a Notice of Violation was issued for a violation associated with a Yellow SDP finding involving a failure to maintain portions of the emergency core cooling system (ECCS) filled with water in accordance with design control requirements. The violation cited the licensee's failure to establish adequate design control measures to assure that the design basis for the ECCS was appropriately translated into specifications, procedures, and instructions.

FirstEnergy Nuclear Operating Company (Perry) EA-04-214

On March 29, 2005, a Notice of Violation was issued for a violation associated with a White SDP finding involving the failure to follow the requirements of the Perry Emergency Plan during an event that was classified at the Alert level. The violation cited the licensee's failure to properly implement the required standard emergency classification and action level scheme.

On February 24, 2005, the NRC issued a Severity Level III Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $55,000 for violation of NRC's employee protection regulations by a licensee contractor, Williams Power Corporation, at the Perry site.

AmerGen Energy Company, LLC (Oyster) EA-04-213

On March 1, 2005, a Notice of Violation was issued for violations associated with a White SDP finding involving untimely actions to change an Emergency Action Level threshold value used to declare a General Emergency or a Site Area Emergency and revise supporting emergency procedures. The violations cited the licensee's failure to maintain an emergency classification and action level scheme and the failure to properly implement the configuration change process in accordance with the Technical Specifications.

Entergy Nuclear Operations, Inc. (Vermont Yankee) EA-04-173

On February 2, 2005, a Notice of Violation was issued for a violation associated with a White SDP finding involving the failure to issue tone alert radios to the entire populace within the emergency planning zone (EPZ). The violation cited the licensee's failure to follow its emergency plan to

establish the means to provide early notification and clear instruction to the populace within the plume exposure pathway EPZ.

Tennessee Valley Authority (Sequoyah Unit 1) EA-04-223

On January 26, 2005, a Notice of Violation was issued for a violation associated with a White finding involving binding problems with the breaker mechanism operated cell slide assembly for the 1A Residual Heat Removal pump. The violation cited the licensee's failure to correct conditions adverse to quality based on the identification of binding problems during previous surveillance testing.

Duke Energy Corporation (Catawba Units 1 and 2) EA-04-189

On January 24, 2005, a Notice of Violation was issued for a Severity Level III violation involving the failure to provide complete and accurate information involving a proposed amendment to allow the radiation of four mixed oxide lead test assemblies.

FPL Energy Seabrook, LLC (Seabrook Station) EA-04-139

On December 2, 2004, a letter was issued documenting the NRC's decision to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy and refrain from issuing enforcement action for a violation involving the licensee's failure to seek prior NRC approval in accordance with the 10 CFR 50.59 rule in effect at the time, involving a design change to the turbine building. Enforcement discretion was appropriate based on the fact that even though this issue was a violation of the "old" 10 CFR 50.59 rule, it would not have been a violation of the "new" 10 CFR 50.59 rule.

Nuclear Management Company, LLC (Point Beach) EA-04-038

On September 29, 2004, a letter was issued documenting the NRC's decision to exercise enforcement discretion in accordance with Section VII.B.5 of the Enforcement Policy for a violation involving discrimination against a contractor worker for raising safety concerns at the Point Beach Plant. Discretion was warranted to encourage prompt correction of violations, a safety conscious work environment, and resolution of employment discrimination issues without the intervention of the NRC. The NRC also issued enforcement discretion to the contractor, Day and Zimmerman Nuclear Power Systems.

Day and Zimmerman Nuclear Power Systems (Point Beach) EA-04-105

On September 29, 2004, a letter was issued documenting the NRC's decision to exercise enforcement discretion in accordance with Section VII.B.5 of the Enforcement Policy for a violation involving discrimination against a contractor worker for raising safety concerns at the Point Beach Plant. Discretion was warranted to encourage prompt correction of violations, a safety conscious work environment, and resolution of employment discrimination issues without the intervention of the NRC. The NRC also issued enforcement discretion to the Nuclear Management Company.

American Electric Power Company (D.C. Cook) EA-04-109

On September 29, 2004, a Notice of Violation was issued for a Severity Level III violation involving an application for renewal of a Senior Reactor Operator license that was not complete and accurate in all material respects.

Duke Energy Corporation (Oconee) EA-04-115

On September 24, 2004, a Notice of Violation was issued for a violation associated with a White SDP finding involving inconsistent fire response procedures that could result in the failure to maintain pressurizer level within the required indicating range. The violation cited the licensee's inadequate fire response procedures.

Virginia Electric and Power Company (Surry) EA-04-005

On September 15, 2004, a Notice of Violation was issued for a violation associated with a White SDP finding involving ineffective safe shutdown procedures during a postulated fire that could have resulted in a reactor coolant pump seal loss of coolant accident. The violation cited the licensee's ineffective alternative shutdown capability and response procedures for a postulated fire in the Emergency Switchgear Room Number 1 and 2.

Nebraska Public Power District (Cooper) EA-04-120

On June 25, 2004, a Notice of Violation was issued for a violation associated with a White SDP finding involving a high failure rate on the licensed operator biennial requalification written examinations. The violation cited the failure to consistently implement all elements of a systems approach to training in the licensed operator requalification program.

Carolina Power and Light Company (Brunswick Steam Electric Plant, Unit 2) EA-04-076

On June 2, 2004, a Notice of Violation was issued for a violation associated with a White SDP finding involving the failure to take adequate corrective action for conditions adverse to quality associated with the No. 3 emergency diesel generator (EDG 3) jacket water cooling (JWC) system. The corrective maintenance performed to stop a pipe coupling leak on the JWC supply line to the turbo charger for EDG 3 failed to correct the leak. The violation also cited the failure to comply with Technical Specification 3.8.1, AC Sources Operating, because due to the ongoing leak the EDG 3 was inoperable while the plant was in Mode 1 for a period in excess of seven days.

Industry Trends

The NRC initiated an Industry Trends Program (ITP) to monitor trends in indicators of industry performance as a means to confirm that the safety of operating power plants is being maintained. Should any long-term indicators show a statistically significant adverse trend, the NRC will evaluate them and take appropriate regulatory action using its existing processes for resolving generic issues and issuing generic communications. The NRC formally reviews these indicators as part of the Agency Action Review Meeting (AARM) each year, and any adverse trends are reported to Congress in the NRC's Performance and Accountability Report.

No statistically significant adverse trends have been identified to date, based on level or declining long-term trends in the indicators developed by the former NRC Office for Analysis and Evaluation of Operational Data (AEOD) and the Accident Sequence Precursor (ASP) program.

Item # 6 Handout 1

Advisory Committee on Reactor Safeguards

Proactive Initiative Safety Management

05/05/2005

ACRS Process

- Proactive – Options for Committee consideration
- Reactive – Review NRC Staff's response to Commission's SRM dated August 2004 (SRM-04-0111)

Proactive Initiative: Options for Consideration

- Hold Workshop to examine events and experience
- Hold Workshop to assess analytical techniques
- Review application of tools to assess NRC's safety culture
- Identify ways to enhance safety culture (internal/external)
- Perform Independent Data Analysis

Reactive: Review Staff's Response to SRM

Three major areas of NRC activity:

(1) Improve the ROP treatment of cross-cutting areas to more fully address Safety Culture:

- Human Performance
- Problem Identification and Resolution
- Safety Conscious Work Environment

(2) Develop a process for determining the need for a safety culture evaluation for plants in the degraded cornerstone columns

(3) Develop a process for conducting safety culture reviews (i.e., for plants with degraded cornerstones), including guidance and training

.

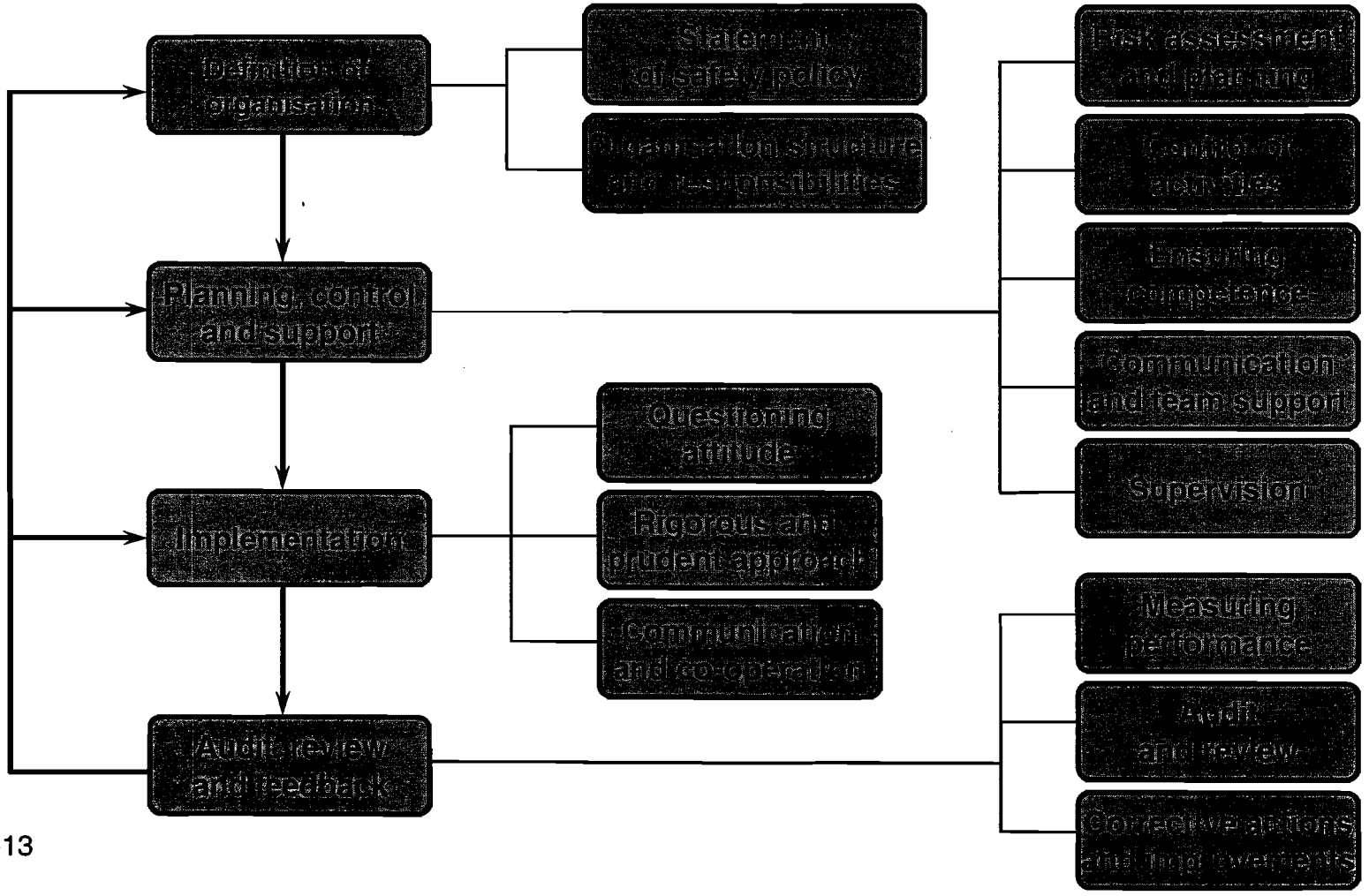
NRC's Preliminary Schedule

- Staff's Response Plan, Spring 2005
- Stakeholder Interactions, (throughout development process beginning Sept. 2005)
- Methodology for use in ROP, Dec. 2006
- Commission Paper, Dec. 2006

Safety Management

- The safety management system comprises those arrangements made by the organization for the management of safety in order to promote a strong safety culture and achieve good safety performance. – INSAG-13 (10/99)

Components of Safety Management (Plant Operation)



INSAG-13

Recommendation on Use of Terminology

- Safety management when focus is on the system and arrangements made by the organization for the management of safety.

- Safety culture when focus is on the output characteristics and attitudes in the organization or individuals .

Proactive Initiative Options for Consideration

1. Workshop to examine significant events and associated experience
2. Workshop to assess analytical techniques and application
3. Advise on application of tools to assess NRC's safety culture
4. Enhance safety culture (internal and external)
5. Perform data analysis

Safety Management Initiative 1

- Workshop to examine significant events for a "common thread"
  - Leading indicators
  - Lessons learned
  - Corrective actions
  - Generic applicability

Safety Management Initiative 2

- Workshop that seeks to advance analytical methods (technical workshop):
  - HRA techniques
  - Performance Indicators
  - Formal decision making

Safety Management Initiative 3

- NRC Safety Culture:
  - Examine NRC Baseline
  - Examine process for conducting safety culture reviews
  - Solicit Stakeholder Feedback
  - Advise Commission - information for future use

Safety Management Initiative 4

- Enhance Safety Culture (internal and external)
  - Role of ACRS
    - Promote Questioning Attitude
    - Organizational Learning
  - Working Group – explore other government advisory groups, models for insights

Safety Management Initiative 5

- Data Collection and Analysis – relationship of data to changes in safety culture, e.g.,
  - Recurring events and conditions
  - Human error induced failures
  - Common cause failures

Proposed Selection Criteria

- Is the work:
  - Consistent with the Commission's Strategic Plan
  - Proactive and not being done by others
  - Asking important questions (see key questions)
  - Generating needed information

ACRS-Proactive Initiative

Consistent with the Commission's Strategic Plan:

- Keep abreast of new technologies and opportunities (safety strategy)
- Enhance NRC process and products by supporting the use of good science (effectiveness strategy)
- Ensure excellence in Agency Management (management strategy)

Key Questions

- Do significant events have a common underlining causes that could be highlighted as a leading indicator of safety management shortcomings?
- Can available analytic techniques be extended and used to assess the impact of safety management?
- Are the tools used to assess NRC's safety culture capturing all the information necessary to improve internal safety culture?
- Can ACRS/ACNW do more to promote safety culture both internally and externally?
- Can different data bases be brought together to draw new insights on safety management?

Potential Output Information

- White Papers
  - Clarification of issues, and pathway for resolution
  - Identification of limitations, uncertainties
  - Basis for decisions and recommendations
- Workshop or Working Group
  - Draw insights from different perspectives (poke and probe)
  - Prioritize new information
  - Draw insights from synergism of participants
  - Generate new information on latest developments

Safety Management Initiative: Options for Consideration

- Hold Workshop to examine events and experience
- Hold Workshop to assess analytical techniques
- Review application of tools to assess NRC's safety culture
- Identify ways to enhance safety culture (internal/external)
- Perform Independent Data Analysis

Backup Slides

Safety Culture

- "Safety culture is that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance."
  - NSAG-4

Commission's SRM on Safety Culture (1)

- Develop tools that allow inspectors to rely on more objective findings
- Use findings and indicators already available
- Enhance problem identification and resolution initiatives
- Enhance Inspector training
- Use insights from INPO and international community for training development
- Develop a process to determine the need for safety culture evaluation (i.e., for plants with degraded cornerstones)
- Develop a process for conducting safety culture reviews.

Commission's SRM on Safety Culture (2)

- Monitor industry efforts
- Monitor foreign regulators
- Involve stakeholders
- Encourage licensees
- Not use surveys of licensee personnel
- Consider using "safety management rather than "safety culture"

Role of the Regulator (K.Astrand, STUK)

1. The regulatory body should maintain a high-level safety culture in its own organization.

2. The regulatory body should maintain national safety culture in the interaction between the regulator and the licensees.

3. The regulatory body should be able to evaluate the safety culture level in the licensee organization, and notice the weak signals of change.

However, in regulatory oversight the emphasis should be on Safety Management, not on culture.

Item 9 Handout #1

STEAM GENERATOR TUBE INTEGRITY PROGRAM

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522nd ACRS Meeting
May 6, 2005

Overview

- Research is Being Conducted in a Variety of Areas
- NRR has Requested Research Related to ISI Capabilities, ISI Reliability, and Rupture, Burst, and Leakage Models
- NRR will use this Information in the Review of Licensee Submittals and to Provide Guidance to Regional Inspectors
- In Response to ACRS Feedback, Additional Work is Being Conducted on Crevice Chemistry and the Relative Susceptibility of Various Tube Materials to Cracking

OUTLINE OF PRESENTATION

- Task 3 - Tube Integrity and Integrity Predictions
  - Objective
  - Failure Models
  - Leak Rate Models
  - Pressurization Rate Testing
  - Secondary side Depressurization Study
  - Constant Pressure Crack Growth Study
  - Statistical Treatment of Models
  - Summary of Results
  - Future Work on Tube Integrity
- Task 1 - Assessment of Inspection Reliability
- Task 2 - ISI Technology
- Task 4 - Degradation Modes
- Conclusion

Task 3 - Tube Integrity and Integrity Predictions

- Objective – to Evaluate and Validate Models for Leak/Rupture Behavior, Failure Pressures, Leak Rates for Degraded SG Tubes – Normal and Accident Conditions

Tube Integrity and Integrity Predictions (Cont)

- SG Tube Materials are very Ductile
  - Failure Under Design Basis Conditions is by Plastic Instability
  - Failure Under Severe Accident Conditions is by Creep and/or Plastic Instability
- Real Cracks have Complex Shapes
  - Bounding Equivalent Rectangular Crack Method can give Conservative Results
  - Methods Developed for Realistic Prediction of Ligament Rupture
  - Effort is Ongoing to Develop more Realistic Predictions of Burst

Tube Failure Models

- The Model for Predicting the Pressure to Cause Plastic Collapse of a Tube Containing a Through Wall Axial Crack is the Erdogan Model:

  $P_{cr} = \sigma h/mR_m = P_b/m$

  σ = flow Stress

  h = tube wall thickness

  R_m = mean radius of the tube

  P_b = failure pressure of the unflawed tube

  m = constant related to the flaw size and geometry (Computed from LEFM Model of Erdogan)

Tube Failure Models (Cont)

- For Part Throughwall Axial Cracks, the Pressure Required to Fail the Radial Ligament is Given by:

$$P_{sc} = \sigma h/m_p R_m = P_b/m_p$$

$$m_p = (1 - a/mh)/(1 - a/h)$$

$$a = \text{crack depth}$$

if $P_{cr} > P_{sc,}$ the throughwall crack is stable

- M_p is a Measure of the Stress Magnification in the Ligament; Useful Characterization of the Severity of a Crack for both Design Basis and Severe Accident Conditions

Tube Failure Models (Cont)

- The Equations for P_{cr} and P_{sc} Underestimate the Ligament Rupture Pressures for Short and Deep Cracks
- ANL Proposed the Following:

$$m_p = [1 - \alpha(a/mh)]/(1 - a/h)$$

$$\alpha = 1 + \beta(a/h)^2 (1 - 1/m)$$

β is a constant ≈ 1

ANL Modification Predicted Better for Short and Deep Cracks

Tube Failure Models (Cont)

- Equivalent Rectangular Crack Method – Rectangular Cracks have been Considered up to this Point – Actual Cracks may not be Rectangular and may Contain Ligaments
- For Complex Cracks, use the Equivalent Rectangular Crack Method – Crack Depth Profile Determined by Eddy Current or Fractography
- Series of Equivalent Rectangular Cracks Selected and the one with the Lowest Ligament Rupture Pressure (highest m_p) is Selected

Modeling for Predicting Rupture Pressure, Leak Rate, and Burst Pressure

- Equivalent Rectangular Crack Models Give Reasonable Results for Initial Ligament Rupture, but do not Predict well Subsequent Tearing of the Remaining Ligament Under Increasing Pressure or the Final Burst Pressure

Models for Circumferential Cracks and Severe Accidents

- Models for Circumferential Cracks
  - Models have been Developed – Using Fracture Mechanics Approach Instead of Plastic Instability
  - Model Correlates with TW EDM Laboratory Results
- Models for Severe Accidents
  - Creep Rupture Model (Combined with ANL m_p and Linear Damage Rule) Predicts Failure Temperatures more Accurately than flow Stress Model

Simple Orifice Model

- The Leak Rate Model based on Simple Orifice Flow Through a Crack with an Opening Area A is:

$$Q = C_d A \sqrt{(2\Delta p/\rho)}$$

C_d = coefficient of discharge = 0.6

Δp = is the pressure differential

ρ = mass Density of water

Simple Orifice Model

- For an Axial Crack:

  $A = 2\pi(c_e)^2 V_o \sigma/E$

  c_e = function of c, $(\sigma/\sigma_y)^2$, tube mean radius, tube wall thickness

  Where c is half the crack length

  V_o = function of c_e, the tube mean radius and tube wall thickness

  E = modulus of elasticity

Simple Orifice Model (Cont)

- Tests Show that due to Short Transit time Across the SG wall, Leaks over a Range of Crack Sizes can be Described by a Single Phase Orifice flow Model with an Opening Based on the Cracking Opening Area
- The Leak rate is a Function of L/D where L is the Crack Length and D is 2 times the Crack Opening
- Good Agreement for Slits, Orifices, and Open Cracks
- Models tend to Overestimate Leak Rates for Actual Cracks Because Remaining Ligaments the Crack Opening Area and Meandering Crack Paths Increase L

Simple Orifice Model (Cont)

- SCC tend to Undergo Incremental Ligament Rupture with Increasing Pressure Before Cracks Become Unstable Which Would Cause Leakage at Pressures Lower than Predicted
- The Equivalent Crack Method had been Generalized to Predict Incremental Ligament Rupture After Initial Ligament Rupture
- Predictions Based on Fractography Tend to be more Accurate than Those Based on EC

Test Facilities

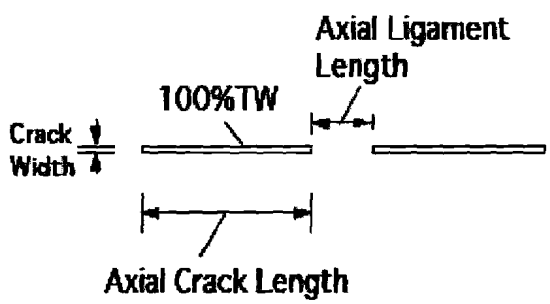
- Room-Temperature, High-Pressure Test Facility
  - Maximum Pressure - 7500 psi
  - Maximum Leak Rate – 12.8 gpm
  - Maximum Volume – Unlimited
- High Temperature, Pressure, and Leak Rate Test Facility
  - Maximum Temperature – 650 F
  - Maximum Pressure – 3000 psi
  - Maximum Leak Rate – 400 gpm
  - Maximum Volume – 200 Gallons

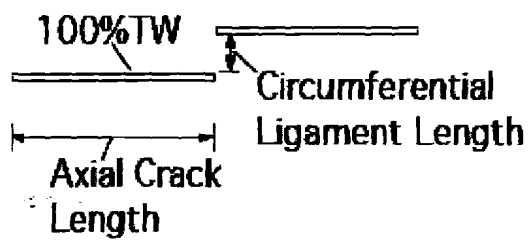
Testing - Pressurization Rate Effects

- Industry may Conduct Burst Tests to Demonstrate Adequate Margin as part of Condition Monitoring
  - Tests Conducted as part of Assessment of a Flaw at a Domestic Plant Suggested that There is a Pressurization rate Effect on Ligament Rupture Pressure
- Determination of rate Effect Inconclusive Since Protocols for fast and slow rate Tests Differed
  - Slow Rate Tests Conducted in 2 Steps, no Bladder and Foil Until Ligament Rupture, then Bladder and Foil Until Unstable Burst Pressure Reached
  - Fast Rate Tests Conducted with foil and Bladder from the Beginning
- Specimen to Specimen Geometry Variations Could Account for a Major Part of the rate Effect

Pressurization Rate Effects (Cont)

- Specimens Containing 1 inch Rectangular and Trapezoidal Notches 80-90% TW, and two 0.5 inch 80% TW Flaws Separated by a 0.05 inch Axial Flaw or a 0.05 inch Circumferential Ligament were Tested at Quasi-Steady State, 1000, 2000, 6000, and >10,000 psi/s
- No Effect of Pressurization Rate up to 6000 psi/s
- 2000 psi/s is the Maximum Industry rate





Secondary Side Depressurization Study

- Typical Analyses of Depressurization Events did not Consider the Bending Loads Imposed on a tube by the TSP when it is Locked to the Tubes by Corrosion Products
  - Concern with Dynamic Loads Raised by ACRS
- RES Calculated Dynamic Loads on TSP with RELAP5 and Benchmarked Against Experiments
  - Large SLB Produces Greater Pressure Drop than Small SLB or FWLB
  - Pressure Loading Acting on TSPs Transferred to Tubes Locked by Corrosion Products and Deposits

Secondary Side Depressurization Study (Cont.)

- Detailed FEA and Fracture Mechanics Analysis were Carried out for Model 51 SG TSPs and tubes
  - Loads on Tubes are Primarily Axial
  - Dynamic Loads have Virtually no Effect on Failure of Tubes with Axial Cracks
- If only one or two Tubes are Locked, the Stresses on the Locked Tubes Exceeds the Ultimate Tensile Strength
  - Because Displacements are Limited, Unflawed Tubes Would not Rupture, but the Tolerance for Circumferential Cracks Would be Severely Limited
  - If > 1.5% of Tubes are Locked, the Maximum Axial load is < 3 Kips and TW Circumferential Cracks < 180° are Stable

Constant Pressure Crack Growth Study

- Objective
  - Determine the Influence of Flaw Geometry on Flaw Tearing and Subsequent Leak Rate Behavior
  - Determine the Mechanism for Flaw Growth and Increases in Leak Rates at Constant Pressure
  - All Testing Conducted using the Room Temperature-High Pressure Test Facility

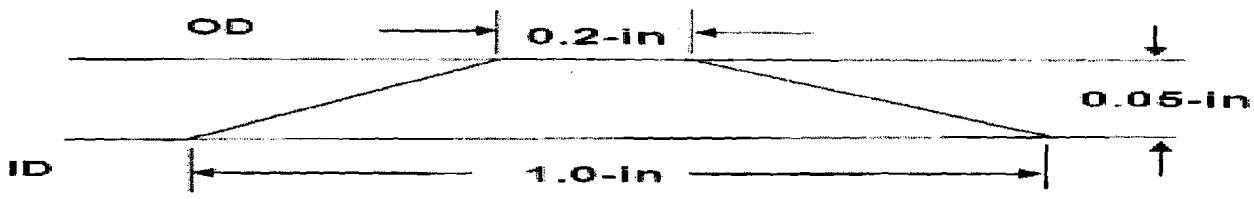
Time Dependent Crack Growth

- Early work on SCC Showed that the leak Rates are time Dependent
  - Attributed to Tearing of Ligaments and Opening of the Crack due to Limited time Dependent Deformation (Steady-State Creep rate very low at Operating Temperatures)
- Recent Tests show that, at Least at Room Temperature, Actual Crack Growth Occurs and at high Rates

Time Dependent Crack Growth Tests Program

- Test Material – Alloy 600, 7/8 inch Diameter Tubes, 0.05 inch Wall Thickness
- EDM Flaw Shape
  - Trapezoidal - 0.2 in OD, 1.0 in ID
  - Trapezoidal - 1.0 in OD, 0.2 in ID
  - Rectangular – 0.2 in, 0.4 in, 0.6 in,
  - EDM Notch Width 0.007 in
- With and Without a Foil and Bladder
- Open to air
- With a Small Shroud (1 ½ in Diameter)
- With a Large Shroud (4 in Diameter)

Trapezoidal Flaw Design

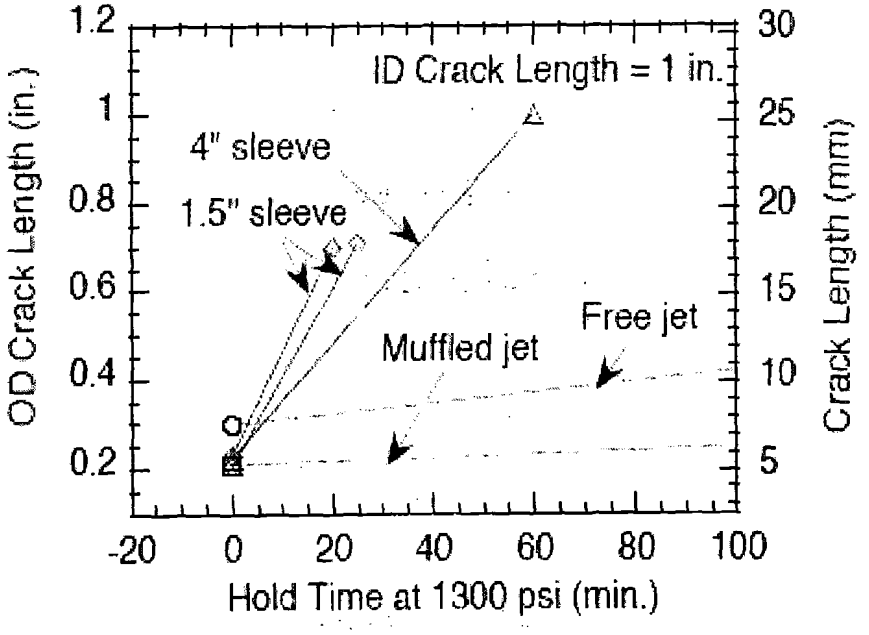


EDM Notch Approximately 0.007 Inches Wide

Test Results

- Jet-Free Tests with Bladder/Foil
  - No Crack Extension when Pressurized Using Nitrogen (wet or dry)
  - Low Crack Growth rate Using the Facility Pump
- Crack Growth Rates with Active jets
  - With Active jets Present, OD Increased from 0.2 in. to 1.0 in. in hours
  - Tests with Shrouds Filled with Water (to Simulate Expected jet Interactions with surrounding Tubes) Produced Higher Crack Growth Rates than Unconfined jets

Effect of Test Conditions on Crack Length for Trapezoidal Cracks



Proposed Mechanisms for Increase in Leak Rate at Constant Pressure

- Mechanisms for Crack Growth
  - Jet Erosion of the Crack Faces
  - Rapid Flaw Corrosion at Room Temperature
  - Jet/Flaw Structural Dynamic Interaction Resulting in Fatigue Crack Growth
  - Pressure Oscillation from the Pump Causing Crack Growth

Statistical Treatment of Models

- Candu Tube Inspection Assessment (CANTIA) – Developed by Dominion Engineering, Inc. for the Canadian Nuclear Safety Commission
- Determines Probabilities of Failure and leak rate from Primary to Secondary side During Normal Operation and Design Basis Accident Conditions
- Integrity, leak rate, and Degradation Models in CANTIA Specifically Intended for CANDU Steam Generators

Statistical Treatment of Models

- ANL Modified the CANTIA code Maintaining Basic Monte-Carlo Structure but Incorporating the ANL Revised Models for Predicting Ligament, Unstable Burst, Crack Opening Area, and Leak Rate of Flawed Alloy 600 Tubes
- Source Language was Updated from Visual BASIC 3.0 to Visual BASIC 6.0
- Basic Flaw was Changed from 1-D to 2-D
- Added two Models for Stress Corrosion Crack Growth Rate (the Scott Model and the Ford and Andresen Model)

Summary of Results

- Models Presented for Plastic Collapse of a Tube with a TW Axial Crack and a Part TW Axial Crack – Original Model Underestimated Ligament Rupture Pressures for Short deep Cracks, ANL Modification Provided Better Prediction
- Equivalent Rectangular Crack Method Presented – Gives good Results for Initial Ligament Rupture, not as good for Subsequent Tearing
- Simple Orifice Model Presented
  - Good Agreement for Slits, Orifices, and Open Cracks
  - Models tend to Overestimate Leak Rates for Actual Cracks Because Remaining Ligaments the Crack Opening Area and Meandering Crack Paths Increase L

Summary of Results (Cont)

- Pressurization Rate Effects Presented – No Effect at Typical Industry test Rates
- Secondary Side Depressurization Study Presented
  - Dynamic Loads have Virtually No Effect on Axial Flaws
  - If >1.5% of Tubes are Locked, TW Circ Cracks < 180° are Stable
- Constant Pressure Crack Growth Results Shown
  - Active jets Produce Increased Growth Rate with Time
- Statistical Treatment of Models Presented

Future Work on Tube Integrity

- Conduct Tests on Complex Morphology Cracks and Develop Predictive Models for Leak and Rupture Pressure
- Assess Alternatives to the Equivalent Rectangular Crack Method to Estimate Failure Pressures and Leak Rates
- Continue Development of the CANTIA Code

Task 1 - ASSESSMENT OF INSPECTION RELIABILITY

- Objective – Evaluation of Existing ISI Methods for Detection of Current Day Flaws
- Review of EC Round Robin on NRC/ANL SG Mockup
  - Mockup also used to Assess new Probe Designs
- Signal-to-Noise Issue
- X-Probe Evaluation

Steam Generator Mockup Round Robin

- Eddy Current Data Collected by Qualified Industry Team to Current Industry Practices and Qualification Proceedures
- 11 Qualified Analysis Teams Participated in the Round Robin Using the ANL/NRC SG Mockup
- Teams Consisted of a Primary, a Secondary, and 2 Resolution Analysts and a Qualified Data Analyst to Resolve Disputes
- The Differences Between the Teams was not Great, Although one team did not do as well as the Others
- Flaws were Missed Because
  - Signals were too Complex (Phase Angle did not show Expected Behavior)
  - Low Signal-to-Noise
  - Human Error

Task 2 - Research on ISI

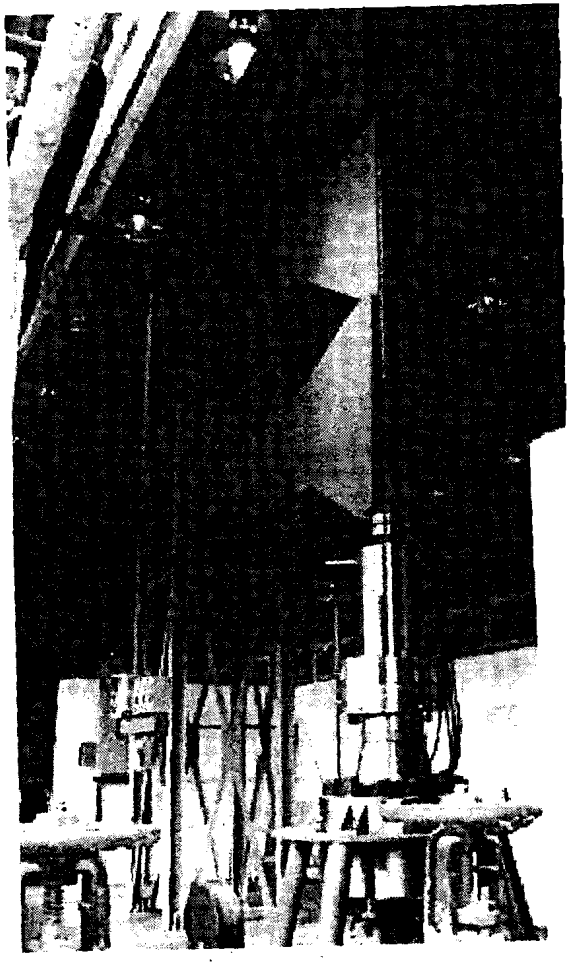
- Objective – Evaluate Advanced NDE and Signal Analysis Techniques – Improved Flaw Detection and Sizing
  - Practical need for Advanced Characterization techniques for Round Robin Because is was Impractical to Characterize Hundreds of Flaws Metallographlly

Task 4 - Degradation Modes

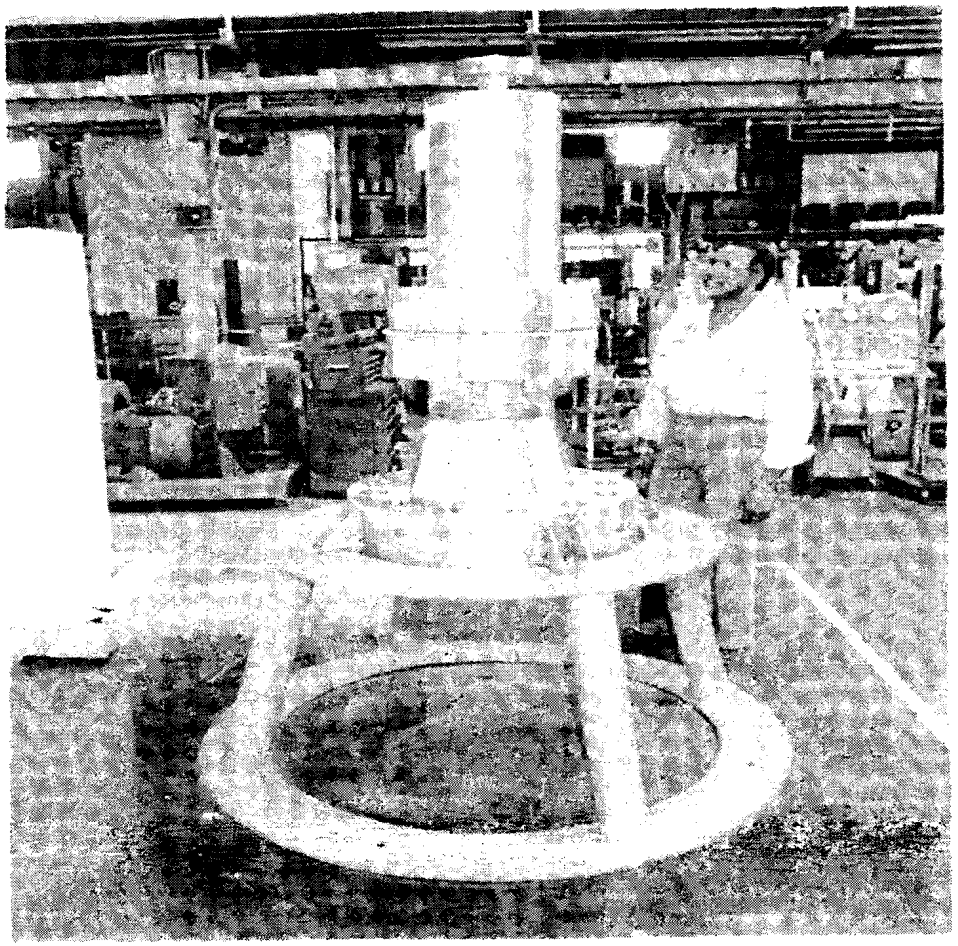
Objective –

- Evaluate and Validate Models for Degradation Modes
- Improve Understanding – Crevice Conditions, SCC Initiation, Evolution, Growth
  - Assess Implications of Known Susceptibility of Alloy 690 TT to SCC Under some Laboratory Conditions for Future Field Experience

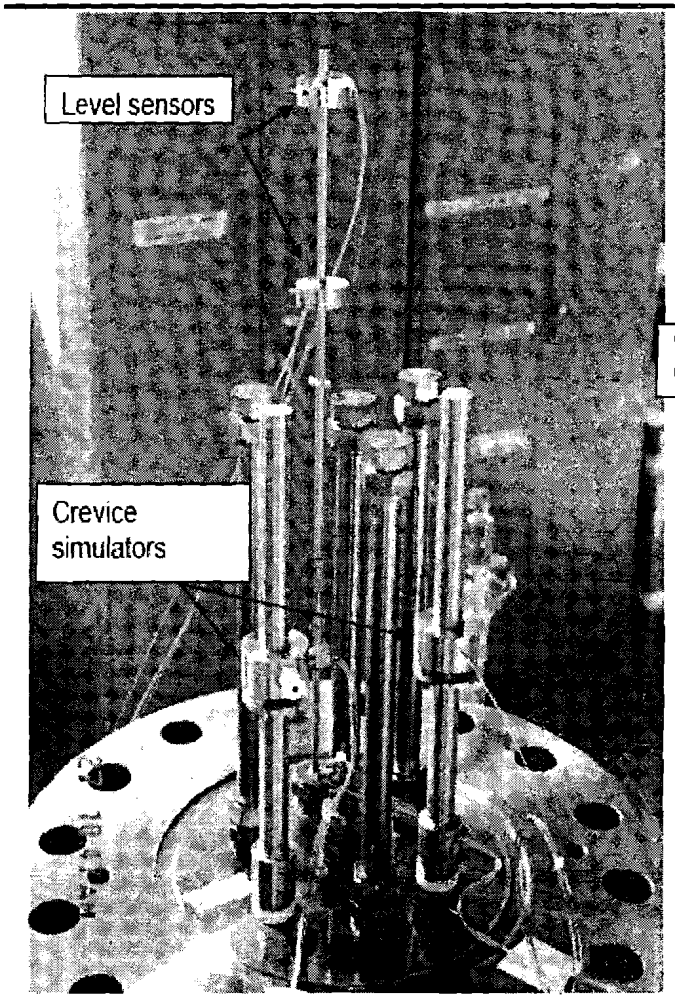
Model Boiler

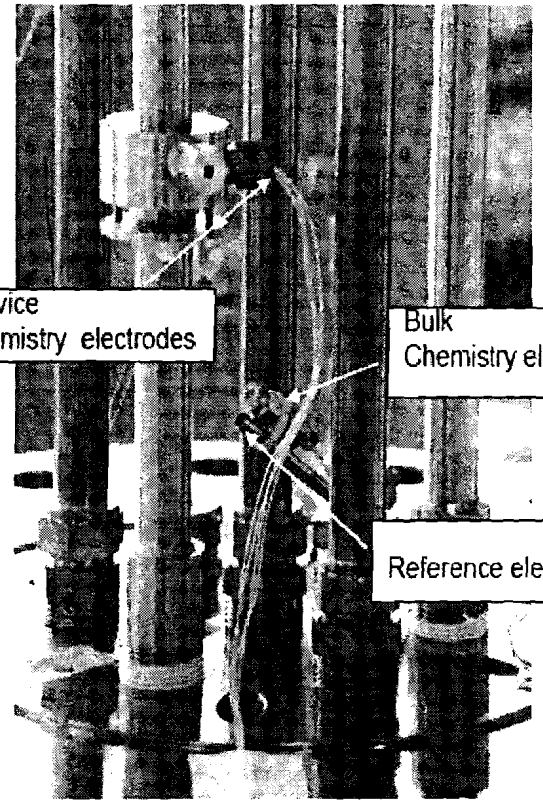


Argonne Model Boiler (Primary / Secondary Chambers)



Model Boiler
Crevice / Bulk Water Instrumentation





Overall Conclusion

- Task 3 - Tube Integrity and Integrity Predictions
  - Models Developed for Predicting Rupture and Burst Pressures for Axial and Circ TW and Part TW Cracks
  - Developed Models for Severe Accidents
  - Models tend to Predict Ligament Rupture Better than Burst Pressure
  - Simple Orifice Model Provides Good Agreement for Idealized Cases, not as good for Actual Cracks
  - On Pressurization rate Effect At Industry test Rates
  - Secondary side Depressurization Study Showed no Effect of Dynamic Loads on Axial Cracks, if a few Tubes are Locked, need big TW Circ Cracks to fail
  - Cracks Grow at Constant Pressure if Active Jets are Present

Overall Conclusion (Cont)

- Task 3 -Assessment of Inspection Reliability
  - SG Round Robin – Teams were Fairly Consistent – Flaws Missed due to Complex Signals, Low Signal-to-Noise, Human Error
- Task 2 – ISI Technology – Advanced Characterization Techniques Developed to Supplement Destructive Evaluation
- Task 4 Degradation Modes – Showed the Model Boiler

Overview

- Research is Being Conducted in a Variety of Areas
- NRR has Requested Research Related to ISI Capabilities, ISI Reliability, and Rupture, Burst, and Leakage Models
- NRR will use this Information in the Review of Licensee Submittals and to Provide Guidance to Regional Inspectors
- In Response to ACRS Feedback, Additional Work is Being Conducted on Crevice Chemistry and the Relative Susceptibility of Various Tube Materials to Cracking

Item # 10 Handout 1

NRC DIGITAL SYSTEM RESEARCH PLAN FY 2005 THROUGH FY 2009

Michael E. Waterman, Sr. I&C Engineer
William E. Kemper, Section Chief
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Engineering Research Application Branch
Division of Engineering Technology
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OVERVIEW

- Provides a flexible, adaptable framework for identifying NRR, NMSS, and NSIR research initiatives
  - 27 Projects across 6 Research Programs
- Oriented toward providing more consistent processes for regulating nuclear applications
  - Technical guidance
  - Regulatory-based objective acceptance criteria
  - Assessment tools and methodologies
  - Review and inspection procedures
  - Staff training
- Draft – comments to be incorporated

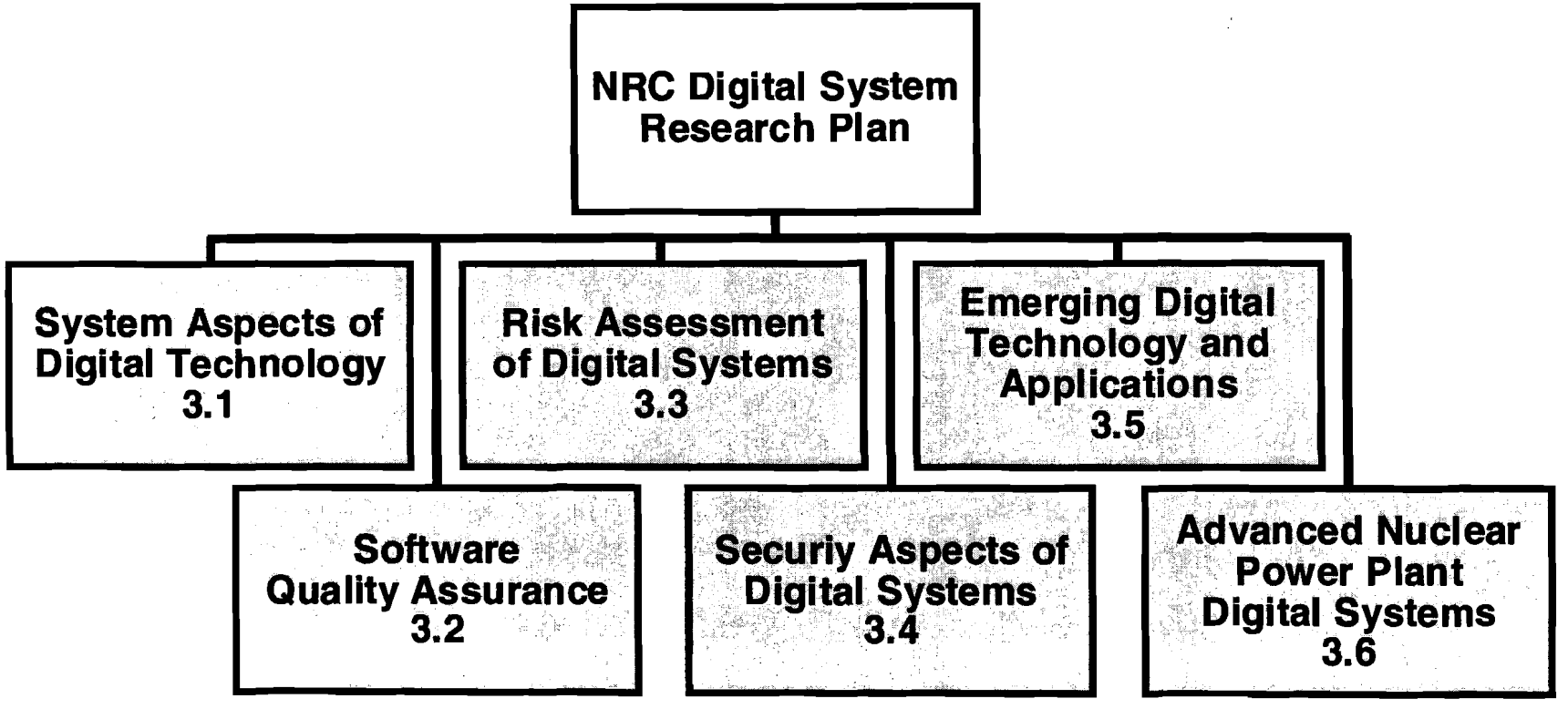
CURRENT SITUATION

- Issues facing NRC
  - Licensees are replacing analog systems with digital systems
  - Licensing these digital systems presents challenges to NRC
    - Increased complexity
    - Rapid changes in digital technology
    - New failure modes
  - NRC licensing processes should be kept current
  - A risk-informed, performance based safety assessment process should be developed for licensing digital systems

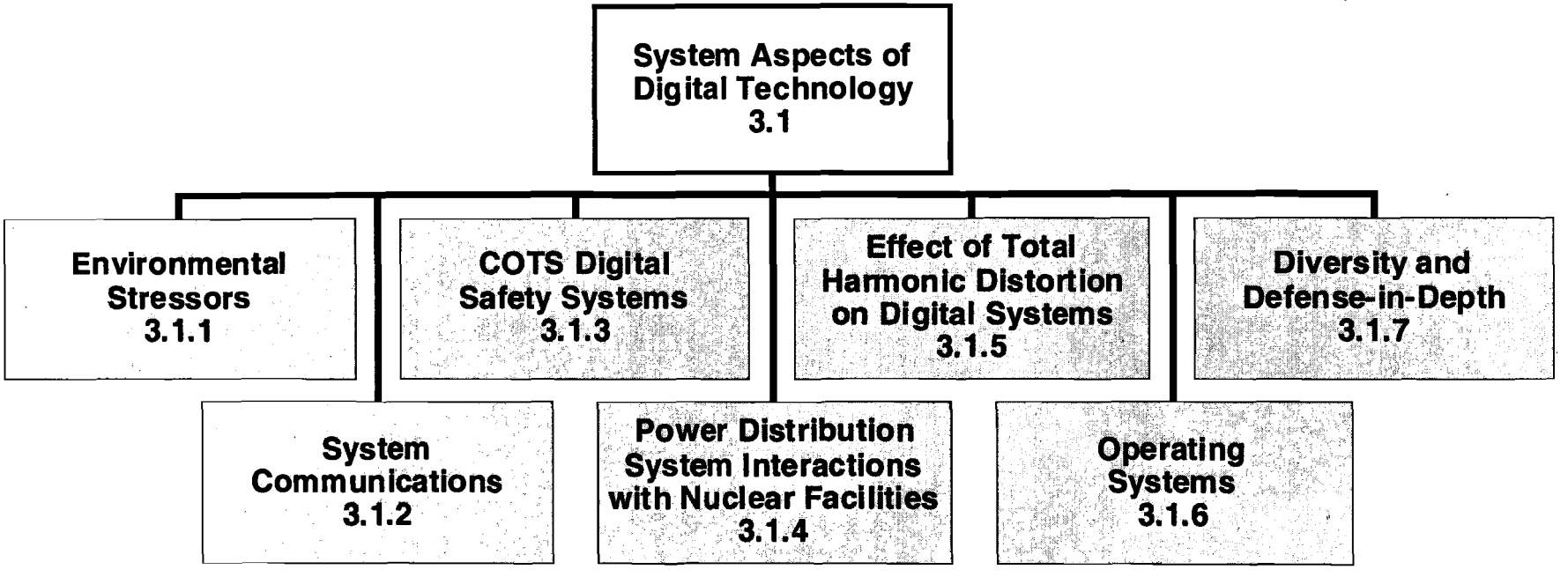
RESEARCH FOCUS

- Structured to develop better methods and understand new technologies
  - Risk-informed (e.g., risk assessment capabilities)
  - Performance based (e.g., dependability assessments)
  - Objective and repeatable (e.g., software quality evaluation methodologies)
- Broad-based, focusing on improving traditional review methods for
  - New applications of existing technologies
  - Advanced technologies
  - New issues and regulatory requirements

RESEARCH PROGRAMS



SYSTEM ASPECTS OF DIGITAL TECHNOLOGY PROGRAM 3.1



ENVIRONMENTAL STRESSORS PROJECT 3.1.1

- Environmental compatibility for safety-related I&C systems depends on maintaining the expected environment in the nuclear power plant and qualifying the equipment to withstand that environment
- Specific, comprehensive regulatory guidance
  - EMI/RFI – Updated RG 1.180
  - Lightning – DG-1137
  - Environmental qualification (EQ) – DG-1077

SYSTEM COMMUNICATIONS PROJECT 3.1.2

- The trend in digital safety systems is towards networked intrasystem architectures using dedicated communication microprocessors and proprietary communication protocols
- NRC requires expertise to evaluate these complex digital communication systems and the failure analysis techniques for these architectures
- The research will provide acceptance criteria and methodologies for reviewing these systems

COTS DIGITAL SAFETY SYSTEMS PROJECT 3.1.3

- The nuclear industry is retrofitting existing analog systems with COTS-based digital systems
- Research will evaluate methods for performing more quantitative safety assessments
  - Fault injection method for estimating digital system (HW, SW, HW+SW) dependability in COTS
- This project will further refine these methods for incorporation into NRC review methodologies by using realistic safety-related COTS systems as test beds

ELECTRICAL POWER DISTRIBUTION SYSTEM INTERACTIONS PROJECT 3.1.4

- NPP digital-controlled power equipment sensitivity to changes in grid voltages has resulted in undesirable equipment responses
- Research will support RES/DSARE/AREAB efforts to model highly distributed, complex systems composed of digital, analog, discrete, high voltage, high current power components to determine the effects of grid voltage fluctuations on digital equipment

EFFECT OF THD ON DIGITAL SYSTEMS PROJECT 3.1.5

- Newer digital components are more sensitive to total harmonic distortion (THD)
  - Higher IC circuit densities
  - Lower voltage requirements for memory states
- THD could be a potential CMF mechanism
- Currently, no methods exist in NRC to evaluate the effect of THD on digital system components
- This research project will evaluate the effect of THD on digital systems and provide guidance on acceptable THD thresholds

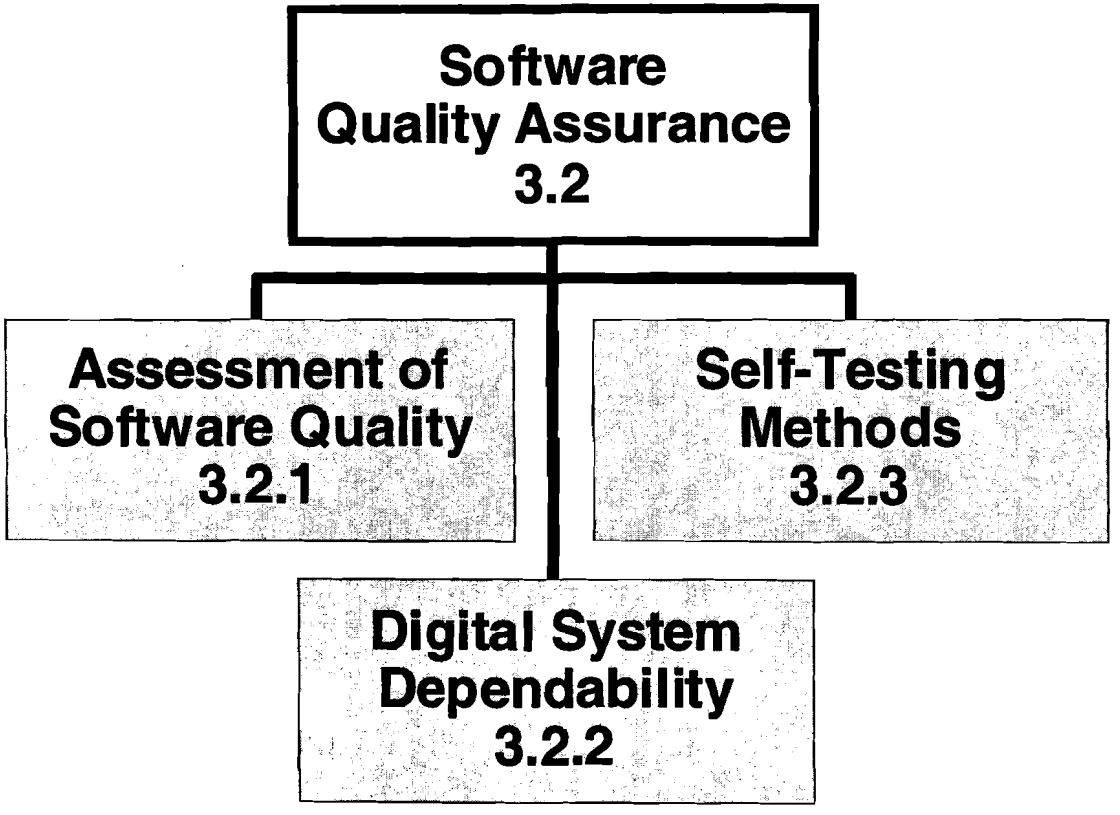
OPERATING SYSTEMS PROJECT 3.1.6

- NRC has not been able to assess proprietary COTS operating system characteristics
- RES initiated a study of operating system characteristics
- The results were inconclusive
- Further research will
  - Identify safety-critical design aspects of operating systems
  - Develop processes for performing safety assessments of operating systems

DIVERSITY AND DEFENSE-IN-DEPTH PROJECT 3.1.7

- D3 position and guidance are deterministic
- The nuclear power industry has proposed using risk insights from PRAs
- This project will
  - Verify, deterministically, that existing guidance (SRP BTP HICB-19) is realistically conservative
  - Evaluate NUREG/CR-6303 coping strategies
  - Perform case studies of digital safety system configurations to evaluate their susceptibility to CMF
  - Evaluate the fault injection process as a methodology for identifying CMF vulnerabilities

SOFTWARE QUALITY ASSURANCE PROGRAM 3.2



ASSESSMENT OF SOFTWARE QUALITY PROJECT 3.2.1

- NRC evaluates the quality of digital systems development processes manually
- This research project is developing a more effective and thorough supporting process
- Complements the fault injection testing assessment methodology already developed for digital system dependability testing
- HRP is evaluating SWE practices and criteria that may be effective in assuring software quality

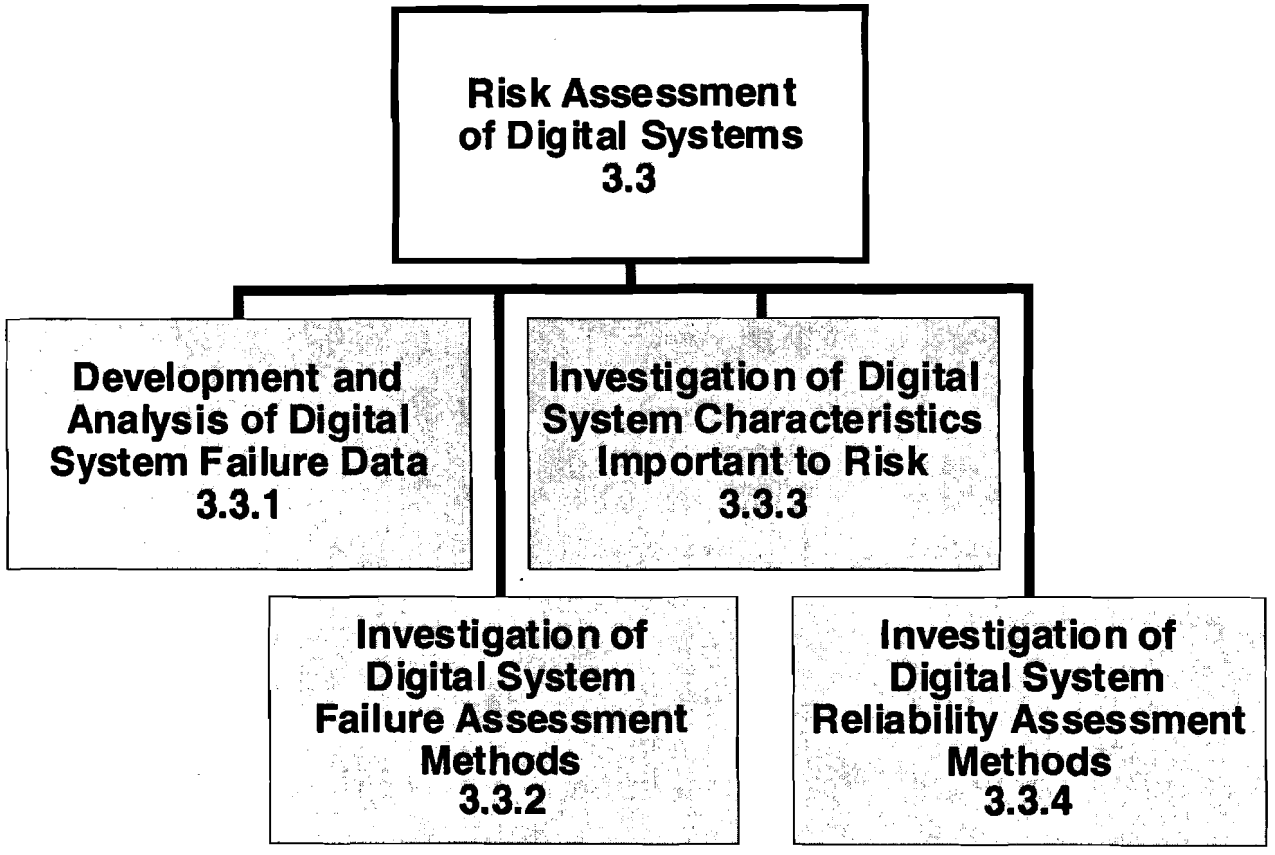
DIGITAL SYSTEM DEPENDABILITY PROJECT 3.2.2

- Safety significant errors in digital systems may not be detected by V&V processes
- Methods are needed to evaluate digital systems
- A fault injection methodology has been developed to evaluate dependability
- This project will produce a process for using this tool to determine the dependability of digital safety systems
  - Three SR COTS platforms will be evaluated

SELF-TESTING METHODS PROJECT 3.2.3

- Self-testing methods test hardware and software continuously to improve availability
- The technical issues concern
  - Effectiveness in determining system performance
  - Adverse effects on safety system performance
  - Identifying acceptable self-testing methods
  - The amount of self-testing that is sufficient
- This research project will develop technical guidance and review methodologies for evaluating self-testing features in digital systems

RISK ASSESSMENT OF DIGITAL SYSTEMS PROGRAM 3.3



DEVELOPMENT AND ANALYSIS OF DIGITAL SYSTEM FAILURE DATA PROJECT 3.3.1

- The NRC is risk-informing its activities
- Assessing failure probabilities requires that the NRC have a standard process for collecting, analyzing, and using digital system data
- The purpose of this research project is to
  - Collect and assess digital system failure data
  - Evaluate digital system failure assessment methods and data used by defense, aerospace, and other industries
  - Develop a process to identify the frequency, severity, cause, and possible prevention of digital system failures
  - Maintain the digital system reliability data for use in modeling digital systems in PRAs

INVESTIGATION OF DIGITAL SYSTEM FAILURE ASSESSMENT METHODS PROJECT 3.3.2

- To support risk assessments, NRC should develop or identify methods for assessing digital system failure modes
- Guidance and criteria on the use of these methods and how to support risk assessments of digital systems in an integrated process should be defined
- This research project will
  - Survey analytical methods for identifying digital system faults and their impact on safety
  - Describe the advantages and disadvantages of each method
  - Provide guidance for using digital system failure assessment techniques, and the criteria for using the techniques

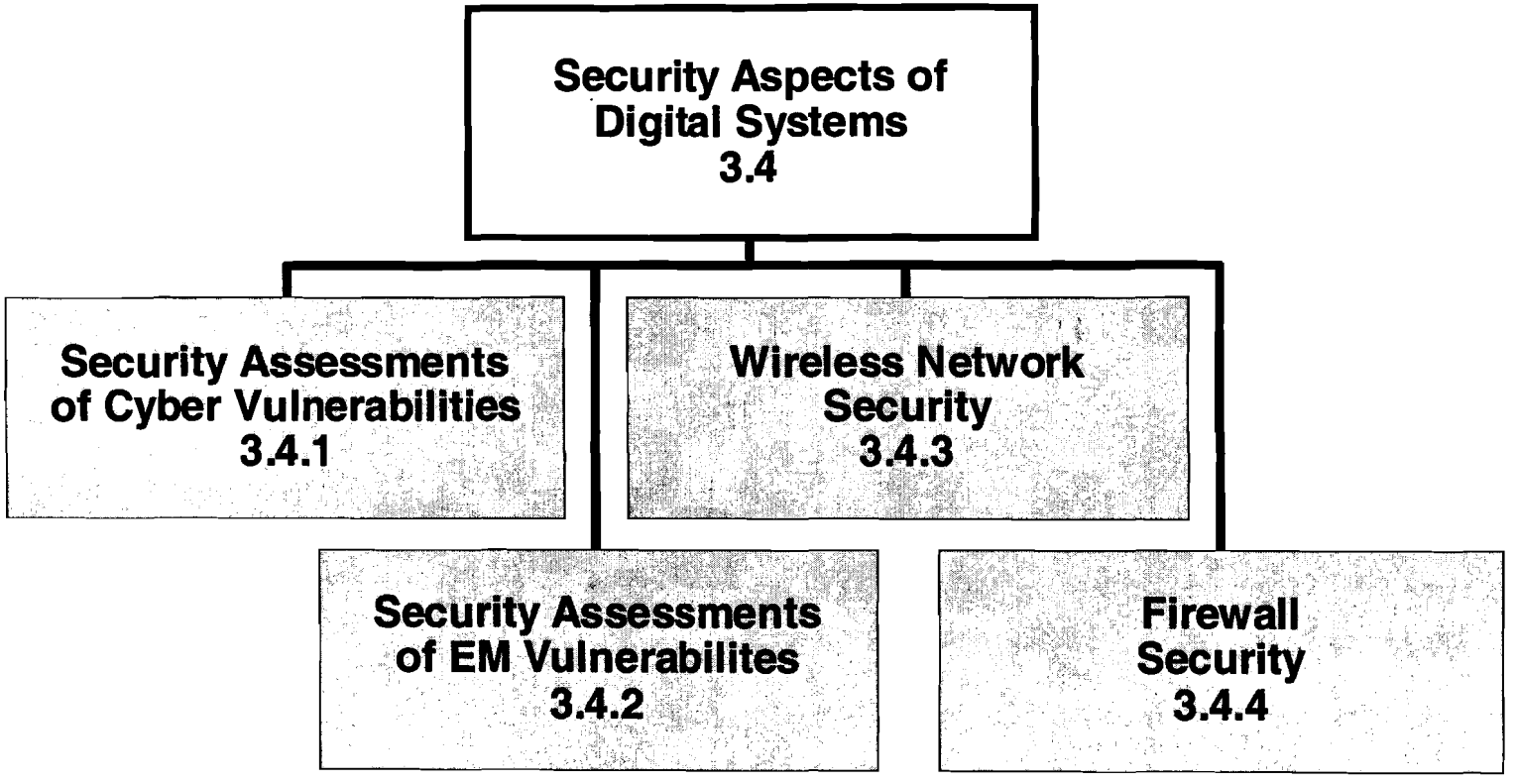
INVESTIGATION OF DIGITAL SYSTEM CHARACTERISTICS IMPORTANT TO RISK PROJECT 3.3.3

- PRAs model digital systems as "black boxes"
- Need to incorporate risk models into PRAs
- Need a consistent approach and acceptance criteria for reviewing risk-informed systems
- This research project will
  - Develop risk models of digital systems
  - Identify digital systems to be modeled and the level of detail to be modeled
  - Identify sub-components that may warrant attention
  - Develop a methodology for performing these activities

INVESTIGATION OF DIGITAL SYSTEM RELIABILITY ASSESSMENT METHODS PROJECT 3.3.4

- The NRC needs a standard methodology for analyzing digital system reliability so that acceptance criteria can be applied to risk-inform safety system designs
- This research project will
  - Identify digital system reliability assessment methods
  - Develop a digital system reliability assessment methodology
  - Conduct case studies to assess the methodology
  - Support the development of acceptance criteria (Reg. Guide 1.17x)

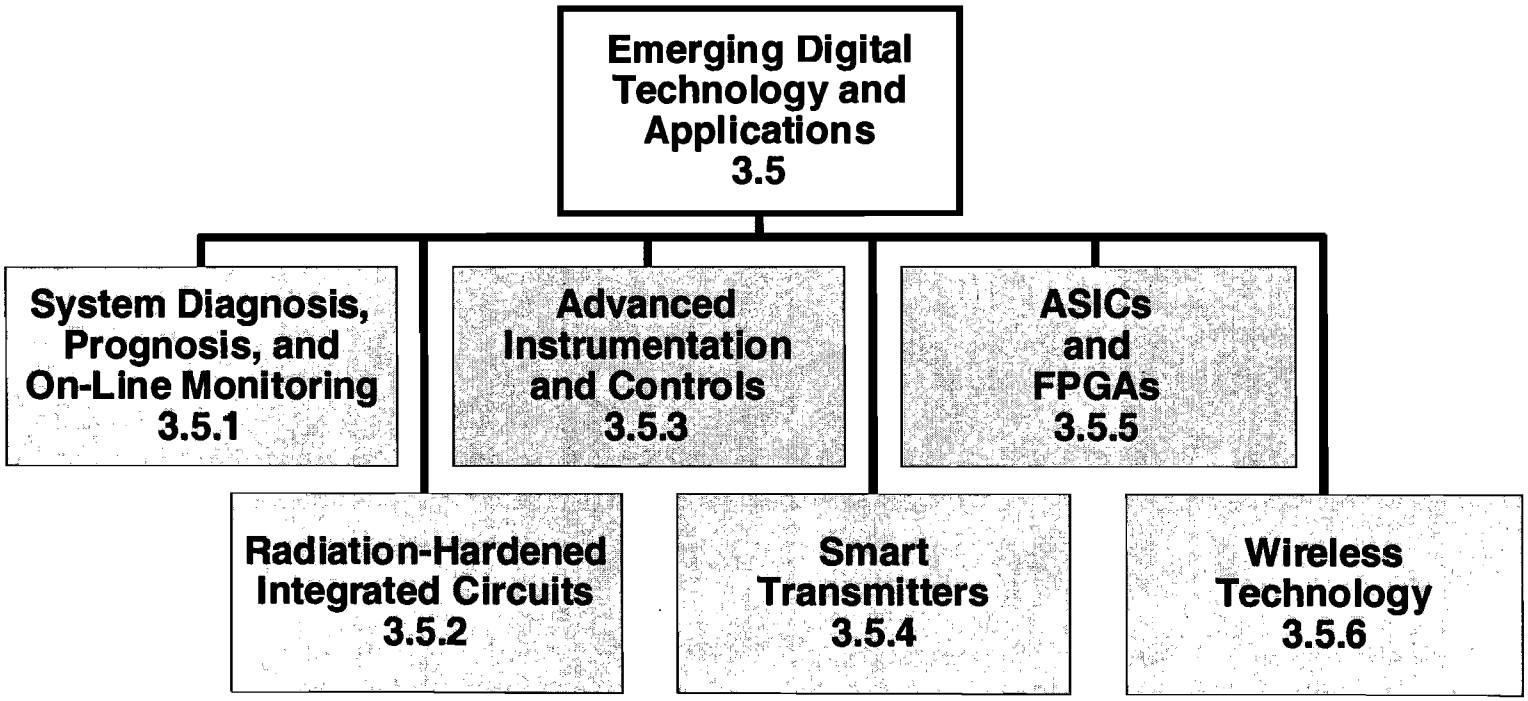
SECURITY ASPECTS OF DIGITAL SYSTEMS PROGRAM 3.4



SECURITY ASPECTS OF DIGITAL SYSTEMS

- Cyber security is an NRC concern that has been heightened since the events on 9/11
- Digital system security requires addressing potential vulnerabilities during system development and after installation
- Four projects are being initiated
  - Security of digital platforms
  - Site-specific protocol analysis
  - Secure network design techniques
  - Guidelines for NPP cyber security policy development

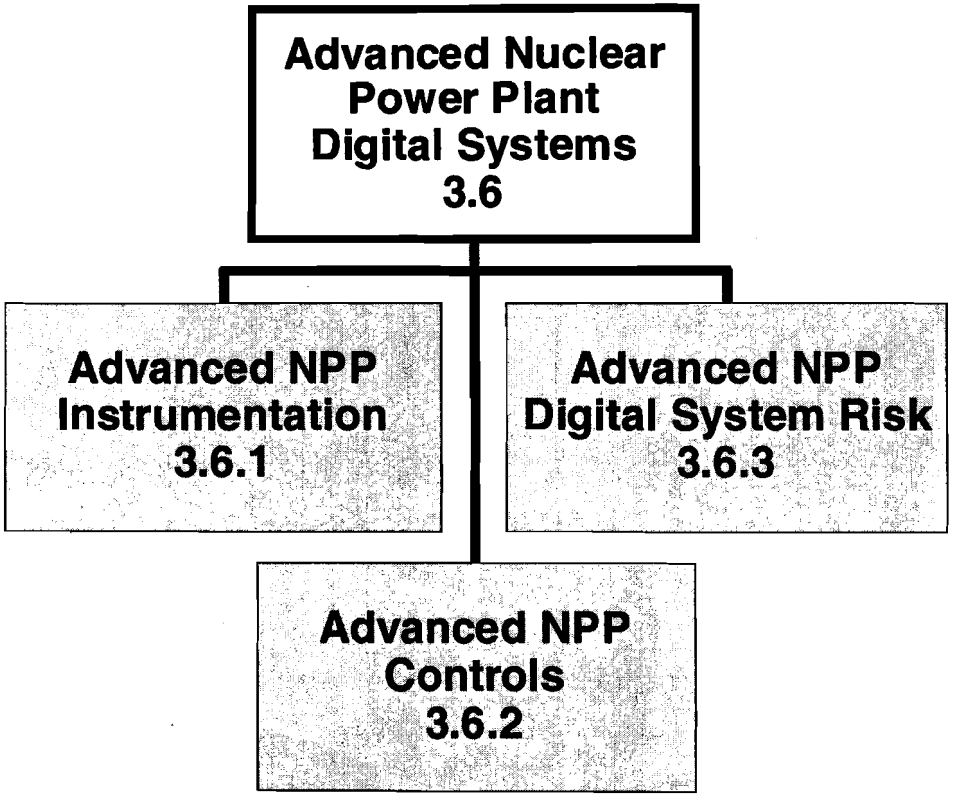
EMERGING DIGITAL TECHNOLOGY AND APPLICATIONS PROGRAM 3.5



EMERGING DIGITAL TECHNOLOGY AND APPLICATIONS

- Knowledge about new, emerging technologies is critical for NMSS, NRR, and NSIR staff to license safety related applications in an effective and realistic manner
- This research will provide regulatory guidance for reviewing NPP applications
- Ongoing projects include
  - Emerging technology evaluations
  - On-line Monitoring
  - Advanced flow meters
  - Wireless technologies

ADVANCED NPP DIGITAL SYSTEMS PROGRAM 3.6



ADVANCED NPP DIGITAL SYSTEMS

- Advanced reactor designs (ACR-700, EPR, ESBWR, PBMR, etc.) may apply new I&C technologies, and thereby present challenges for identifying risk-informed characteristics
  - Robotics, fuzzy logic controls, autonomous controls, fully integrated DCSs, new instrumentation, etc.
- Research projects are dependent on future advanced reactor design pre-application submittals
  - No research in progress at this time

SUMMARY

- Provides a flexible, adaptable framework for identifying NRR, NMSS and NSIR research initiatives
- Broad-based program oriented toward providing more consistent processes for regulating nuclear applications; improving review methods for new applications of existing technologies, advanced technologies and new issues; and developing regulatory requirements
- RES is looking forward to working closely with the ACRS as these programs are implemented