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

February 3, 2006

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

Dear Chairman Diaz:

SUBJECT: SUMMARY REPORT - 528th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, DECEMBER 7-10, 2005, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 528th meeting, December 7-10, 2005, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following report, letters, and memoranda:

REPORT:

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

- Vermont Yankee Extended Power Uprate, dated January 4, 2006

LETTERS:

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

- Draft Final Generic Letter 2005-XX, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs, dated December 21, 2005

- Early Site Permit Application for the Grand Gulf Site and the Associated Final Safety Evaluation Report, dated December 23, 2005

MEMORANDA:

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

- Proposed Revision to Standard Review Plan Section 17.5, "Quality Assurance Program Description Design Certification, Early Site Permit and New License Applicants," dated December 16, 2005

- Proposed Revision to Regulatory Guide 1.76, "Design Basis Tornado and Tornado Missiles for Nuclear Power Plants," and Standard Review Plan, Sections 2.3.1, "Regional Climatology," and 3.5.1.4, "Missiles Generated by Tornado and Extreme Winds," dated December 16, 2005

- Draft Regulatory Guide DG-1120, "Transient and Accident Analysis Methods," and Standard Review Plan Chapter 15, Section 15.0.2, "Review of Transient and Accident Methods," dated December 16, 2005

HIGHLIGHTS OF KEY ISSUES

1. Final Review of the Vermont Yankee Extended Power Uprate Application and the Associated Safety Evaluation

The Committee reviewed the application by Entergy Nuclear Vermont Yankee for an Extended Power Uprate (EPU) for the Vermont Yankee Nuclear Power Station. The Committee considered several technical issues, most notably, the request by the licensee for containment overpressure credit related to the emergency core cooling system (ECCS) pump Net Positive Suction Head calculation during design basis loss-of-coolant accident (LOCA) and anticipated transient without scram (ATWS) scenarios. The Committee also considered the licensee's actions related to steam dryer cracks, which have been discovered at other BWRs that have implemented EPUs.

Entergy, the NRC staff, and representatives of the State of Vermont and the New England Coalition made presentations regarding this application and many comments were submitted by members of the public. Members of the public especially emphasized the need for an independent inspection of the plant, and they commented that the inspection performed by the NRC staff was not sufficiently thorough to support the uprate.

Committee Action

The Committee issued a report dated January 4, 2006, recommending that the Vermont Yankee Extended Power Uprate be approved. The Committee also recommended that the change in the licensing basis associated with the requested containment overpressure credit be approved.

The Committee also concluded that the monitoring that will be performed during the ascension to uprate power provides adequate assurance that, if resonant vibrational modes are induced in the steam dryers, they will be identified prior to component failure.

The Committee also concluded that load rejection and main steam isolation valve closure transient tests are not warranted, because the planned transient testing program adequately addresses the performance of the modified systems. The staff used EPU Review Standard RS-001 to evaluate this application, and the Committee commented that it provides a structured process for the review of applications for extended power uprates, and its continued use and improvement are encouraged.

2. Draft ACRS Report on the NRC Safety Research Program

The ACRS provides the Commission a biennial report, presenting the Committee's observations and recommendations concerning the overall NRC Safety Research Program. During the December meeting, the Committee discussed its draft 2006 report to the Commission on the NRC Safety Research Program.

Committee Action

The Committee plans to continue its discussion of its draft report on the NRC safety research program during its February 2006 meeting.

3. Early Site Permit Application for the Grand Gulf Nuclear Station and the Associated Final Safety Evaluation Report

The Committee heard presentations by and held discussions with representatives of the NRC staff and System Energy Resources, Inc. (SERI), the applicant for an early site permit (ESP) for the Grand Gulf site. The Committee discussed the application and the associated NRC staff's final safety evaluation report (FSER).

SERI seeks a site permit for a reactor or a set of reactor modules of total power up to 4300 MWt on a site adjacent to the current Grand Gulf Nuclear Power Station 1, a boiling water reactor (BWR/6) with a Mark III containment. With the additional unit or modules, the total nuclear generating capacity at the Grand Gulf site could be as high as 8600 MWt. The Grand Gulf site had previously been approved for two units, but the second unit was not completed.

The SERI application for an ESP does not specify a particular power plant technology for the new reactor to be placed on the proposed site. The ESP, instead, uses a plant pararneter envelope of power plant characteristics that is intended to bound the reactor technology that could eventually be selected.

Committee Action

The Committee issued a letter to the NRC Executive Director for Operations (EDO) on this matter dated December 23, 2005, commending the NRC staff on the development of a comprehensive and readable FSER. The Committee agreed with the staff's three permit conditions for an ESP and 26 action items for the combined license phase. The Committee, however, stated that the FSER should be issued once the staff has made more explicit analyses of the hazards posed by explosions in transportation accidents on the Mississippi River. The NRC staff also needs to provide additional guidance to applicants concerning the discussion in an application of major features of the emergency planning for a proposed site.

4. Draft Final Generic Letter, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Fire Protection Regulations"

The Committee heard presentations by and held discussions with representatives of the staff regarding the Draft Final Generic Letter, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Fire Protection Regulations." The staff provided the Committee a summary of its responses to the comments received during the public comment period. The staff explained the justification for the Draft Final Generic Letter. The Office of Nuclear Regulatory Research (RES) tests showed the Hemyc fire barrier failed to meet the 1-hour requirement. A representative from the Nuclear Energy Institute (NEI) provided NEI's view on the Draft Final Generic Letter. NEI said this Generic Letter is another example of a generic communication imposing new regulatory requirements, and the NRC has the right to amend its regulations but this should be done in the more disciplined rulemaking process. The staff stated the Generic Letter is not imposing a new burden and does not constitute a backfit according the Committee to Review Generic Requirements.

Committee Action

The Committee issued a letter to the NRC EDO on this matter dated December 21, 2005, recommending that the staff issue the Generic Letter.

5. Meeting with the NRC Commissioners

The Committee met with the NRC Commissioners to discuss items of mutual interest. Topics for discussion included license renewal, early site permits, new plant licensing, proposed alternative embrittlement criteria in 10 CFR 50.46, fire protection matters, power uprate technical issues, and future ACRS activities.

In a Staff Requirements Memorandum (SRM) dated December 20, 2005, resulting from this meeting, the Commission requested that the ACRS inform the Commission of its plans to manage the increased workload resulting from the anticipated receipt of new reactor designs and combined license applications. Also, the ACRS shall make among its highest priorities its role in the resolution of GSI-191.

Committee Action

This was an information briefing. The Committee plans to consider this matter during future meetings as further progress has been made by the staff.

6. Proposed Program Plan and Advance Notice of Proposed Rulemaking for Risk-Informing 10 CFR Part 50

The Committee met with the NRC staff to discuss a draft Commission Paper which outlines a formal program plan to make a risk-informed and performance-based revision to 10 CFR Part 50, including revisions to the applicable Regulatory Guides, Standard Review Plan, or other guidance documents. The program plan was developed in response to the Commission's SRM dated May 9, 2005. The draft Commission Paper also included an Advance Notice of Proposed Rulemaking that seeks stakeholder feedback on rulemaking approaches for making technical requirements for power reactors more risk-informed and performance-based.

7. Staff Activities Associated with Responding to the Commission's SRM related to Safety Conscious Work Environment and Safety Culture

The Committee heard presentations by and held discussions with representatives of the NRC staff on NRC's safety culture initiatives described in SECY-04-0111 [Recommended Staff Actions Regarding Agency Guidance in the Areas of Safety Conscience Work Environment and Safety Culture] and the associated Commission's SRM dated August 30, 2004.

The staff described what had been done organizationally with respect to Commission's direction. This included establishing a Steering Committee, a Safety Culture Working Group, and a support Regional team. Also, the staff discussed the November 29th and 30th, 2005 public meeting and the results that lead to the development of a conceptual approach titled Option G, and how Option G would address safety culture within a regulatory framework, including information sources, documentation, assessment, and follow-up which could include NRC's request to have licensees perform self-assessment of their safety culture. Mr. Cobey indicated that using a phased approach, safety culture would be addressed at different levels. At the lowest level, this would include looking for safety culture issues within baseline inspection of licensee's root cause analysis. In a degraded cornerstone column, the staff may request a licensee to have an independent assessment. And in a multiple/repetitive degraded cornerstone column, the NRC may perform its own evaluation of a licensee's safety culture. A representative of NEI provided insights on INPO's safety culture initiative.

Committee Action

This was an information briefing. The Committee plans to decide on a course of action on this matter following a joint meeting of the ACRS Subcommittees on Human Factors and on Reliability and Probabilistic Risk Assessment scheduled to be held on January 25, 2006.

8. Election of ACRS Officers for CY 2006

The Committee re-elected Graham B. Wallis as ACRS Chairman, William J. Shack as ACRS Vice-Chairman, and John D. Sieber as Member-at-Large for the Planning and Procedures Subcommittee.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's response of November 28, 2005, to comments and recommendations included in the ACRS interim letter dated October 19, 2005, concerning the safety aspects of the license renewal application for the Browns Ferry Nuclear Plant Units 1, 2, and 3. The Committee decided that it was satisfied with the EDO's response.

  The EDO response stated that a licensee commitment to implement operating experience and aging management program reviews before entering the period of extended operation will be included in the safety evaluation report for the Browns Ferry extended power uprate application currently under review. The staff also committed to address the Committee's concerns regarding Unit 1 operating experience, the Unit 1 Periodic Inspection Program, and the terminology used for license-renewal related inspections in the final safety evaluation report related to the license renewal of the Browns Ferry Nuclear Plant Unit 1, 2, and 3.

- The Committee considered the EDO's November 3, 2005 response to the ACRS letter of September 23, 2005, concerning the Committee's review of the proposed technical basis for revision of the embrittlement criteria in 10 CFR 50.46(b). The Committee decided that it was satisfied with the EDO's response.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from November 3, 2005, through December 7, 2005, the following Subcommittee meetings were held:

- Power Uprates - November 15-16, 2005

The Subcommittee reviewed the application by Entergy Nuclear Northeast (Entergy) for an extended power uprate for the Vermont Yankee Nuclear Power Station. This meeting was held in Brattleboro, VT, to encourage public participation.

- Reliability and Probabilistic Risk Assessment - November 17-18, 2005

The Subcommittee discussed the details of the Standardized Plant Analysis Risk (SPAR) models development program.

- Power Uprates - November 29-30, 2005

The Subcommittee continued its review of the application by Entergy Nuclear Northeast (Entergy) for an extended power uprate for the Vermont Yankee Nuclear Power Station.

- Planning and Procedures - December 7, 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 plans to discuss with the NRC staff the lessons learned as a result of the review of the early site permit applications for Grand Gulf, North Anna, and Clinton. The Committee plans to discuss the staff's analyses of the hazards posed to the proposed Grand Gulf site by explosions in transportation accidents on the Mississippi River. In addition, the Committee plans to discuss the NRC staff's guidance concerning the major features of the emergency planning for a proposed site.

- The Committee would like to hear a briefing from the NRC staff after the staff has reviewed the licensees' responses to the Generic Letter 2005-xx, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs."

- The Committee plans to review the draft final version of the SRP Section 17.5, "Quality Assurance Program Description Design Certification, Early Site Permit and New License Applicants," after reconciliation of public comments.

- The Committee plans to review the draft final revision to Regulatory Guide 1.76, "Design Basis Tornado and Tornado Missiles for Nuclear Power Plants," and SRP Sections 2.3.1, "Regional Climatology," and 3.5.1.4, "Missiles Generated by Tornado and Extreme Winds," after reconciliation of public comments.

- The Committee plans to review additional guidance being developed by the staff for use in the consideration of overpressure credit.

PROPOSED SCHEDULE FOR THE 529th ACRS MEETING

The Committee agreed to consider the following topics during the 529th ACRS meeting, to be held on February 9-11, 2006:

- Evaluation of Human Reliability Analysis (HRA) Methods Against Good Practices
- Proposed Revision to SRP Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs"
- Draft ACRS Report on the NRC Safety Research Program
- FERRET Reactor Vessel Fluence Methodology
- Proposed revisions to SRP Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," and related matters.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

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DECEMBER 7-10, 2005

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- Draft Regulatory Guide DG-1120, "Transient and Accident Analysis Methods," and Standard Review Plan Chapter 15 Section 15.0.2, "Review of Transient and Accident Methods," dated December 16, 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

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MINUTES OF THE 528th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS DECEMBER 7-10, 2006 ROCKVILLE, MARYLAND

The 528th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on December 7-10, 2006. Notice of this meeting was published in the *Federal Register* on November 22, 2005 (65 FR 70638) (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, and Dr. Victor H. Ransom. 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 1:00 p.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 Vermont Yankee Extended Power Uprate Application and the Associated Safety Evaluation (Open)

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

The Committee reviewed the application by Entergy Nuclear Vermont Yankee for an Extended Power Uprate (EPU) for the Vermont Yankee Nuclear Power Station. The Committee considered several technical issues, most notably, the request by the licensee for containment overpressure credit related to the emergency core cooling system (ECCS) pump Net Positive Suction Head calculation during design basis loss-of-coolant accident (LOCA) and anticipated transient without scram (ATWS) scenarios. The Committee also considered the licensee's actions related to steam dryer cracks, which have been discovered at other BWRs that have implemented EPUs.

Entergy, the NRC staff, and representatives of the State of Vermont and the New England Coalition made presentations regarding this application and many comments were submitted by members of the public. Members of the public especially emphasized the need for an independent inspection of the plant, and they commented that the inspection performed by the NRC staff was not sufficiently thorough to support the uprate.

Committee Action

The Committee issued a report dated January 4, 2006, recommending that the Vermont Yankee Extended Power Uprate be approved. The Committee also recommended that the change in the licensing basis associated with the requested containment overpressure credit be approved.

The Committee also concluded that the monitoring that will be performed during the ascension to uprate power provides adequate assurance that, if resonant vibrational modes are induced in the steam dryers, they will be identified prior to component failure.

The Committee also concluded that load rejection and main steam isolation valve closure transient tests are not warranted, because the planned transient testing program adequately addresses the performance of the modified systems. The staff used EPU Review Standard RS-001 to evaluate this application, and the Committee commented that it provides a structured process for the review of applications for extended power uprates, and its continued use and improvement are encouraged.

III. Draft ACRS Report on the NRC Safety Research Program (Open)

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

The ACRS provides the Commission a biennial report , presenting the Committee's observations and recommendations concerning the overall NRC Safety Research Program.

During the December meeting, the Committee discussed its draft 2006 report to the Commission on the NRC Safety Research Program.

Committee Action

The Committee plans to continue its discussion of its draft report on the NRC safety research program during its February 2006 meeting.

IV. Early site Permit Application for the Grand Gulf Nuclear Station and the Associated Final Safety Evaluation Report (Open)

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

Dr. Dana A. Powers, Early Site Permit Subcommittee Chairman, stated that the purpose of this meeting was to hear a briefing by representatives of the NRC staff and System Energy Resources, Inc. (SERI), the applicant, regarding the early site permit application and the associated final safety evaluation report for the Grand Gulf site.

Mr. G. Zinke, Entergy, stated that on October 16, 2003, SERI submitted an ESP application for the Grand Gulf ESP site. The Grand Gulf ESP site is in Claiborne County near Port Gibson, Mississippi, approximately 25 miles south of Vicksburg, Mississippi, and is adjacent to the existing nuclear power reactor operated by Entergy Operations, Inc. The ESP site identified in the application is collocated with the Grand Gulf Nuclear Station, Unit 1. In its application, SERI seeks approval for an ESP that could support a future application to construct and operate additional nuclear unit(s) at the ESP site, with a total nuclear generating capacity of up to 8,600 megawatts thermal (Mwt), and a maximum 4,300 Mwt per unit.

In the Site Safety Analysis Report (SSAR) of the ESP application, SERI provided a list of postulated design parameters, referred to as the plant parameter envelope (PPE). The applicant stated that the PPE approach provides sufficient design details to support the NRC's review of the ESP application. SERI stated that the PPE is intended to bound multiple reactor designs and the actual reactor design selected would be reviewed at the combined license (COL) stage to ensure that the design fits within the PPE. The PPE references the following designs:

- ACR-700 (Atomic Energy of Canada, Ltd.)
- Advanced Boiling Water Reactor (General Electric)
- AP1000 (Westinghouse)
- Economic and Simplified Boiling Water Reactor (General Electric)
- Gas Turbine Modular Helium Reactor (General Atomics)

- International Reactor Innovative and Secure (IRIS) Project (Consortium led by Westinghouse)

- Pebble Bed Modular Reactor (PBMR (Pty) Ltd.)

Mr. Raj Anand, NRC staff, received an ESP application in September and October 2003 from Dominion Nuclear North Anna, LLC (Dominion), for the North Anna site; Exelon Generation Company, LLC (Exelon), for the Clinton site; and System Energy Resources, Inc. (SERI), a subsidiary of Entergy Corporation, for the Grand Gulf site. All three applications were accepted.

The regulations at 10 CFR Part 52 and 10 CFR Part 100, "Reactor Site Criteria," that apply to an ESP do not require that an ESP applicant provide specific design information. However, some design information may be required to address 10 CFR 52.17(a)(1), which calls for "an analysis and evaluation of the major structures, systems, and components of the facility that bear significantly on the acceptability of the site under the radiological consequence evaluation factors."

The NRC staff completed its final draft safety evaluation report (SER) on October 21, 2005. The SER summarized the results of the staff's technical evaluation of the suitability of the proposed site for a nuclear power plant(s) falling within the PPE that SERI specified in its application.

During the 523rd meeting on June 1-3, 2005, the Committee met with the NRC staff and SERI representatives and discussed the application and the associated draft SER. This matter was also discussed during the meeting of the ACRS Early Site Permits Subcommittee on May 16, 2005. On June 14, 2005, the ACRS issued its interim letter and concluded that the staff has prepared a quality SER of the SERI application for the Grand Gulf ESP. However, the ACRS stated in its letter that the draft SER should be augmented with a more complete exposition on threats posed by transportation accidents on the river adjacent to the proposed site.

The ACRS also commented in the interim letter that it continues to question the defensibility of the methods used by the staff and the applicant to prognosticate the weather at the site over the next 65 years based on historical frequencies of severe weather events. On August 12, 2005, the EDO responded to the ACRS interim letter. The staff concluded that considering the effects of climate change in ESP reviews is not required by existing NRC regulations and would be a departure from previous license reviews.

The staff agreed to add the following statement in the final SER: "The staff acknowledges that long-term climatic change resulting from human or natural causes may introduce changes into the most severe natural phenomena reported for the site. However, no conclusive evidence or consensus of opinion is available on the rapidity or nature of such changes. If in the future the ESP site is no longer in compliance with the terms and conditions of the ESP (e.g., if new information shows that the climate has changed and that the climatic site characteristics no longer represent extreme weather conditions), the staff may seek to modify the ESP or impose requirements on the site in accordance with the provisions of 10 CFR 52.39, “Finality of Early Site Permit Determinations,” if necessary, to bring the site into compliance with Commission requirements to assure adequate protection of the public health and safety.”

In the final SER, the NRC staff identified, in Appendix A, the proposed permit conditions that it will recommend to the Commission impose should an ESP be issued to the applicant. Appendix A also includes a list of COL action items or certain site-related items that will need to be addressed should this ESP be referenced as a part of a COL or construction permit application. The staff determined that these deferred items do not affect the staff's regulatory findings at the ESP stage. In addition, Appendix A lists the site characteristics and the bounding parameters identified by the staff for the ESP site.

Committee Action

The Committee issued a letter to the NRC Executive Director for Operations (EDO) on this matter dated December 23, 2005, commending the NRC staff on a comprehensive and readable FSER. The Committee agreed with the staff's three permit conditions for an ESP and 26 action items for the combined operating license phase. The Committee stated in its letter that the FSER should be issued once the staff has made more explicit analyses of the hazards posed by explosions in transportation accidents on the Mississippi River. The NRC staff also needs to provide additional guidance to applicants concerning the discussion in an application of major features of the emergency planning for a proposed site.

V. Draft Final Generic Letter, “Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Fire Protection Regulations” (Open)

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

The Chairman of the Fire Protection Subcommittee provided an introduction to the staff. The Committee had the benefit of presentations and discussions with representatives of the staff regarding the staff's Draft Final Generic Letter, “Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Fire Protection Regulations.”

The staff provided a brief background on the draft final generic letter and explained the primary purpose of the draft final generic letter. The staff provided the Committee a summary of six comment letters that the staff received during the public comment period. The major comments were in the areas of (1) backfit determination, (2) schedule, (3) Hemyc testing, (4) risk-informing, (5) Generic Letter (GL) 86-10, sSpplement 1, (6) miscellaneous, and (7) administrative. The staff provided their responses to the public comments.

The staff explained the justification for the draft final generic letter. The Office of Nuclear Regulatory Research (RES) tests showed failures of the Hemyc fire barrier in the 33 - 42 minute range, which does not meet the 1-hour regulation. In addition, the industry performed independent testing that the RES staff observed. The Industry testing showed failures of the Hemyc fire barrier in the 27 - 47 minute range, which does not meet the 1-hour regulation.

The staff requested the Committee's endorsement to issue the generic letter.

A representative from the Nuclear Energy Institute (NEI) provided its view on the staff's draft final generic letter. NEI said this generic letter is another example of a generic communication imposing new regulatory requirements, and the NRC has the right to amend its regulations but this should be done in the more disciplined rulemaking process. The staff said they are using the generic letter appropriately. The staff stated the generic letter is not imposing a new burden and does not constitute a backfit according the Committee to Review Generic Requirements.

Committee Action

The Committee issued a letter to the NRC Executive Director for Operations dated December 21, 2005, recommending the staff issue the Generic Letter, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Fire Protection Regulations."

VI. Meeting with the NRC Commissioners, Commissioners' Conference Room, One White Flint North, Rockville, MD (Open)

The Committee met with the NRC Commissioners to discuss items of mutual interest. Topics for discussion included license renewal, early site permits, new plant licensing, proposed alternative embrittlement criteria in 10 CFR 50.46, fire protection matters, power uprate technical issues, and future ACRS activities.

In a Staff Requirements Memorandum (SRM) dated December 20, 2005, resulting from this meeting, the Commission requested that the ACRS inform the Commission of its plans to manage the increased workload resulting from the anticipated receipt of new reactor designs and combined license applications. Also, the ACRS shall make among its highest priorities its role in the resolution of GSI-191.

Committee Action

This was an information briefing. The Committee plans to consider this matter during future meetings as further progress has been made by the staff.

VII. Proposed Program Plan and Advance Notice of Proposed Rulemaking for Risk-Informing 10 CFR Part 50 (Open)

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

The Committee met with the NRC staff to discuss a draft Commission Paper which outlines a formal program plan to make a risk-informed and performance-based revision to 10 CFR Part 50, including revisions to the applicable Regulatory Guides, Standard Review Plan, or other guidance documents. The program plan was developed in response to the Commission's SRM dated May 9, 2005. The draft Commission Paper also included an Advance Notice of Proposed

Rulemaking that seeks stakeholder feedback on rulemaking approaches for making technical requirements for power reactors more risk-informed and performance-based.

VIII. Staff Activities Associated with Responding to the Commission's Staff Requirements Memorandum (SRM) related to Safety Conscious Work Environment and Safety Culture (Open)

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

ACRS member Mario Bonaca opened the meeting by describing NRC's initiative and associated objectives to enhance the reactor oversight process to more fully address safety culture. Dr. Bonaca then turned the meeting over to Mr. Michael Johnson, Director of the Office of Enforcement (OE). Mr. Johnson initiated the staff's presentation by introducing other staff members which included Isabelle Schoenfeld, Chief of the Safety Culture Working Group, and James Cobey, Chief of the Reactor Project's Branch III, Region I. Mr. Johnson briefly summarized the ACRS letter that resulted from their last meeting on safety culture several years ago, and the purpose of the meeting to bring the Committee up-to-date on what had transpired in the intervening years.

Mr. Johnson turned the meeting over to Ms. Schoenfeld to provide background information and to summarize the drivers of the work. Ms. Schoenfeld indicated that there was strong Congressional interest in the staff's initiative, specifically from Senators on the Environment and Public Works Committee. She summarized the direction provided by the Commission in SRM-SECY 04-0111, to enhance the reactor oversight process (ROP) treatment of cross-cutting issues, to ensure inspectors are trained on safety culture, to determine when to do specific safety culture evaluations, to continue to monitor of industry and international efforts on safety culture, and to involve stakeholders in making changes to the ROP. Ms. Schoenfeld then described what the Commission told the staff not to do, including not to revise the 1989 policy statement, not to develop an inspection process for systematically assessing safety culture, not to use NRC surveys of licensee personnel, not to proactively work with the international community to develop objective performance indicators, not to engage the industry to develop an industry process to address safety culture, and not to develop intervention strategies when the licensee had failed to take appropriate action. Several members questioned what was meant by "not to do" and Mr. Johnson indicated that the Commission wanted the staff to do more, and not simply "touch" what industry was doing, or rely on performance indicators.

Dr. Wallis questioned whether other countries used performance indicators and Mr. Persensky, RES, responded that the staff is reviewing a draft IAEA document on safety culture, and that the international community was coming up with a similar approach as the staff, i.e., using multiple measures rather than indicators. Dr. Apostolakis questioned why after all these years, licensees still do not have a good safety culture. Mr. Johnson stated that Davis-Besse was a watershed event that indicated that a plant can have green indicators yet still have underlying problems with respect to safety culture. Dr. Bonaca questioned the staff that in hindsight, could not one identify indicators at Davis Besse that safety culture was degraded, and whether there were problems with the ROP itself. Mr. Johnson indicated that at the time, the staff probably did not do a good job in terms of questioning, documenting, or being able to handle safety culture problems within the process, and bring it forward for action. Mr Johnson stated that one of the things they plan to do is to go back to Davis-Besse after changes to the ROP, to see if the staff would now be better able to address the issues of Davis-Besse.

Dr. Apostolokis returned to his question on why there was deterioration to begin with and Mr. G. Cobey from Region I responded that safety culture was really at the heart of the problem, and the staff just did not recognize it as the root cause. Mr. Sieber indicated that a plant with a safety culture that is deteriorating as a sleeper, could look pretty good if measuring only certain things like corrective action work. He indicated that the Agency needs to be proactive in looking for indication that the standards are low, and the degree of inquisitiveness of employees is low.

Ms. Schoenfeld then described what had been done organizationally to respond to the Commission's direction, which included establishing a Steering Committee headed by Mr. Johnson, a Safety Culture Working Group, and a support Regional team led by Gene Cobey. Ms. Schoenfeld indicated that a comprehensive review of safety culture approaches had been performed, including the international community, and a Commission paper issued on the status and schedule of the staff's initiative. Ms. Schoenfeld indicated that the Commission turned the paper into a notation vote, and they were still waiting for an SRM. She also stated that several stakeholder meetings had been held.

Dr. Bonaca commented that to really look at safety culture issues one would have to be intrusive, but the direction the staff is taking discourages intrusiveness, i.e., feedback from industry is do not get too close. Mr. Johnson indicated that the focus is in deciding what belongs to industry and what belongs to NRC. Mr. Johnson stated that during the staff's meeting with the Commission on lessons learned from Davis-Besse, the Commission indicated that the staff was on the wrong path, and they should take a fresh start, i.e., re-engage with stakeholders.

Mr. Cobey described the November 29th and 30th stakeholder meetings that followed as a very productive meeting. Attendees included Billy Garr, David Lochbaum, Paul Blanche, NEI, INPO, Dave Collins. He indicated that the meeting was facilitated by NRC/OGC Chip Cameron, and that the meeting built on ten conceptual approaches, that finally lead the group to one optimum approach entitled Option G. Mr. Cobey indicated that by the end of the meeting they had general agreement on how to proceed on all issues.

Mr. Johnson indicated that knowing what industry had done in the wake of Davis-Besse gave everyone a better understanding on where they were, and where they wanted to go. Mr. Cobey described the previous December 15th stakeholder meeting as a meeting to discuss what's important to safety culture and to come closer in alignment on how to address it in the ROP. Mr. Cobey described Option G as a four-element framework, information sources, documentation, assessment, follow-up actions. The information sources regarding plant status activities performed by residents would be unchanged, it would leave baseline inspection procedures unchanged, except for IP 71152 Problem Identification and Resolution. The Special Inspection Procedures for event follow-up would be enhanced, but NRC's Inspection of

Allegations would remain unchanged. Cross-cutting aspects of findings would remain unchanged. Documentation including inspection reports and letters would remain unchanged.

Dr. Apostolokis questioned whether inspectors knew what to look for, and Mr. Cobey indicated that inspectors already knew what cross-cutting aspect to look for so this would not change. Dr. Bonaca indicated that the ROP does not count repeat events. Mr. Cobey indicated that although it is not based on a strict count, a sufficient number of repeat events greater than minor would be looked at for cross-cutting aspects. Otherwise, under Option G, licensee's would be expected to correct minor issues, and the staff would not become engaged until it became a more significant issue, and then using a graded approach.

Dr. Wallis questioned whether any enhancements would have detected the issues at Davis-Besse before it became significant. Mr. Johnson stated that the staff was going to look at that. Mr. Cobey indicated that NRC's assessment process in Manual Chapter 0305 would be largely unchanged by Option G, but would be adjusted to make crosscutting issues more closely align with what is important to safety culture, i.e., it would identify 15-16 items important to safety culture and link them to crosscutting issues.

Dr. Apostolakis questioned the use of resources in evaluating safety culture, and Mr. Johnson agreed that the staff would need to understand it in the context of performance. Mr Cobey then indicated that the second envisioned change to the framework would be the outputs of the allegation and traditional processes as inputs into the assessment process, and licensees may have to do a safety culture assessment, either by themselves or an independent party.

Drs. Apostolakis and Wallis, respectively, questioned Mr. Cobey on how one would judge the adequacy. Mr. Cobey indicated that the NRC would review the results of the self-assessment for reasonableness, but the details have not been worked out yet, and it would become part of the December 15th stakeholder meeting that is being planned. Mr Cobey indicated that the staff had to be careful on how to communicate the results in a public arena so as not to create an adverse effect. Dr. Powers questioned holding back information from the public, but Mr. Cobey indicated that it was only with respect to certain organizations within the plant that were unwilling to raise issues, not to label individuals that are then reluctant to even speak to the NRC.

Mr. Cobey discussed what would happen as a plant's performance deteriorates and moved across the action matrix from left to right. For a white finding, only a minimal change in follow-up action would occur, to validate that the licensee did a root cause analysis, and if safety culture is a driver, that there is appropriate corrective actions. The next column (where the licensee would have two white findings under the same cornerstone) the staff would perform IP 95002. The procedure would be enhanced to determine whether safety culture was a driver, and whether an independent safety culture assessment needed to be performed. Dr. Powers questioned why INPO's assessment would not be adequate and what was meant by "independent," and Mr. Cobey indicated that the INPO review could be used depending on when it had been done, and Mr. Johnson indicated that NRC had lots of success with independent safety assessments. Mr. Sieber questioned whether INPO plant evaluation is a safety culture assessment. Ms. Schoenfeld indicated that INPO did do safety culture evaluations. From the audience, Tony Harris, a "loanee" to NEI from STARS Alliance, stated that in the wake of Davis-Besse and in response to INPO Significant Operating Experience Report (SOER) - 024, every licensee performed a safety culture assessment, and INPO looks at those evaluations every two years. Fundamental to all this is the problem identification and resolution program. At Davis-Besse, Mr. Harris indicated that they were pushing things out and this could indicate a problem.

Dr. Apostolakis requested that the staff address INPO's attributes at the upcoming Subcommittee. Mr. Johnson indicated that it would be addressed.

Mr. Cobey then addressed NRC's response when a plant moves into a multiple repetitive degraded cornerstone on the action matrix. NRC would evaluate what is important to the licensee's safety culture to determine whether licensee's assessment, performance improvement plan, and their corrective actions were adequate to address the problems. As performance degrades, NRC would become more intrusive. For example, IP 95001 would be used to see if a license included what's important to safety culture in their corrective actions, IP 95002 would be used to see what is important to the licensee's safety culture and if they were drivers of poor performance. If performance degraded further, NRC would independently perform a safety culture evaluation.

Dr. Bonaca questioned what was important to safety culture, and Mr. Cobey indicated that they were called "components" or "subcomponents," and Mr. Johnson indicated that they have a good set to bring to the subcommittee.

Mr. Cobey indicated that they still have a lot of work to do including revising the plan, holding additional meetings, conduct training for inspectors, test the plan, and go back to see if it would adequately address Davis Besse, and be ready to implement the process by March 2006. In closing, Dr. Bonaca indicated that the Committee would write a letter on NRC's safety culture initiative once a product is provided, and would discuss ACRS proactive initiative further following the ACRS Retreat. Dr. Ransom questioned whether change in management issues had been considered, and Mr. Cobey indicated that it would be considered in the supplemental type of inspection and to discuss it further at the Subcommittee in January. Dr. Apostolakis stated that this would be a very important Subcommittee meeting.

<u>Committee Action</u>

This was an information briefing with no Committee action at this time. A Joint Subcommittee on Human Factors and Reliability & Probabilistic Risk Assessment will be held January 25, 2006, to examine current status of NRC's safety management/culture initiatives, and associated approaches to address safety culture, including international experience. Following that meeting, the Committee will decide whether to write a letter. A March letter may be written if the staff presents a product on safety culture to the Full Committee for review.

IX. Election of ACRS Officers for CY 2006 (Open)

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

The Committee re-elected Graham B. Wallis as ACRS Chairman, William J. Shack as ACRS Vice-Chairman, and John D. Sieber as Member-at-Large for the Planning and Procedures Subcommittee.

X. Executive Session (Open)

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

A. Reconciliation of ACRS Comments and Recommendations/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 response of November 28, 2005, to comments and recommendations included in the ACRS interim letter dated October 19, 2005, concerning the safety aspects of the license renewal application for the Browns Ferry Nuclear Plant Units 1, 2, and 3. The Committee decided that it was satisfied with the EDO's response.

  The EDO response stated that a licensee commitment to implement operating experience and aging management program reviews before entering the period of extended operation will be included in the safety evaluation report for the Browns Ferry extended power uprate application currently under review. The staff also committed to address the Committee's concerns regarding Unit 1 operating experience, the Unit 1 Periodic Inspection Program, and the terminology used for license-renewal related inspections in the final safety evaluation report related to the license renewal of the Browns Ferry Nuclear Plant Unit 1, 2, and 3.

- The Committee considered the EDO's November 3, 2005 response to the ACRS letter of September 23, 2005, concerning the Committee's review of the proposed technical basis for revision of the embrittlement criteria in 10 CFR 50.46(b). The Committee decided that it was satisfied with the EDO's response.

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

The Committee heard a report from the ACRS Chairman and the Executive Director, ACRS, regarding the Planning and Procedures Subcommittee meeting held on December 7, 2005. The following items were discussed:

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

Member assignments and priorities for ACRS reports and letters for the December ACRS meeting was 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 March 2006 was discussed. The objectives were:

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

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

- Meeting with the NRC Commissioners

The ACRS met with the NRC Commissioners on Thursday, December 8, 2005 to discuss the following topics:

  - Overview (GBW)
    - Major Accomplishments
    - License Renewal
    - Early Site Permits
    - Future ACRS Activities
  - Issues Related to New Plant Licensing (including technology Neutral Framework) (TSK/MME)
  - Proposed Alternative Embrittlement Criteria in 10 CFR 50.46 (DAP/RC)
  - Fire Protection Matters (GEA/JGL)
  - Power Uprate Technical Issues (RSD/RC)

Any follow-up items resulting from this meeting and a course of action for addressing them should be discussed by the Committee.

- <u>ACRS Retreat in 2006</u>

During its November 2005 meeting, the Committee approved a list of topics, and lead member assignments, for discussion during the retreat which is scheduled to be held on January 26-27, 2006. A proposed schedule for the retreat was discussed. It is suggested that this meeting be held at the Marriott Hotel in Rockville.

- <u>Candidates for Potential Membership on the ACRS</u> (Closed)

The ACRS Member Candidate Screening Panel sent its recommendations to the Commission in November 2005 for appointment of two ACRS members to fill the vacancies on the Committee in the areas of Materials and Metallurgy and Plant Operations. Additionally, there are two potential candidates for membership on the ACRS with expertise in thermal-hydraulics and other areas. The Panel is seeking additional candidates.

- <u>Election of Officers for CY 2006</u>

During its December 2005 meeting, the Committee will elect Chairman and Vice Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee. During the November ACRS meeting, it was requested that those members who do not wish to be considered for all or any of these offices should inform the ACRS Executive Director in writing by November 21, 2005.

- <u>Quadripartite Meeting Status</u>

On December 5, 2005, a planning meeting among the Quadripartite members (ACRS, GPR, NSC, RSK) was held in Germany to discuss and finalize logistical and technical issues of interest including the format of the abstracts which are due on February 28, 2006. The ACRS Executive Director and Mugeh Afshar-Tous attended this meeting. As a result, few changes were made to the agenda to address the requests of all Quadripartite members. Also, the members agreed to extend an invitation to a representative from Finland to attend the Quadripartite Meeting.

- <u>Worksheets on Skill Set</u>

During the November ACRS meeting worksheets were provided to the members and staff to help identify their technical expertise. The following modifications were made to the worksheets based on comments received: emergency planning has replaced evacuation planning; safeguards & security was added to the list of specific expertise; plant operating experience was subsumed into reactor operating experience. The worksheets will now be used to develop a working draft on ACRS technical expert needs and options to address gaps in technical expertise. The working draft will be distributed to members and staff prior to the January 2006 retreat.

- <u>Other Matters</u>

Jocelyn Mitchell, RES, informed the Subcommittee about a proposed RES plan for ranking the NRC research projects and how ACRS input on quality review will be factored into the overall ranking.

C. <u>Future Meeting Agenda</u>

Appendix IV summarizes the proposed items endorsed by the Committee for the 529th ACRS Meeting, February 9-11, 2006.

The 528th ACRS meeting was adjourned at 12:30 p.m. on December 10, 2005.

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

February 27, 2006

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

FROM: Graham B. Wallis
ACRS Chairman

Graham B. Wallis

SUBJECT: CERTIFIED MINUTES OF THE 528th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), DECEMBER 7-10, 2005

I certify that based on my review of the minutes from the 528TH 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.

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–146, License No. DPR–4]

Saxton Nuclear Experimental Corporation and GPU Nuclear, Inc.; Notice of Termination of Saxton Nuclear Experimental Corporation Facility Amended Facility License No. DPR–4

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of Termination of the Saxton Nuclear Experimental Corporation (SNEC) and GPU Nuclear, Inc., SNEC Facility Amended Facility License No. DPR–4.

SUMMARY: The Nuclear Regulatory Commission (NRC) is noticing the termination of Amended Facility License No. DPR–4 (NRC Docket No. 50–146) for the SNEC facility near Saxton, Pennsylvania.

Background: The SNEC facility is in Bedford County, Pennsylvania. The SNEC facility operated from 1962 to 1972. It was mainly used for research in various aspects of power reactor technology and to train personnel. The reactor was licensed at 23.5 megawatts of thermal energy. Electric power was produced by sending steam produced by operation of the reactor to a nearby coal-fired power station (because the SNEC facility did not have its own turbine or generator). The nuclear steam supply system was a one-loop pressurized water reactor. After shutdown, the reactor fuel was removed from the facility and shipped to what is now the Department of Energy Savannah River Site in South Carolina. Some minor decommissioning activities were done from 1972 to 1974. The facility was then placed in a monitored storage condition. Support structures and buildings were decontaminated and removed between 1987 and 1992. Full-scale decommissioning activities started in May 1998.

In February 2000 the licensees submitted their license termination plan (LTP) for the SNEC facility. Under the provisions of 10 CFR 50.82(a)(10), the NRC approved the LTP by a license amendment dated March 28, 2003. In accordance with the approved LTP, the licensees conducted final status surveys (FSSs) to demonstrate that the facility and site met the criteria in 10 CFR 20.1402 for unrestricted release. The licensees presented the FSS results to the NRC in FSS reports (FSSRs).

The licensees submitted an application for termination of SNEC Amended Facility License No. DPR–4 on September 15, 2005. The application states that GPU Nuclear, Inc., has completed the remaining radiological decommissioning activities and the final radiation surveys of the SNEC Facility and the associated PENELEC site in accordance with an NRC-approved LTP and the final radiation surveys demonstrate that the facility and site area meet the criteria in 10 CFR part 20, subpart E, for the decommissioning and release of the site for unrestricted use.

The NRC did a number of performance-based in-process inspections of the licensee's FSS program during the decommissioning process. The purpose of the inspections was to verify that the FSSs were being done in accordance with the licensees' commitments in the LTP and to evaluate the quality of the FSSs by reviewing the FSS procedures, methodology, equipment, surveyor training and qualifications, document quality control, and survey data. The NRC also did independent confirmatory surveys to verify the licensees' FSS results. The confirmatory surveys consisted of surface scans for beta and gamma radiation, direct measurements for total beta activity, and smear sampling for determining removable-radioactivity levels.

The NRC staff reviewed the FSSRs and concludes that (i) dismantlement and decontamination activities were performed in accordance with the approved LTP; and (ii) the FSSRs demonstrate that the facility and site have met the criteria for decommissioning in 10 CFR part 20, subpart E. NRC is therefore terminating SNEC Facility Amended Facility License No. DPR–4.

FOR FURTHER INFORMATION CONTACT: See the application for license termination dated September 15, 2005 (ML052640047) and NRC Inspection Report Nos. 50–146/2003–201, dated November 12, 2003 (ML033090608), 50–146/2003–202, dated December 17, 2003 (ML033420687), 50–146/2004–201, dated February 10, 2005 (ML050380407), and 50–146/2005–201, dated October 31, 2005 (ML052730465). They are available for public inspection at the Commission's Public Document Room (PDR) at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management System's (ADAMS's) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html* (use the ADAMS ML numbers given above). Persons who do not have access to ADAMS or who have trouble accessing the documents in ADAMS should contact the NRC PDR reference staff by telephone at 1–800–397–4209 or 301–415–4737 or by e-mail to *pdr@nrc.gov.*

Dated at Rockville, Maryland this 7th day of November 2005.

For The Nuclear Regulatory Commission.

Brian E. Thomas,

Branch Chief, Research and Test Reactors Branch, Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation.

[FR Doc. E5–6414 Filed 11–21–05; 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 December 7–10, 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).

Wednesday, December 7, 2005, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

1 p.m.–1:05 p.m.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

1:05 p.m.–3 p.m.: Final Review of the Vermont Yankee Extended Power Uprate Application and the Associated Safety Evaluation (Open)—The Committee will hear presentations by and hold discussions with representatives of the Entergy Nuclear Operations, Inc. and the NRC staff regarding the 20% power uprate application for the Vermont Yankee Nuclear Plant and the NRC staff's associated Safety Evaluation.

3:30 p.m.–5:30 p.m.: Draft ACRS report on the NRC Safety Research Program (Open)—The Committee will discuss the draft ACRS report to the Commission on the NRC Safety Research Program.

5:45 p.m.–6:45 p.m.: Preparation for Meeting with the NRC Commissioners (Open)—The Committee will discuss the topics scheduled for discussion with the NRC Commissioners between 1 and 3 p.m. on Thursday, December 8, 2005.

Thursday, December 8, 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.: Early Site Permit Application for the Grand Gulf Nuclear Station and the Associated Final Safety Evaluation Report (Open)—The Committee will hear presentations by and hold discussions with representatives of the System Energy Resources, Inc. and the NRC staff regarding the early site permit application for the Grand Gulf Nuclear Station and the associated final Safety Evaluation Report prepared by the NRC staff.

10:15 a.m.–11:45 a.m.: Draft Final Generic Letter, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Fire Protection Regulations" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft final Generic Letter on "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Fire Protection Regulations" and a summary of the NRC staff's resolution of public comments received on the public comment version of this Generic Letter.

1 p.m.–3 p.m.: Meeting with the NRC Commissioners, Commissioners' Conference Room, One White Flint North, Rockville, MD (Open)—The Committee will meet with the NRC Commissioners to discuss the following topics: Overview by the ACRS Chairman (License Renewal, Early Site Permits, and Future ACRS Activities); Issues Related to New Plant Licensing (including Technology Neutral Framework); Proposed Alternative Embrittlement Criteria in 10 CFR 50.46; Fire Protection Matters; and Power Uprate Technical Issues.

3:30 p.m.–5 p.m.: Proposed Program Plan and Advance Notice of Proposed Rulemaking for Risk-Informing 10 CFR Part 50 (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the proposed Program Plan and the Advance Notice of Proposed Rulemaking for Risk-Informing 10 CFR Part 50, and related matters.

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

Friday December 9, 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.: Staff Activities Associated with Responding to the Commission's Staff Requirements Memorandum (SRM) Related to Safety Conscious Work Environment and Safety Culture (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding staff activities associated with responding to the Commission's SRM related to safety conscious work environment and safety culture, and related matters.

10:15 a.m.–11:15 a.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

11:15 a.m.–11:30 a.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

11:30 a.m.–12 Noon: Election of ACRS Officers for CY 2006 (Open)—The Committee will elect Chairman and Vice Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee for CY 2006.

1:30 p.m.–3:30 p.m.: Draft ACRS Report on the NRC Safety Research Program (Open)—The Committee will discuss the draft ACRS report to the Commission on the NRC Safety Research Program.

3:45 p.m.–6:45 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Saturday, December 10, 2005, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

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

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 29, 2005 (70 FR 56936). In accordance with those procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting. Persons desiring to make oral statements should notify the Cognizant ACRS staff named below five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Cognizant ACRS staff prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Cognizant ACRS staff if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, as well as the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Sam Duraiswamy, Cognizant ACRS staff (301–415–7364), between 7:30 a.m. and 4:15 p.m., e.t.

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

Videoteleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician (301–415–8066), between 7:30 a.m. and 3:45 p.m., e.t., at least 10 days before the meeting to ensure the availability of this

APPENDIX II

November 18, 2005

REVISED
SCHEDULE AND OUTLINE FOR DISCUSSION
528th ACRS MEETING
DECEMBER 7-10, 2005

WEDNESDAY, DECEMBER 7, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1)	1:00 - 1:05 P.M.	Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD) 1.1) Opening statement 1.2) Items of current interest
2)	1:05 - ~~3:00~~ **3:15** P.M.	Final Review of the Vermont Yankee Extended Power Uprate Application and the Associated Safety Evaluation (Open) (RSD/GBW/RC) 2.1) Remarks by the Subcommittee Chairman 2.2) Briefing by and discussions with representatives of the Entergy Nuclear Operations, Inc. and the NRC staff regarding the 20% power uprate application for the Vermont Yankee Nuclear Plant and the NRC staff's associated Safety Evaluation. Members of the public may provide their views, as appropriate.
	~~3:00 - 4:00~~ **P.M.** **3:15-4:20**	*****BREAK*****
3)	~~4:00 - 5:45~~ P.M. **4:20-4:50**	Draft ACRS Report on the NRC Safety Research Program (Open) (DAP/HPN/SD) 3.1) Remarks by the Subcommittee Chairman 3.2) Discussion of the draft ACRS report to the Commission on the NRC Safety Research Program.
	~~5:45 - 6:00~~ **P.M.** **4:50-5:15**	*****BREAK*****
4)	~~6:00 - 6:45~~ P.M. **5:15-6:15**	Preparation for Meeting with the NRC Commissioners (Open) (GBW, et. al/JTL, et. al) Discussion of the following topic scheduled for discussion with the NRC Commissioners on December 8, 2005: I Overview (GBW) • License Renewal • Early Site Permits • Future ACRS Activities II Issues Related to New Plant Licensing (including Technology-Neutral Framework) (TSK) III Proposed Alternative Embrittlement Criteria in 10 CFR 50.46 (DAP)

IV Fire Protection Matters (GEA)
V Power Uprate Technical Issues (RSD)

THURSDAY, DECEMBER 8, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

5) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)

6) 8:35 - ~~10:15~~ **10:00** A.M. Early Site Permit Application for the Grand Gulf Nuclear Station and the Associated Final Safety Evaluation Report (Open) (DAP/MME)

6.1) Remarks by the Subcommittee Chairman

6.2) Briefing by and discussions with representatives of the System Energy Resources, Inc. and the NRC staff regarding the early site permit application for the Grand Gulf Nuclear Station and the associated final Safety Evaluation Report prepared by the NRC staff.

~~10:15~~ 10:00 - 10:30 A.M. ***BREAK*****

7) 10:30 - ~~11:45~~ **10:50** A.M. Draft Final Generic Letter, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Fire Protection Regulations" (Open) (RSD/JGL)

7.1) Remarks by the Subcommittee Chairman

7.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final Generic Letter on "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Fire Protection Regulations" and a summary of the NRC staff's resolution of public comments received on the public comment version of this Generic Letter.

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

11:45 - 1:00 P.M. ***LUNCH*****

8) 1:00 - 3:00 P.M. Meeting with the NRC Commissioners, Commissioners' Conference Room, One White Flint North, Rockville, MD (Open) (GBW, et. al/JTL, et. al)
Meeting with the NRC Commissioners to discuss the topics listed under Item 4.

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

9) 3:30 - 5:00 P.M. Proposed Program Plan and Advance Notice of Proposed Rulemaking for Risk-Informing 10 CFR Part 50 (Open) (WJS/GEA/MRS/EAT)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed Program Plan and the Advance Notice of Proposed Rulemaking for Risk-Informing 10 CFR Part 50, and related matters.

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

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

10) 5:15 - 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

10.1) Final Review of the Extended Power Uprate Application for the Vermont Yankee Nuclear Plant (RSD/GBW/RC)

10.2) Final Review of the Early Site Permit Application for the Grand Gulf Nuclear Station (DAP/MME)

10.3) Draft Final Generic Letter, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Fire Protection Regulations" (RSD/JGL)

10.4) Proposed Program Plan and Advance Notice of Proposed Rulemaking for Risk-Informing 10 CFR Part 50 (WJS/GEA/MRS/EAT)

FRIDAY, DECEMBER 9, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

11) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)

12) 8:35 - 10:00 A.M. Staff Activities Associated with Responding to the Commission's Staff Requirements Memorandum (SRM) related to Safety Conscious Work Environment and Safety Culture (Open) (MVB/GEA/JHF)

12.1) Remarks by the Subcommittee Chairman

12.2) Briefing by and discussions with representatives of the NRC staff regarding staff activities associated with responding to the Commission's SRM related to safety conscious work environment and safety culture, and related matters.

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

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

13) 10:15 - 11:15 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GBW/JTL/SD)

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

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

14) 11:15 - 11:30 A.M. Reconciliation of ACRS Comments and Recommendations (Open) (GBW, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

15) 11:30 - 12:00 Noon Election of ACRS Officers for CY 2006 (Open) (JTL/SD)
Election of Chairman and Vice Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee.

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

16) 1:30 - 3:30 P.M. Draft ACRS Report on the NRC Safety Research Program (Open) (DAP/HPN/SD)
Discussion of the draft ACRS report to the Commission on the NRC Safety Research Program.

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

17) 3:45 - 6:45 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports listed under Item 10.

SATURDAY, DECEMBER 10, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

18) 8:30 - 12:30 P.M. Preparation of ACRS Reports (Open)
Continue discussion of the proposed ACRS reports listed under Item 10.

~~19) 12:30 - 1:00 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
528th ACRS MEETING
DECEMBER 7-10, 2005

NRC STAFF (12/7/05)
R. Ennis, NRR
B. Dennig, NRR
C. Wu, NRR
T. Scarborough, NRR
C. Holden, NRR
M. Stutzke, NRR
J. Stang, NRR
H. Chernoff, NRR
Q. Nguyen, NRR
P. Lyons, OCM
D. Coe, OCM/PBL
L. Lambros, NRR
J. Bongarra, NRR
G. Imbro, NRR
H. Garg, NRR
D. Roberts, NRR
J. Tatum, NRR
C. Boyd, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
J. Dreyfuss, Entergy-VY
L. Gucha, Entergy
M. Dick, GE-Entergy
J. Head, Entergy
J. McCann, Entergy
P. Perez, Areva
B. Slifey, Entergy-VY
M. Palionis, Entergy-VY
J. Callaghan, Entergy-VY
P. Rainey, Entergy-VY
E. Duda, Entergy-VY
C. Nichols, Entergy-VY
J. Oddu, PPL-Susquehanna
B. Croke, Entergy-VY
L. Quintana, GE Energy
D. Yasi, Stone & Webster
M. Detamore, PPL-Susquehanna
V. Andersen, Erin Engineering
E. Betti, Entergy-VY
B. Hobbs, Entergy-VY
B. McVren, VY
N. Rademaker, Entergy-VY
D. Langley, TVA
J. Wolcott, TVA
D. Fappone, GE
J. Thayer, Entergy-VY
S. Zinda, McMaster University
S. Hambric, Penn State U
S. Hofman, VT Dept of Public Service
W. Sherman, VT Dept of Public Service
M. F. Repko, US Senate Env. Committee
P. J. Atherton, Public
R. Shadis, New England Coalition

Appendix
528th ACRS Meeting

NRC Staff (12/8/05)

G. Mizuno, OGC
R. Woods, RES
N. K. Bagchi, NRR
B. Richter, NRR
J. Monninger, RES
A. Klein, NRR
Q. Nguyen, NRR
D. Frumkin, NRR
R. Gallucci, NRR
P. Qualls, NRR
R. Barrett, RES
J. Birmingham, NRR
E. McKenna, NRR
C. Ader, RES
J. Lyons, NRR
N. Kadambi, RES
D. Harrison, NRR
C. Grimes, NRR
C. Holden, NRR
R. Ennis, NRR
K. Campe, NRR
M. Blumberg, NRR
R. Anand, NRR
N. Patel, NRR
J. Lee, NRR
B. Harvey, NRR
L. Dudes, NRR
W. Beckner, NRR
G. Bagchi, NRR
Y. Li, NRR
T. Cheng, NRR
P. Prescott, NRR
M. Kotzalas, NRR
J. Calvo, NRR
R. Weisman, NRR
J. Dixon-Herrity, OEDO
A. Lavretta, NRR
M. Salley, RES
S. Weerakkody, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

K. Sutton, Morgan Lewis
G. Cesare, Enercon
G. Young, Entergy
A. Schneider, Enercon
W. K. Hughey, Entergy
M. Bourgeois, Entergy
G. Zinke, Entergy
W. Lettis, WLA
B. Maher, Exelon
A. Marion, NEU
B. Jamar, NEI
D. Pappone, Ge Nuclear
N. Chapman, SERCH/Bechtel
B. Haipelin, CQ
B. Slifer, entergy-VY
D. Yasi, Stone & Webster
B. Hobbs, Entergy-VY
C. Nichols, Entergy-VY
L. Gucwa, Entergy-VY
P. Rainey, Entergy-VY
S. Hofman, VT Dept. of Public Service
W. Sherman, VT Dept. of Public Service

Appendix
528th ACRS Meeting

NRC Staff (12/9/05)
M. Johnson, OE
J. Persensky, RES
V. Barnes, RES
R. Lerch, RIII
F. Guether, NRR
T. Ghosh, NMSS
I. Schoenfeld, OE
L. Jarriel, OE
J. Mitchell, RES
M. Stutzke, NRR
C. Holden, NRR
D. Roberts, NRR
L. Smith, RIV

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
T. Harris, NEI
S. Hoffman, VT Dept. of Public Service
B. Hobbs, Entergy-VY
C. Nichols, Entergy-VY
W. Sherman, VT Dept. of Public Service

NRC Staff (12/10/05)
M. Stutzke, NRR
R. Ennis, NRR
C. Holden, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC (12/10/05)
S. Hofman, VT Dept. of Public Service
C. Nichols, Entergy
B. Hobbs, Entergy
L. Gucwa, Entergy

January 25, 2006

REVISED
SCHEDULE AND OUTLINE FOR DISCUSSION
529th ACRS MEETING
FEBRUARY 9-11, 2006

THURSDAY, FEBRUARY 9, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - 10:00 A.M. Evaluation of Human Reliability Analysis (HRA) Methods Against Good Practices (Open) (GEA/EAT)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff regarding the draft NUREG report on the Evaluation of HRA Methods Against Good Practices specified in NUREG-1792, "Good Practices for Implementing Human Reliability Analysis."

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

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

3) 10:15 - 11:45 A.M. Proposed Revisions to SRP Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs" (Open) (RSD/JGL)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding proposed revisions to the Standard Review Plan (SRP) Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," and related matters.

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

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

4) 12:45 - 2:15 P.M. FERRET Reactor Vessel Fluence Methodology (Open) (RSD/CS)

4.1) Remarks by the Subcommittee Chairman

4.2) Briefing by and discussions with representatives of the NRC staff regarding the FERRET methodology which is used to predict the fluence on the reactor vessel wall due to neutron leakage from the core.

Representatives of the Westinghouse Electric Corporation may participate, as appropriate.

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

5) 2:30 - 5:00 P.M. Draft ACRS Report on the NRC Safety Research Program (Open) (DAP/HPN/SD)

Discussion of the draft ACRS report to the Commission on the NRC Safety Research Program.

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

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

Discussion of proposed ACRS reports on:

6.1) Evaluation of Human Reliability Analysis Methods Against Good Practices (GEA/EAT)

6.2) Proposed Revisions to SRP Section 14.2.1 (RSD/JGL)

6.3) Response to the Commission SRM dated December 20, 2005 regarding ACRS plans to manage the anticipated increased workload in the areas of advanced reactor designs and combined license applications (TSK/JHF)

FRIDAY, FEBRUARY 10, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

8) 8:35 - 9:30 A.M. Subcommittee Reports (Open)

8.1) Report by and discussions with the Chairman of the ACRS Subcommittee on Plant License Renewal regarding interim review of the Brunswick Nuclear Plant license renewal application and the associated NRC staff's draft Safety Evaluation Report (JDS/JGL).

8.2) Report by and discussions with the Chairman of the ACRS Subcommittee on Human Factors regarding the Safety Conscious Work Environment and Safety Culture (MVB/JHF).

8.3) Report by and discussions with the Chairman of the ACRS Subcommittee on Thermal-Hydraulic Phenomena regarding proposed Revision 4 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (GBW/RC).

8.4) Report by and discussions with the Chairman of the ACRS Subcommittee on Regulatory Policies and Practices regarding the draft Regulatory Guide, "An Approach for Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements" (WJS/MRS).

9:30 - 9:45 A.M. ***BREAK***

9) 9:45 - 12:15 P.M. **(11:00-11:15 A.M. BREAK)** Draft ACRS Report on the NRC Safety Research Program (Open) (DAP/HPN/SD)
Discussion of the draft ACRS report to the Commission on the NRC Safety Research Program.

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

10) 1:15 - 2:15 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GBW/JTL/SD)

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

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

11) 2:15 - 2:30 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.

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

12) 2:45 - 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

12.1) Evaluation of Human Reliability Analysis Methods Against Good Practices (GEA/EAT)

12.2) Proposed Revisions to SRP Section 14.2.1 (RSD/JGL)

12.3) Response to the Commission SRM dated December 20, 2005 regarding ACRS plans to manage the anticipated increased workload in the areas of advanced reactor designs and combined license applications (TSK/JHF)

<u>SATURDAY, FEBRUARY 11, 2006, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND</u>

13)	8:30 - 12:30 P.M. **(10:30-10:45 A.M. BREAK)**	<u>Preparation of ACRS Reports</u> (Open) Continue discussion of the proposed ACRS reports listed under Item 12, and the draft ACRS report on the NRC Safety Research Program as needed.
14)	12:30 - 1:00 P.M.	<u>Miscellaneous</u> (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.

<u>NOTE:</u>

- **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
528th ACRS MEETING
December 7-10, 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 December 7-10, 2005

2 Final Review of the Vermont Yankee Extended Power Uprate Application and the Associated Safety Evaluation

2. Extended Vermont Yankee Extended Power Uprate presentation by Entergy [Viewgraphs]
3. NRC Staff Review of Proposed Extended Power Uprate for Vermont Yankee Nuclear Power Station presentation by NRR [Viewgraphs]
4. Vermont Yankee Power Uprate [Handout #1]
   a. Draft Summaries (non proprietary/proprietary sections) of the 11/29-30/2005 Power Uprate Subcommittee Meeting
   b. VY Response to staff RIAs
   c. Letter report by G. Leitch
   d. Letter reports by S. Banerjee
   e. Letter/emails/comments from members of the public re: VY Power Uprate
5. Vermont Yankee Extended Power Uprate Containment Overpressure Credit presentation by B. Sherman, VT Dept of Public Service [Viewgraphs]
6. Pilot Engineering and Design Inspection at Vermont Yankee presentation by R. Shadis, New England Coalition [Viewgraphs]

6 Early Site Permit Application for the Grand Gulf Nuclear Station and the Associated Final Safety Evaluation Report

7. Early Site Permit Application for Grand Gulf Nuclear Station presentation by Entergy, System Energy Resources, Inc. [Viewgraphs]
8. Presentation to the ACRS on the Early Site Permit Application for the Grand Gulf Site by R. Anand, NRR [Viewgraphs]
9. Grand Gulf Nuclear Station Early Site Permit DSER Open Item Resolution Summary Table [Handout]

Appendix
528th ACRS Meeting

7 Draft Final Generic Letter, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with First Protection Regulations"

10. Final Draft Generic Letter on Hemyc and MT Fire Barriers presentation by NRR [Viewgraphs]
11. Final Draft Generic Letter on Hemyc and MT Fire Barriers presentation by A. Lavretta and D. Fumkin [Viewgraphs]

8 Meeting with the NRC Commissioners, Commissioners Conference Room, One White Flint North, Rockville, MD

See Meeting Notebook FACA File

13 Future ACRS Activities/Report of the Planning and Procedures Subcommittee

12. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - December 6, 2005 [Handout #13.1]

Appendix
528th ACRS Meeting

MEETING NOTEBOOK CONTENTS

TAB — DOCUMENTS

2 Vermont Yankee Nuclear Power Station Extended Power Uprate

1. Table of Contents
2. Proposed Schedule
3. Status Report
4. List of References
5. Written Comments received from the Public

6 Early Site Permit Application for the Grand Gulf Nuclear Station and the Associated Final Safety Evaluation Report

6. Table of Contents
7. Proposed Agenda
8. Status Report
9. ACRS Interim Letter dated June 14, 2005
10. EDO response to ACRS Interim Letter dated August 12, 2005
11. FSER/Appendix A dated October 21, 2005

7 Draft Final Generic Letter 2005-XX, "Impact of Potentially Degraded HEMYC/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs"

12. Proposed Agenda
13. Status report
14. Memorandum from J. Lyons, NRR to J. Larkins, ACRS, dated November 2005, Subject: Proposed Generic Letter 2005-XX, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs"
15. Memorandum from J. Larkins, ACRS, to L. Reyes, EDO, dated July 7, 2005, Subject: Proposed Generic Letter 2005-XX, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs"

9 Proposed Program Plan and Advanced Notice of Proposed Rulemaking for Risk-Informing 10 CFR Part 50

16. Table of Contents
17. Proposed Schedule
18. Status Report
19. Draft SECY-05-XXXX "Staff Plan to make a Risk-Informed, Performance Based Revision to 10 CFR Part 50
20. ACRS Committee Letter Report, "Report on Two Policy Issues Related to New Plant Licensing," to Chairman Diaz dated September 21, 2005
21. Draft Program Plan for Risk-Informed, Performance-Based Part 50 Reactor Requirements
22. Draft Advanced Notice of Proposed Rulemaking on Approaches to Risk-Inform and Performance-Base the Requirements for Nuclear Power Reactors, RIN AH81
23. ACRS Committee Letter Report, "Draft Commission Paper on 'Risk-Informed

Alternatives to the Single Failure Criterion',” to Chairman Diaz dated June 10, 2005

12 Safety Culture

24. Introductory Remarks
25. NRC Presentation
26. Committee Discussion

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
528th FULL COMMITTEE MEETING

December 7-10, 2005

TODAY'S DATE: December 7, 2005

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	Rick Ennis	NRR/DORL
2	Bob Dennig	NRR/ADR/ACVB
3	CHENG-IH (John) Wu	NRR/DE/EEMB
4	Thomas Scarbrough	NRR/DCI
5	JOHN G. LAMB	ACRS STAFF
6	CORNELIUS HOLDEN	NRR/DORL
7	MARTY STUTZKE	NRR/DRA/APLA
	John Stang	NRR/DROL
9	HAROLD CHERNOFF	NRR/PSCB
10	~~Sarah Hefer~~	
11	QUYNH NGUYEN	NRR/ADRA/DPR
12	P.B. Lyons	OCM/PBL
13	D.H. Cox	OCM
14	Lambros Lois	NRR/DSS/SBWB
15	J. BONGARRA	NRR/DIRS/IOLB
16	Gene Imbro	NRR/DE
17	Hukam Garg	NRR/DE
18	Darrell J. Roberts	NRR/DORL/LPL1-2
	James Tatum	NRR/DSS/SBPB
20		

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
528th FULL COMMITTEE MEETING
December 7-10, 2005

December 7, 2005
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

	NAME	AFFILIATION
1	John Dreyfuss	Entergy-VY
2	LEN GUCWA	ENTERGY
3	MICHAEL J. DICK	GE ENERGY
4	JERALD HEAD	ENTERGY
5	John McCann	ENTERGY
6	PEDRO B. PÉREZ	AREVA
7	Bruce C. Slifer	Entergy - VY
8	MARK E PALIONIS	ENTERGY - VY
9	Jim Callaghan	Entergy - VY
10	PAUL RAINEY	ENTERGY - VY
11	Ed Duda	Entergy - VY
12	CRAIG J. NICHOLS	ENTERGY - VY
13	JOHN M. ODDO	PPL-SUSQUEHANNA
14	Bryan Croke	Entergy VY
15	Louis Quintana	GE Energy
16	DAN YASI	Stone & Webster
17	Michael B. Detamore	PPL Susquehanna
18	VINCENT ANDERSEN	ERIN ENGINEERING
19	Enrico Betti	Entergy VY
20	Brian Hobbs	Entergy - VY

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
528th FULL COMMITTEE MEETING
December 7-10, 2005

December 7, 2005
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

	NAME	AFFILIATION
1	Bott. McElhee	VY
2	Norm Rademacher	Entergy VY
3	DAVID LANGLEY	TVA
4	J D Wolcott	TVA
5	DAN PAPPONE	GE
6	JAY THAYER	ENTERGY VY
7	Samir Ziada	McMaster Univ.
8	Steve Hambric	Penn State University
9	Sarah Hofmann	VT Dept of Public Service
10	William Sherman	VT Dept. of Public Service
11	Mary Frances Repko	U.S. Senate Environment Committee
12	Chris Boyd	USNRC
13	L J Atherton	public
14	Raymond Shadis	New England Coalition
15		
16		
17		
18		
19		
20		

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
528th FULL COMMITTEE MEETING

December 7-10, 2005

TODAY'S DATE: December 8, 2005

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION
1	Geary S. Mizuno	OGC / Rulemaking & Fuel Cycle
2	Roy Woods	RES / DRAA / PRAB
3	N. K. BAGCHI	NRR / ~~DRIP~~ DRA / PRAB
4	BRIAN RICHTER	NRR/DRA/PRAB
5	John Monninger	NRC/RES/DRAA/PRAB
6	Alex Klein	NRR/DRA/AFPB
7	QUYNH NGUYEN	NRR/ADRA/DPR/PGCB
	Daniel Frumkin	NRR/DRA/AFPB
9	Ray Gallucci	NRR/AFPB
10	Phil QUALLS	NRC
11	RICHARD BARRETT	RES
12	Joe Birmingham	NRC
13	Eileen McKenna	NRC
14	Charles Ader	NRC
15	Jim Lyons	NRR/DRA
16	N. P. KADAMBI	RES/DSARE
17	Donnie Harrison	NRR/DRA/APLA
18	Chris Grimes	NRR/DPR
	CORNELIUS Holden	NRR/DORL
20	Rick Ennis	NRR/DORL

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
528th FULL COMMITTEE MEETING

December 7-10, 2005

TODAY'S DATE: December 8, 2005

NRC STAFF - PLEASE SIGN BELOW

PLEASE PRINT (CLEARLY)

	NAME	NRC ORGANIZATION	
1	KAZIMIERAS CAMPE	NRR/DRA	415-2858
2	Mark Blumberg	" "	415-1083
3	RAJ ANAND	"	415-1146
4	NITIN PATEL	NRR/DRA	415-3201
5	Jay Lee	NRR/DRA	415-1080
6	Brad Harvey	NRR/DRA	415-4118
7	Laura A Dudes	NRR/DNRL	415-0146
	WILLIAM BECKNER	NRR/DNRL	415-1126
9	Goutam Bagchi	NRR/DE	415-3305
10	Yong Li	NRR	415 4144
11	Thomas Cheng	NRR	415-2770
12	Paul Prescott	NRR/DE/EQVA	415-3022
13	Margie Kotzalas	NRR/DRA/AADB	415-2737
14	Jose Calvo	NRR/DRNL	415-3257
15	Robert Weisman	OGC/RP	415-1696
16	Jennifer Dixon-Herrity	EDO/TRPS	415-1733
17	JOHN G. LAMB	ACRS STAFF	
18	Angie Lavretta	NRR/DRA	415-3285
	Mark Salley	RES	X2840
20	Sunil Weerakkody	NRR/DRA	X2870

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
528th FULL COMMITTEE MEETING
December 7-10, 2005

December 8, 2005
Today's Date

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14	Benton Halperin	CQ
15	Bruce Slifer	Entergy VY
16	Dan Yasi	Stone & Webster
17	Brian Hobbs	Entergy VY
18	Craig J. Nichols	Entergy VY
19	Len Gucwa	Entergy VY
20	Paul Rainey	Entergy VY

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
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	NAME	AFFILIATION
1	Sarah Hofmann	Vt Dept. of Public Service
2	William Sherman	Vt Dept of Public Service
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
528th FULL COMMITTEE MEETING

December 7-10, 2005

TODAY'S DATE: December 9, 2005

NRC STAFF - PLEASE SIGN BELOW

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	NAME	NRC ORGANIZATION
1	Michael Johnson	OE
2	J Persensky	RES
3	Valerie Barnes	RES
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6	Tina Ghosh	NMSS
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10	Marty Stutzke	NRR/DRA/APLA
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
528th FULL COMMITTEE MEETING
December 7-10, 2005

December 9, 2005
Today's Date

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3	Sarah Hofmann	VT Department of Public Service
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
528th FULL COMMITTEE MEETING
December 7-10, 2005

December ~~9~~ 10, 2005
Today's Date

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ITEMS OF INTEREST

528th ACRS MEETING

DECEMBER 7-10, 2005

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
528th MEETING
December 7-10, 2005

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

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No. S-05-018

"Public Confidence and the Nuclear Regulatory Commission"

Prepared Remarks by
The Honorable Gregory B. Jaczko
Commissioner
U.S. Nuclear Regulatory Commission
before the
Nuclear Power and Global Warming Symposium
Warrenton, VA
November 8, 2005

INTRODUCTION

I am glad to be here today.

I know that the subject of this conference is Nuclear Power and Global Warming. You have been engaged in discussions about whether the expansion of nuclear power offers a safe and viable alternative to the effects the burning of fossil fuels have on the environment.

While these are important issues, it is not appropriate for me, in my job as an independent regulator, to discuss the proper role of nuclear power. Decisions about contracting or expanding nuclear power are for the public to make through the actions of the Administration, the Congress, and ultimately the private sector.

The role of the Nuclear Regulatory Commission in my view is not to promote or discourage this initiative but rather to ensure that any new plant that may get built will be safe and secure. The mission of the NRC is to "license and regulate the Nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of public health and safety, promote the common defense and security, and protect the environment."

The most important requirement for the NRC to accomplish that mission is to ensure public confidence in what we do. The public demands it, the industry needs it, and it is our job.

I am afraid that there is a lot more work to do in this area. For example, I am often asked by members of the public if nuclear power plants are safe. This question illustrates my point. The fact

that there is concern about the safety of nuclear power plants shows the NRC still has a long way to go to convince the public that it is an effective independent regulatory body that can be trusted to ensure the safe use of nuclear materials.
Working to improve public confidence is something we can all agree needs to be done. It is even more important today as we work to maintain effective regulatory oversight of the current fleet of 103 operating nuclear reactors in 33 States — as well as thousands of radioactive materials licensees throughout the country — while preparing to review applications for new nuclear power plants for the first time in decades.

I can confirm for you that the talk about a potential "nuclear Renaissance" is serious and real. Currently, it appears likely that over the next three years the NRC will receive applications from numerous utilities and consortiums to construct new nuclear reactors. In this environment I believe that the only way for the agency to proceed is to ensure that NRC staff are **wedded to safety – not shackled to schedules**. Doing this will require additional resources and a strong commitment to safety culture.

RESOURCES

Let me begin with the issue of resources, which could shackle the NRC staff if not managed correctly.

The industry should expect an efficient and effective NRC process for reviewing applications for new plants. This will require the hiring and training of hundreds of new NRC staff and additional resources for the NRC to develop guidance on enhanced margins of safety utilizing innovative measures and new policy on incorporating security into new reactor designs.

I have encouraged the Commission to work with the Congress to secure additional resources needed to achieve these goals and to ensure that reviewing new applications will not negatively affect on-going safety work.

PUBLIC'S ROLE

The public must also play its critical role in developing sound government policy as new licenses are considered. The NRC is made up of dedicated civil servants who come to work every day wanting to make the right safety decisions, and they need to hear from the public to help them do their jobs.

Of course, that dialogue can be productive only if the NRC is open and transparent in every step of the process. The NRC must be open with information and transparent in the processes we use to make decisions. In a post-September 11th world, we can not always fully achieve our goal of openness, but we can always be transparent as an agency – both to the public and to the licensees. In other words, while specific pieces of information may need to be protected for the NRC to accomplish its public safety and security mission, the *process* the Commission uses to make policy decisions should always be open, accessible, and well understood by all.

For the NRC to do its job, our stakeholders must see an unbiased agency whose primary goal is ensuring the safe use of nuclear materials.

LICENSEES' ROLE

The industry can also help improve public confidence and avoid the shackles of schedules. Any applications licensees submit for new reactors must be thorough and high-quality. The burden is on the industry to convincingly address all of the necessary safety and security issues.

The NRC should be clear and firm about its standards and must not be afraid to reject applications that do not meet them. Prematurely accepting inadequate applications will only create scheduling pressures on the NRC staff.

Only with the necessary resources – and through consistent responsible actions on the part of the NRC staff, the industry, and the public – can we be certain to break the shackles of arbitrary schedules and ensure we are ensconced in a happy marriage with safety.

SAFETY CULTURE

Beyond resources, there is another issue that we must focus on to ensure that there are no shackles on NRC staff and the industry. We must show the public that we value a questioning attitude. We must reinforce a culture at the agency and in the industry in which everyone feels empowered, emboldened and encouraged to ask the next question, the difficult question, and not to simply accept what is presented to them.

If public confidence is the key to effectively regulating the nuclear industry, the foundation is achieving an environment focused on safety and security – a concept known as safety culture. The NRC considers "safety culture" to involve a work environment where management and employees are dedicated to asking questions and promoting safety.

Safety culture at the NRC is like a pot beginning to boil. You are familiar with the proverbial "watched pot" just when you begin to see individual bubbles forming. Those first bubbles are like the divergent views at the NRC. Unfortunately, in my view, the NRC has a tendency to take the pot off of the stove before it reaches a full boil. I would like to see a raging boil of divergent views reach its way directly to the Commission to ensure we have access to all of the information we need.

If we look at the history of the nuclear industry, we find that problems almost inevitably appear as a result of a loss of this questioning attitude, a deteriorating safety and security culture. One of the biggest challenges in this arena is complacency, and unfortunately, complacency is most likely to be recognized only after it seeps in and contributes to a degraded safety and security environment.

DAVIS-BESSIE

The most recent and well-investigated example of this can unfortunately be found at the Davis-Besse Nuclear Power Station in Ohio.

On March 5, 2002, the licensee for Davis-Besse discovered cracks and corrosion in the reactor pressure vessel head, which is the top of the reactor coolant system pressure boundary. During repair of the identified cracks, a cavity the size of a football was discovered that extended completely through the 6-inch thick carbon steel cap all the way down to a thin stainless steel liner.

Even after years of operating experience and armed with the information about a potential problem that the NRC provided, the industry as a whole failed to implement an effective corrective action program to identify and manage this type of cracking and corrosion. The licensee failed to effectively implement its operating experience review program and catch this corrosion before it became a serious safety issue. The NRC failed to ensure that the safety issue was identified and corrected even though it knew about generic problems with this important component of a plant.

As a result, the NRC instituted a Davis-Besse Lessons Learned Task Force and recommendations from this task force have been implemented. But our work is far from over. This event did not occur decades in the past at the infancy of this industry and the NRC, but rather only a few years ago with a mature regulator and a mature industry relying on a record of safety that led to complacency.

The Davis-Besse incident is a clear example of why the public lacks confidence in the industry and why the questioning attitude at the heart of safety culture is essential for continued nuclear reactor safety. Employees - both of the NRC and the industry - must feel empowered to ask the difficult questions. Ensuring this happens is at the core of safety culture.

EMERGENCY PREPAREDNESS

I want to wrap up my talk with an important topic that I believe serves as a barometer for how we can measure the public confidence in the NRC – emergency preparedness. After all, the emergency planning effort is the most tangible way the nuclear industry affects its neighbors.

When I travel to nuclear power plants I always try to meet with local elected officials and citizen groups. One of the most frequent issues I hear from these stakeholders is concern about the emergency preparedness plans in the 10-mile zones around the plants.

It is the NRC's responsibility to evaluate a licensee's onsite emergency plan and the agency relies on the Federal Emergency Management Agency – FEMA – to provide recommendations about the adequacy of State and local emergency plans. This system makes sense because FEMA is the agency with the emergency management expertise and the relationships with state and local governments to address all hazards.

I do believe, however, that the NRC should take prompt action to eliminate any doubts or concerns about *radiological* emergency plans. Input from FEMA is crucial but the NRC has the ultimate authority and responsibility to ensure the adequate protection of public health and safety around nuclear power plants. The Commission and the public should not be left to wonder if alert and notification procedures are in place, transportation resources are available, and reception and care centers are arranged.

I want to be able to visit any of the 65 nuclear power plant sites in this country and hear – not only from the licensees, but also from the public – that there is complete confidence in the emergency plans in place. No other outcome will more clearly demonstrate to the public that the NRC is wedded to safety and committed to improving public confidence .

CONCLUSION

As I conclude my remarks I hope to have helped frame and clarify some of the issues you are pursuing here this week.

The NRC must work to improve the confidence of the public in its capabilities and intentions to effectively regulate the nuclear industry in whatever shape it takes in the future. We can all agree that our goals should be a safe and secure future in which the health of our families and communities is guaranteed and our environment is protected. Working together – industry, the public, and the NRC – is the best way to avoid the arbitrary shackles of schedules and ensure the industry and the NRC staff remain wedded the imperative of safety.

Again, I thank you for the invitation to speak to you today, I commend for your efforts to learn more about and report on these important issues, and I look forward to any questions you may have.

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

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No. S-05-017

Comments of Commissioner Jeffrey S. Merrifield
Saxton Nuclear Power Plant
November 8, 2005

- Good Morning. On behalf of the United States Nuclear Regulatory Commission (NRC), it is indeed, a pleasure to be here today to participate in the celebration of the completion of the decommissioning of Saxton Nuclear Power Plant.

- Nearly 44 years ago, on November 15, 1961, to be precise, our predecessor, the Atomic Energy Commission (AEC), issued a license to the Saxton Nuclear Experimental Corporation to operate an experimental pressurized water reactor.

- The reactor, which first went critical on April 12, 1962, was not built primarily to generate electricity. Instead, it was intended as a research and developmental program to demonstrate how a nuclear reactor could be operated under utility operating conditions.

- AEC documents dating to the early 60's describe the Saxton effort as "'generating knowledge' about getting more heat, and hence more electricity, out of nuclear fuel and thereby reducing the future costs of power generation."

- While Saxton only generated power at 23.5 thermal megawatts, a mere fraction of a modern nuclear power plant, it laid the foundation for understanding how "better and more powerful reactors" could be built in the future.

- Saxton served as a pioneer in the nuclear industry through its use of boron in cooling water to control the chain reaction and it was the first privately owned reactor to use plutonium as fuel.

- Operating more than 11 years until it shut down in May of 1972, Saxton was distinguished by the fact that it operated with neither fanfare nor serious incident. Decommissioned at a time when our current 104 reactor fleet was in its boom years, Saxton quickly faded in the memory of the AEC.

- As I was preparing to come here today, I was struck by two facts. The first is, that as far as we can tell, I am one of the first, if not the first Commissioner of either the AEC or the NRC to have visited this site.

- Part of that reason results from tradition. As a matter of practice, Commissioners typically did not attend reactor groundbreaking or commissioning ceremonies, because of a concern that this would be perceived as an endorsement of promotion of nuclear power.

- Beginning in the mid-1980's it became more of a habit for NRC Commissioners to visit operating nuclear plants to oversee their safe operations, but obviously, by that time, Saxton was long shut down.

- The second fact that struck me is that counter to my intuition, at the time Saxton was first conceived and built, virtually no consideration was given as to what to do with the reactor site when power operations were completed.

- In our society today, it would be inconceivable to think that a nuclear power plant could be licensed and built with virtually no consideration about what to do with the radioactively contaminated building after its useful life was complete. Yet that is precisely what happened in 1961.

- As a side note, Saxton did not even have to face one of the most difficult issues confronting many other reactors that have gone through decommissioning. The fuel used at Saxton was owned by the federal government and consequently was returned to the Savannah River site in South Carolina when the reactor ceased operations in 1972. Other reactors have not been so lucky in resolving the issue of where spent fuel will be sent to complete the decommissioning process.

- At the time of Saxton's shut down in 1972, the AEC was only in the very early stages of deciding what to do with these decommissioned reactors.

- In 1977, in testimony before the House Committee on Science and Technology, the NRC stated that virtually all of the 11 test reactors that had closed by that time had chosen mothballing as the alternative for decommissioning. Referencing Saxton in particular, the NRC expected that the "the residual radioactivity may be removed after about 50 years" – or about the year 2027.

- It was not until 1988, after more than 11 years of effort, that the NRC issued a final rule that required utilities to specify how they would assure that adequate funding was available to clean up a site after a plant ceased operation. In addition, this rule required them to outline how they would conduct the decommissioning, how long it would take, and how they would protect public health and safety in the process.

- For the first time, the Commission, following the lead of communities like this one, began to ask for a more robust explanation of how decommissioning would result in the unrestricted use of the site and a greater justification for choosing options such as entombment or mothballing for decommissioning sites instead of returning the site to its original condition.

- Today, if you look upon the field where a power reactor used to sit, it is hard to believe that our predecessors could have been so short sighted. While the promises of nuclear power certainly gleamed in the eyes of many Americans, it is unfortunate that it took so long for the final pages in the history of Saxton to be turned.

- Yet today is a day of celebration. Through the dedication of many local residents who participated as members of the Saxton Citizens Task Force, attended one of the many meetings held regarding decommissioning, or cheered on those who did, this effort resulted from significant community involvement and planning.

- Likewise, General Public Utilities (GPU), which built and operated the reactor, and which is now represented by First Energy, took the responsibility for the decommissioning of this site, at a cost many times in excess of the cost to build it in the first place.

- This is also an important event for my Agency, the NRC, for it represents the fulfillment of our obligation to license nuclear facilities in a manner that protects public heath, safety and the environment. Today, unlike our predecessor the AEC, environmental stewardship is a much more important element of our mission.

- Like its pioneering days of the early 1960's, Saxton is also one of the pioneers in a new effort: providing for a decommissioning that allows for productive reuse of the site by the local community. This site can be used safely for any number of activities, which is a goal we would like to achieve for every decommissioned site.

- In our nation today, we are on the precipice of a number of utilities considering the decision of whether or not to build new nuclear reactors in the United States. After a long dormancy, as many as 6-8 utilities may seek combined operating license applications with the NRC in the next few years.

- As Saxton helped to create the conditions for the operation of large nuclear reactors, the efforts of this community, this utility, and our Agency, which resulted in the decommissioning of Saxton, have also set a new stage for nuclear power. While many questions may be asked about the cost or need to build a nuclear reactor, Saxton has answered the question as to whether reactors can be fully dismantled after they fulfill their useful life. Communities all across America will benefit from the hard-fought lessons learned here in Saxton.

- Again, I want to thank you for allowing me to join you today.

NEW INFRASTRUCTURE FOR

NEW NUCLEAR POWER PLANTS

Chairman Nils J. Diaz

U.S. Nuclear Regulatory Commission

at the

26th Annual INPO CEO Conference

November 2, 2005

Atlanta, Georgia

[TITLE SLIDE] Good afternoon. It is a pleasure to be here and have the opportunity to address you again at INPO's CEO Conference. I would like to thank INPO for their kind invitation and their hospitality. Also participating at this meeting are NRC Commissioners Merrifield, Jaczko, and Lyons, as well as NRC's EDO, Luis Reyes and Regional Administrators Bill Travers and Jim Caldwell. You can direct any difficult questions to them.

It has been my privilege to attend this conference for 10 consecutive years, not a record but a good run. During that time, I have witnessed the positive transformation in the performance of the NRC and the nuclear industry. These performances, and specifically the licensees' interrelated safety and reliability performance, have been instrumental in establishing the necessary platform for increased public acceptance of nuclear energy, and for the support expressed for it by the President and the Congress of the United States. [SLIDES 2 and 3]

At the potential cost of showing my years – of experience, that is – I would like to state that the NRC is a stronger and better agency, that I believe the industry is also stronger and better, and our nation continues to get stronger and better. Challenged, yes, but stronger. The nuclear industry and the NRC have key roles to play in ensuring the continuing strength and improvement in the nation's energy security and well being. The roles are changing, and changing fast; moving from a perennial defensive position to a front line opportunity to maintain and enhance energy security, and indeed, the security and well-being of the nation. We should welcome the new roles and the opportunity, but, as the saying goes, be careful what you ask for .... With the passage of the Energy Policy Act of 2005 and other key factors in place, it appears that the game plan for potential new reactor licensing applications is being set. The next two slides summarize some factors influencing the prospects of constructing nuclear power plants:

[SLIDE 4] Factors positively influencing the prospects of constructing new nuclear power plants:

Support by the President and the Congress for expanding the use of nuclear power, including incentives for the first six plants
Concerns with the Nation's energy security
High cost of oil and natural gas
Environmental considerations
Low and stable electrical production costs from nuclear
Low interest rates and inflation
Renewed interest by utilities in building new nuclear power plants
NRC's establishment of an improved licensing process

[SLIDE 5] Factors with potential negative influence on the prospects of constructing new nuclear power plants:

High capital cost of new nuclear power plants
Financing considerations
New licensing processes have not yet been fully tested

[SLIDE 6] The potential is large and the tasks are sobering. There is a need for a new infrastructure for new nuclear power plants that includes:

Improved environmental assessments
Improved techno-legal framework
Improved reactor design and construction

Reliable suppliers
Well-qualified personnel

I am going to focus my comments this afternoon on the licensing process and the needed infrastructure, and the connectivity between them, with emphasis on the NRC role. Before I go any further, I want to clearly set where the NRC is today, regarding its key role, in the nation's energy security. In 1997, the NRC's Strategic Plan stated – [SLIDE 7]

> NRC's mission is to regulate the Nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of the public health and safety, to promote the common defense and security, and to protect the environment.

In 2004, the NRC's Strategic Plan stated that the NRC's mission is to — [SLIDE 8] License and regulate the Nation's civilian use .... to ensure adequate protection of public health and safety, promote the common defense and security, and protect the environment.

The addition of the word "license" derives directly from the NRC's statutory responsibility to review and decide, in a timely fashion, applications for licenses, and to grant the applications if the requisite health, safety, security, and environmental standards are met, and to deny them if they are not. Furthermore, NRC's Strategic Plan for 2004 – 2009 includes one Strategic Objective [SLIDE 9]:

> Enable the use and management of radioactive materials and nuclear fuels for beneficial civilian purposes in a manner that protects public health and safety and the environment, promotes the security of our nation, and provides for regulatory actions that are open, effective, efficient, realistic, and timely.

This is a profound and challenging objective that reflects the NRC's statutory responsibilities. On a personal note, it represents most of what I have tried to achieve during my tenure as Commissioner and then Chairman of the United States Nuclear Regulatory Commission.

I could talk for a while about the meaning of the word "enable;" it is a catchy and very complex word. In fact, I recently have been spending a significant amount of time discussing its full meaning with the senior managers of the agency, in the context of the Strategic Objective. Instead, I am going to turn the tables on the nuclear industry and show one example of how you have enabled and should continue to enable the use of nuclear energy for beneficial civilian purposes, in a manner that protects public health and safety, common defense and security, and the environment.

Last year, I talked to you about "Excellence in Safety Management," and in particular about Unplanned Reactor Shutdowns of more than 6 months duration [Slide 10]. Congratulations, you have managed well the country's nuclear fleet. The average for extended plant shutdowns of over six months has decreased from about six per year for the period of 1979 to 1997 to about one per year since then. I have frequently stressed the contribution of safety to reliability, and vice versa. I believe the improved performance has also resulted in enhanced safety as well as reliability. I am sure that major, additional impacts can be surmised from every period of performance shown on this figure.

As demonstrated by the unplanned extended shutdown issue, the challenge faced by the nuclear industry and by the NRC in the upcoming years is far broader and deeper than new reactor licencing and its infrastructure. A necessary enabler of new reactor licensing is continuing safety and reliability of operating nuclear power plants. Excellence in operations is an enabler, and I appreciate INPO's contributions in this regard. Excellence in safety, security, and preparedness is an enabler. Excellence in applications for license renewals, power uprates, and license amendments is an enabler. Better, wider use of risk-informed and performance-based PRAs for design, operation, and maintenance, is an enabler. We should enable the capabilities of the new generation of managers and practitioners to effectively perform their duties. And, last but not least, to enable the licensing of new reactors, the industry needs to submit for the docket, high quality, complete, indeed excellent, thoughtfully assembled applications that clearly conform to the regulations and are fully supported by the vendor, architect/engineer, constructor, and supplier (i.e., the complete infrastructure). The NRC will then be able to do its job, in accordance with the quoted strategic objective. All these and more need to be done well. There is an old saying, often overused but dramatic, that may fit the present situation: "Failure is not an option."

The potential deployment of new nuclear power plants comes after a long hiatus in nuclear power plant licensing and construction. The lack of predictable financing, electric demand, design, construction, and regulation resulted in the long delays or cancellations of the 70's and 80's. There should be at least one good result from that experience: everyone today should be better prepared. There is no forgiveness in this business; expect none, for you will get none. However, there are rewards for anticipation, for preparation, and for simplicity. Simplicity is the mother of wisdom and the grandmother of achievement.

The entire nuclear business is different and still changing and no one should underestimate the difficulty in successfully engaging it in the construction mode. However, everyone today knows better, and should be able to execute better. Yet, it has been difficult to establish where are the horses in relationship to the carriage. Just a few years ago, the vendors were ahead of the utilities, with three banked standard certified designs. There were no buyers. There now appears to be a significant expression of interest from utilities, specifically in three reactor designs that are not yet completed for use in a COL [SLIDE 11]. A regulatory note: the governing technical rule for standard certified designs is 10 CFR 52.47. It requires that an evolutionary design, like the EPR, provide an essentially complete design prior to certification. For reactor designs with simplified, inherent passive or other means to accomplish their safety functions, like the AP 1000 and the ESBWR, the scope of the design must be complete and the applicant must demonstrate the performance of safety functions. [SLIDE 12] Either way, at COL application time, it will be highly desirable that you submit the complete safety case, ready for rigorous review, and ultimately for hearing. A standard certified design, with a complete rulemaking, has definite advantages. On the other hand, there is much good to be said about an essentially complete design. There is also much good to be said about a COL application supported by an adjudicated Early Site Permit and a Standard Certified Design by rulemaking. I fully understand that Part 52 permits different schemes for a COL, yet there is something to be said for simplicity. I believe that – [SLIDE 13]

Expectations and Permutations
are often not a good
Combination

That having been said, assuming the submission of a top-notch COL application, with an approved site and certified design, that clearly meets all regulatory requirements, the current estimated time from application to a decision on a COL, including adjudication, is about three years. [SLIDE 14] Assuming that the inspections, tests, analyses, and acceptance criteria (ITAAC) specified in the COL are satisfied during and on completion of construction, the time for construction, ITAACs, fuel loading, and initial operations is currently estimated to be about five years, for a total time estimate of eight years from COL application to initial operation.

I proposed that the need to continue doing all we are used to doing well is now challenged by having to do all we are going to have to do very soon, very well. I believe the NRC needs to, and you need to, recognize that –

- the challenge to be faced is likely to be more complicated than originally presumed;
- there is a need to stay ahead of the curve to meet external and internal expectations;
- the ability to recognize problems and the flexibility to address them promptly as they occur is essential; and,
- available resources must be used wisely and new resources must be sought when needed.

You will notice that all four of the conclusions imply that a successful outcome to a major challenge requires careful and complete preparation, the ability to perceive the challenge from a complete perspective, and the ability to bring the needed resources to bear on resolution in a timely manner.

Let me summarize for you some of my own views and concerns:

- I envision the next few years as posing the most serious challenge the NRC and the industry have faced in a generation, a "rising tide" of new responsibilities and difficult decisions that cover a wide spectrum of activities. Not the least of these is the potential necessity to resume new licensing activities after a long hiatus, with a new set of rules, players, reactors, construction methods, infrastructure, and high national demands.

- The NRC and the industry are <u>more</u> experienced with adjusting to downsizing and tight budgets than we are with expanding projects and resources. The preparation necessary to do more with less is quite different than preparing to do much more <u>with</u> more.

- Success in this context requires increased attention and sensitivity to external expectations, which are and will remain extremely high and extend across the board to the public, industry, the Congress, and the world – everyone will be watching and judging our actions. NRC will expect high quality and that schedules be followed. Industry will expect that NRC timetables will be met. The public will demand enhanced safety and must receive broad access to the decision-making process. Congress will expect all of this, and accountability.

- The NRC and the industry have made and will continue to make a number of necessary changes to address key issues, including the integration of existing reactors' safety, security, and emergency preparedness, and to address early the safety/security interface for new reactors. Everyone must ensure that these new frameworks will work as they should in the new, changed environment and must be willing to made additional changes should they prove necessary.

- The NRC is not a technical agency. It is a techno-legal agency. It must realize, internalize, and act according to that fact. The nuclear industry must also pay close attention to the techno-legal interface. The techno-legal interface is a key to doing the job right; it has to be transparent, and yes, managed.

- The NRC and the industry will face the new challenges that lie ahead with the largest increment of new staff and new managers that has been required in some time.
- Effective communications, internal and external, is being raised to a new level; everyone is affected.

- Everyone must live up to the standards that are required.

Someone told me the other day: the train is leaving the station. [SLIDE 15] I asked of myself and the NRC staff, and maybe you should ask yourself:

If the train is leaving the station,
Am I on it?
Do I know where it is going?
Do I know what to do once I am on the train?
Do I know if everybody on the train knows what they should do?
Do I have the plan, the tools, the resources I will need to get it to its destination?
Do I and everyone else know what to expect during the trip?
Do I know what to do when it gets to its final destination?

The Nuclear Regulatory Commission realizes that we have to anticipate, define, prepare, and execute at a different level. We are going to keep doing well all the things we do well, day in and day out, but we are going to be better at managing the new and the different, the one or two of a kind, the first, and then the second, and so on, and so on. We are going to pay special attention to the techno-legal interface, internally and externally, because they have to march forward together. I cannot emphasize enough the importance that this interface has for everyone that is going to be involved in new nuclear power plant construction.

At the beginning of my talk, I mentioned the Energy Policy Act, the NRC mission, and the new Strategic Objective, which is indeed a governing objective. I am going to articulate a high and viable goal for the NRC that captures all of the above in one measurable outcome:

Use of the Energy Policy Act's risk insurance program should not be the consequence of NRC fault in the thoroughness and timeliness of its review.

I am sure you realize I am placing the burden on you. If you come to the NRC with a COL application, it should have one requisite ultimate quality: the safety case and other required components are of such excellence that the application can pass the tests of staff review, NRC hearing, and courts of law. Anything less lacks the predictability the nation expects and many demand.

On a final note, I believe you should know that I am an optimist, and that I am very encouraged by the seriousness of the energy debate and the solutions that our country is adopting and considering to secure our energy future. The fact that my many scars are reacting to the weather front should be food for thought, and, I hope, action. I am convinced that the NRC and the nuclear industry have the capability to respond to the challenges ahead.

Thank you.

November 9, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - COMSECY-05-0047 - SEMIANNUAL REPORT - STATUS OF IMPLEMENTATION OF DAVIS-BESSE LESSONS LEARNED TASK FORCE REPORT RECOMMENDATIONS

The Commission has approved the staff's recommendation to discontinue the semiannual report on the status of the implementation of the Davis-Besse Lessons Learned Task Force recommendations.

The staff provided a status update to the Commission on the efforts to develop an NRC lessons learned program in July 2005 and again in the November 1, 2005 Commission briefing. The staff should provide the Commission semi-annual updates on this effort in the form of an information paper or memorandum to the Commission.

The staff should continue to collect operating experience from international sources, with specific consideration given to obtaining information related to advanced reactor designs that are likely to be proposed to be built in the U.S.

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

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS

DOCKETED 11/21/2005

Nils J. Diaz, Chairman
Edward McGaffigan, Jr.
Jeffrey S. Merrifield
Gregory P. Jaczko[1]
Peter B. Lyons

SERVED 11/21/2005

In the Matter of	)	
	)	
U.S. DEPARTMENT OF ENERGY	)	
	)	Docket No. PAPO-00
(High Level Waste Repository: Pre-Application Matters)	)	
	)	

CLI-05-27

MEMORANDUM AND ORDER

The NRC staff and the U.S. Department of Energy ("DOE") have appealed the Pre-License Application Presiding Officer ("PAPO") Board's ruling that certain drafts of DOE's license application for a geologic repository at Yucca Mountain must be made available on the Licensing Support Network ("LSN").[2] The NRC staff, but not DOE, has asked for a stay of the effectiveness of that order pending a final Commission decision on the appeals. We deny the NRC staff's stay application.

I. BACKGROUND: LSN AND PAPO BOARD'S RULING

In June, 2005, the State of Nevada filed a motion to compel DOE to make available on the LSN two particular drafts of its license application. Nevada asked that DOE be ordered to produce the drafts on or before the date DOE "certifies" that it has made its relevant documents

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[1] Commissioner Jaczko has recused himself from this matter, and did not participate in today's decision.

[2] LBP-05-27, 62 NRC ___(Sept. 22, 2005).

electronically available.[3]

Under our regulations governing repository licensing proceedings (10 C.F.R. Part 2, Subpart J), each participant must place on the LSN "all documentary material (including circulated drafts but excluding preliminary drafts)"[4] on which the participant intends to rely or which are relevant. "Documentary material" includes "all reports and studies, prepared by or on behalf of the potential party ... including all related 'circulated drafts.'"[5] A "circulated draft" is a "nonfinal document circulated for supervisory concurrence or signature in which the original author or others in the concurrence process have non-concurred."[6] Neither "concurrence" nor "concurrence process" is defined in the regulations.

Here, DOE's contractor delivered a draft license application in July, 2004. After undergoing an extensive agency review process, a second version was produced in September 2004. Nevada argued that these two documents constitute "circulated drafts" of relevant "documentary material" that must be made available on the LSN at the time DOE certifies its document collection as electronically available. In a lengthy opinion, the PAPO Board agreed with Nevada.[7] The Board ordered Nevada to include the two draft license applications in the LSN "no later than" the time of DOE's initial certification.[8]

The NRC staff has already certified its document collection. DOE's earlier effort to do

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[3] *See* 10 C.F.R. § 2.1009(b) (each potential party must certify to the PAPO that it has made available the documents identified in 10 C.F.R. § 2.1003). DOE must make its certification at least six months prior to submitting its license application. 10 C.F.R. § 2.1003(a).

[4] 10 C.F.R. § 2.1003(a).

[5] 10 C.F.R. § 2.1001 (definition of "documentary material").

[6] *Id.* (definition of "circulated draft").

[7] LBP-05-27. Apparently, a third draft was produced in November 2004, but it is not subject to the Board order. *See id.* at 1 n.1

[8] *See id.* at 52-53.

so proved unsuccessful.[9] Pursuant to a PAPO-issued "Case Management Order," the NRC staff has an ongoing obligation, on a monthly basis, to supplement its already-certified document collection.[10]

II. STAY DENIED

DOE and the NRC staff disagree with several aspects of the Board decision ordering inclusion of the draft DOE license applications in the LSN. They focus on the meaning of the term "circulated draft." DOE argues that the draft applications in question were not "circulated," within the meaning of our rules, because they were not yet undergoing the formal "concurrence" process used by that agency. Instead, DOE says, the drafts were only "preliminary" drafts, explicitly excluded from mandatory inclusion in the LSN. The NRC staff takes no position on whether or not the DOE draft applications are in actuality "circulated drafts,"[11] but expresses concern that the Board's legal interpretation of the term will drastically expand the number of documents all parties must place on the LSN, including the NRC staff.

The NRC staff asks us to stay the effectiveness of the Board's order. The staff argues that it needs relief, pending the outcome of our appellate review of the PAPO Board's order, of any obligation "to review its current process for identifying circulated drafts to determine whether it needs modification to ensure it is in compliance with the PAPO Board's Order."[12] For several reasons, we find issuance of a stay, which the Commission treats as "an extraordinary equitable

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[9] *See U.S. Department of Energy* (High Level Waste Repository), CLI-04-32, 60 NRC 469 (2004).

[10] *U.S. Department of Energy* (High-Level Waste Repository Pre-application Matters), Second Case Management Order (Pre-License Application Phase Document Discovery and Dispute Resolution), at 21-22 (July 8, 2005).

[11] *See* NRC Staff Appeal of LBP-05-27 and Application for a Stay, at 5 (Oct. 3, 2005).

[12] *Id.* at 14.

remedy,"[13] inappropriate and unnecessary.

First, the NRC staff's stay application fails to meet the "most important" of our stay requirements – "irreparable harm."[14] The staff's process-driven concern – namely, that it will have to review its files to consider whether documents meet the PAPO Board's test for "circulated drafts" – does not come close to irreparable harm in a legally meaningful sense. The Commission has held expressly and repeatedly that "litigation expense, even substantial and unrecoupable cost, does not constitute irreparable injury."[15] Having consistently kept to this strict standard with licensees[16] and with intervenors,[17] we cannot relax it for the NRC staff.

Second, even if a document review by the NRC staff somehow could be considered irreparable harm, the staff has not explained why a stay is necessary to avoid that harm. The PAPO Board decision ordering production of the DOE draft license applications does not, as such, require the NRC staff to do anything. The staff apparently feels that an earlier Board

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[13] *Public Service Company of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-77-27, 6 NRC 715, 716 (1977). *Accord Nuclear Fuel Services, Inc.* (Erwin, TN), LBP-04-2, 59 NRC 77, 79-80 (2004) (citing authorities). *See also Cuomo v. NRC*, 772 F.2d 972, 978 (D.C. Cir. 1985).

[14] *Sequoyah Fuels Corp. and General Atomics* (Gore, Oklahoma Site), CLI-94-9, 40 NRC 1, 7 (1994). *Accord Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant), CLI-01-11, 53 NRC 370, 393 (2001). Under our rules, in considering stay applications, we also consider probability of success on the merits, harm to others, and the public interest. *See Sequoyah Fuels* Corp., CLI-94-9, 40 NRC at 6. Our regulations codify these standards. 10 C.F.R. § 2.342(e).

[15] *Metropolitan Edison Co.* (Three Mile Island Nuclear Station, Unit 1), CLI-84-17, 20 NRC 801, 804 (1984). *Accord Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), CLI-02-11, 55 NRC 260, 263 (2002); *Sequoyah Fuels Corp.*, CLI-94-9, 40 NRC at 6-7. *See generally FTC v. Standard Oil Co.*, 449 U.S. 232, 244 (1980).

[16] *See, e.g., Sequoyah Fuels Corp*, CLI-94-9, 40 NRC at 6-8; *Three Mile Island*, CLI-84-17, 20 NRC at 802-805.

[17] *See, e.g., Private Fuel Storage,* CLI-02-11, 55 NRC at 262-64.

order, on "case management,"[18] which requires monthly supplements of certified document collections, compels the staff to reconsider its approach to "circulated drafts" to ensure compliance with the PAPO Board's newly-articulated definition. But there seems no reason why the staff may not simply ask the PAPO Board to modify the case management order, temporarily, to accommodate the staff's resource concerns during the pendency of the current appeals. Or, as Nevada suggests, the NRC staff seemingly has the option to withdraw its certification, for the time being, while the Commission reviews the current appeals.[19] This would eliminate any need to file supplements. With self-help options available, there appears no reason for the Commission to stay this Board order not even directed to the staff.

Finally, although we have suggested a willingness to consider issuing a stay, despite an absence of real harm, where the chance of reversal on appeal is "overwhelming" or a "virtual certainty,"[20] not even the NRC staff suggests this is such a case. The staff says merely that "the proper interpretation of the term 'circulated draft' is complex and a stay should be granted to maintain the *status quo*."[21] We agree with the staff that the issues here are complex – the Board wrote a 53-page opinion, and the parties (and a prospective *amicus curiae*) have

[18] *See* note 10, *supra*.

[19] *See* State of Nevada's Response to NRC Staff's Appeal, etc., at 10 (Oct. 13, 2005). The deadline for the NRC staff's certification is 30 days after DOE's certification – which has not yet taken place. *See* 10 C.F.R. §§ 2.1003(a), 2.1009(b).

[20] *See Sequoyah Fuels Corp.*, CLI-94-9, 40 NRC at 7.

[21] NRC Staff Appeal of LBP-05-27 and Application for a Stay, at 14 (Oct. 3, 2005). The Staff cited to *Washington Metropolitan Area Transit Commission v. Holiday Tours, Inc.*, 559 F.2d 841, 844-45 (D.C. Cir. 1977) for the proposition that a tribunal may issue a stay when there is a "difficult legal question and the equities in the case suggest that the status quo be maintained." In the instant case irreparable injury has not been demonstrated so as to raise substantial equity concerns. As the *Holiday Tours* Court stated: "An order maintaining the status quo is appropriate when a serious legal question is presented, when little if any harm will befall other interested persons or the public and *when denial of the order would inflict irreparable injury on the movant.*" *Id.* at 844 (emphasis added).

provided us well more than a hundred pages of appellate briefs. But significant and difficult issues do not justify a stay in and of themselves, absent a strong showing of irreparable harm.[22] Where, as here, there is no irreparable harm whatever, the Commission is reluctant to rush to judgment on the merits of an appeal before it has the opportunity to examine the Board's ruling and the parties' arguments in detail.

III. CONCLUSION

For the foregoing reasons, the NRC staff's application for a stay pending appeal is *denied.*

IT IS SO ORDERED.

For the Commission

/RA/

Annette L. Vietti-Cook
Secretary of the Commission

Dated at Rockville, Maryland
this 21st day of November, 2005

[22] *See Sequoyah Fuels Corp.*, CLI-94-4, 40 NRC at 8.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS
WASHINGTON, DC 20555-0001

November 22, 2005

NRC REGULATORY ISSUE SUMMARY 2005-28 SCOPE OF FOR-CAUSE FITNESS-FOR-DUTY TESTING REQUIRED BY 10 CFR 26.24(a)(3)

ADDRESSEES

All licensees authorized to operate a nuclear power reactor, possess or use formula quantities of strategic special nuclear material, or transport formula quantities of strategic special nuclear material.

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to convey the NRC's staff position on the circumstances under which for-cause fitness-for-duty (FFD) testing is required after an accident and to provide the basis for enforcement guidance on the type and severity of personal injury accidents for which this testing must be performed. The staff's current position was established through an enforcement action taken against a nuclear power plant licensee. The information in this RIS is intended for licensees who are required to establish and maintain FFD programs and procedures as specified in Title 10 of the *Code of Federal Regulations*, Part 26 (10 CFR Part 26). No specific action or written response is required.

BACKGROUND

In June 1989, the NRC amended its regulations to incorporate into 10 CFR Part 26, a requirement that licensees establish and implement FFD testing programs at certain types of licensed facilities (54 FR 24494). As part of the final rulemaking process, the agency issued NUREG-1354 (May 1989), "Fitness for Duty in the Nuclear Power Industry: Responses to Public Comments," which addressed comments and questions about the proposed rule. Although the information in NUREG-1354 did not constitute an official interpretation of the regulation as specified in 10 CFR 26.4, licensees used the report in developing and implementing their FFD testing programs and procedures required by 10 CFR Part 26.

During NRC inspections of licensees' FFD testing programs, questions have arisen concerning the meaning of 10 CFR 26.24(a)(3), which requires for-cause testing (1) as soon as possible after any observed behavior indicating possible substance abuse, (2) after any accident involving a failure in individual performance (human error) that results in personal injury, radiation exposure, or release of radioactivity in excess of regulatory limits or actual or potential

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substantial degradation of the level of safety of the plant if there is reasonable suspicion that a worker's behavior contributed to the event, or (3) after receiving credible information that an individual is abusing drugs or alcohol. The questions about 10 CFR 26.24(a)(3) concern the requirements in the second clause for for-cause testing after an accident involving a failure in individual performance resulting in personal injury. Additional questions have arisen about apparent inconsistencies in the NUREG-1354 guidance and the lack of guidance on the type and severity of injuries that require for-cause testing.

SUMMARY OF ISSUE

10 CFR 26.24(a)(3) sets forth the requirements that licensees must implement for for-cause testing. The subsection has three clauses separated by semicolons. The first and third clauses ("as soon as possible following any observed behavior indicating possible substance abuse" and "after receiving credible information that an individual is abusing drugs or alcohol") do not directly relate to a worker's involvement in an accident and appear to be adequately addressed by licensees' FFD testing programs. The second clause of 10 CFR 26.24(a)(3) specifies three conditions for for-cause testing after an accident involving a failure in individual performance. The conditions are separated by commas. In the rest of this RIS, the event-related conditions will be referred to by lower case Roman numerals. Criterion (i) is that the accident involves a failure in individual performance resulting in personal injury, Criterion (ii) is that the accident involves a radiation exposure or release of radioactivity in excess of regulatory limits, and Criterion (iii) that the accident involves an actual or potential substantial degradation of the level of safety of the plant and there is reasonable suspicion that a worker's behavior contributed to the event.

NRC inspections of licensees' FFD testing programs have revealed inconsistencies in the circumstances under which licensees perform for-cause testing under the second condition of 10 CFR 26.24(a)(3). The inconsistencies result from different interpretations of the phrase "if there is reasonable suspicion that a worker's behavior contributed to the event" in Criterion (iii) and the applicability of this phrase to Criterion (i) above. The NRC's position is that the phrase "if there is reasonable suspicion that the worker's behavior contributed to the event" applies only to the phrase to which it is grammatically connected; that is, Criterion (iii), "actual or potential substantial degradation of the level of safety of the plant," and it does not apply to Criterion (i) and (ii).

NRC inspections have revealed that some licensees interpret "reasonable suspicion" in Criterion (iii) to mean "reasonable suspicion of drug use" or some other type of observed behavior that indicates possible substance abuse. This interpretation of "reasonable suspicion" is incorrect. Rather, if a worker is involved in an accident (event) involving a failure in individual performance resulting in any of the conditions specified in Criteria (i) and (ii), the worker is subject to for-cause testing regardless of the worker's observed behavior or any licensee suspicion of substance abuse.

The NRC staff is currently revising 10 CFR Part 26 and is considering the use of the OSHA standard. The standard is "a significant injury or illness that results in death, days away from work, restricted work or transfer to another job, medical treatment beyond first aid, loss of consciousness, or other significant injury or illness diagnosed by a physician or other licensed health care professional, even if it does not result in death, days away from work, restricted work or job transfer, medical treatment beyond first aid, or loss of consciousness..." As such, the NRC staff encourages licensees to perform for-cause testing after accidents involving a failure in individual performance (human error) that results in a significant personal injury or an illness that is recordable at the time of the event, or reasonably could ultimately be recordable under the Department of Labor Occupational Safety and Health Administration (OSHA) standard in 29 CFR 1907.4 and subsequent amendments.

The staff recognizes that the requirements in 10 CFR 26.24(a)(3) contradict information in Section 8.3 of NUREG-1354. Subsection 8.3.3 of NUREG-1354 states: "The NRC believes that post-accident testing is necessary to determine whether the cause of the accident is related to substance abuse, if reasonable suspicion indicates that it may be drug related." The meaning of "reasonable suspicion" in the context of post-accident testing for substance abuse is not the same as in Criterion (iii). Criterion (iii) applies to the worker's behavior. Therefore, the staff is informing licensees not only of the staff's position on the meaning of 10 CFR 26.24(a)(3) but of the type of personal injury for which for-cause testing must be done. If violations of the for-cause testing requirements are identified during the course of NRC inspections, the NRC staff will consider enforcement discretion in accordance with Section VII.B.6 of NUREG-1600, "The General Statement of Policy and Procedures for NRC Enforcement Actions," on a case-by-case basis for violations that occur prior to and for a period of 30 days from the date of this RIS.

The aforementioned staff position regarding the meaning of 10 CFR 26.24(a)(3) was established through enforcement action taken against the Virgil C. Summer Nuclear Station. The enforcement action is described in detail in NRC Integrated Inspection Report No. 50-395/02-02; Exercise of Enforcement Discretion, at the following public internet link: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/REPORTS/sum_2002002.pdf.

BACKFIT DISCUSSION

This RIS communicates a staff position that may be viewed as a backfitting under 10 CFR 50.109 regarding the appropriate interpretation of the requirements with respect to "for-cause" testing in 10 CFR 26.24(a)(3) based upon the Virgil C. Summer Nuclear Station enforcement action. In particular, this RIS makes clear that "reasonable suspicion that a worker's behavior contributed to the event," as an element for determining whether to conduct "for-cause" testing, is relevant only in those circumstances where there is an accident involving a failure in individual performance resulting in actual or potential substantial degradations in the level of plant safety. There are NRC documents [e.g., Section 4.2.4 of NUREG/CR-5227, Fitness for Duty in the Nuclear power Industry, A Review of Technical Issues (September 1988); Section 4.1 of Supplement 1 to NUREG/CR-5227 (May 1989); and Section 4.3 of NUREG-1385, Fitness for Duty in the Nuclear Power Industry: Responses to Implementation Questions (October 1989)] that are consistent with this RIS.

However, the NRC also issued NUREG-1354, Fitness for Duty in the Nuclear Power Industry: Responses to Public Comments (May 1989), which is inconsistent with this RIS. This past guidance on "for-cause" testing after an accident stated, in part: (1) "Post-accident urinalysis screening is to be conducted as soon as possible after accidents involving a failure in individual performance which results in personal injury or in a radiation exposure or release of radioactivity in excess of regulatory limits if there is reasonable suspicion that the worker's behavior contributed to the event [p. 8-6]." (2) ". . . "Unless otherwise stated in 10 CFR

26.24(a)(3), the licensee should determine whether post-accident testing is necessary, based on reasonable suspicion and on the nature and severity of the accident [p.8-6]."

Due to the apparent inconsistencies in the guidance, questions have arisen during NRC inspections of licensees' FFD testing programs regarding "for-cause" testing after an accident involving a failure in individual performance. The interpretation in NUREG-1354 is incorrect, given the grammatical structure of the second clause in 10 CFR 26.24(a)(3).

Backfitting is defined in 10 CFR 50.109, (as well as in all other applicable regulations) in part, as a modification of or addition to the procedures or organization required to operate a facility, which may result from a new or amended provision in the Commission rules or the imposition of a regulatory staff position interpreting the Commission rules that is either new or different from a previously applicable staff position. The NRC has determined that the interpretation of 10 CFR 26.24(a)(3) found in NUREG-1354 is incorrect and cannot be relied upon by licensees in implementation of their FFD testing programs. This constitutes a different interpretation from a previously applicable staff position, and consequently, would be considered a backfit.

A backfit analysis is not required if the staff determines that one of the exceptions to the backfit rule is applicable. The staff has determined that the compliance exception, 10 CFR 50.109(a)(4)(i), (and all other applicable regulations) would apply in this case because any required modifications to a licensee's program would be necessary to bring the licensee into compliance with the correct interpretation of 10 CFR 26.24(a)(3), consistent with the three guidance documents identified above.

Consequently, a backfit analysis is not necessary because pursuant to 10 CFR 50.109(a)(4)(i), 10 CFR 70.76(a)(4)(i), and 10 CFR 76.76(a)(4)(i) we need to bring licensees into compliance with regulations.

FEDERAL REGISTER NOTIFICATION

A notice of opportunity for public comment on this RIS was not published in the *Federal Register* because it is informational and requires no specific action or written response.

SMALL BUSINESS REGULATORY ENFORCEMENT FAIRNESS ACT OF 1996

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

PAPERWORK REDUCTION ACT STATEMENT

This RIS does not contain any information collections and, therefore, is not subject to the requirements of the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.).

CONTACT

Please direct any questions about this matter to the technical contact listed below.

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

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

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Attachment: List of recently issued NMSS Generic Communications

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

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION
Office of Public Affairs Telephone: 301/415-8200
Washington, D.C. 20555-0001 E-mail: opa@nrc.gov
Web Site: http://www.nrc.gov

No. 05-162 December 2, 2005

NRC ACCEPTS GENERAL ELECTRIC'S APPLICATION FOR ESBWR ADVANCED REACTOR DESIGN CERTIFICATION

The Nuclear Regulatory Commission staff has accepted an application from the General Electric Company to certify the Economic Simplified Boiling Water Reactor (ESBWR) advanced nuclear power plant design, after determining the application has sufficient information to be formally "docketed" and reviewed.

The ESBWR is a nuclear power plant capable of producing approximately 1,550 megawatts of electricity. The plant features enhanced safety systems that rely on gravity and natural processes to safely shut down the reactor or mitigate the effects of an accident. It is designed for a 60-year operating life.

"Our staff has enough information to start an evaluation of the entire design," said David Matthews, Director of the New Reactor Licensing Division in the NRC's Office of Nuclear Reactor Regulation. "We'll ask GE for more details on the ESBWR, as needed, while the review goes forward."

If certification is granted, a company that wishes to build and operate a new nuclear power plant could choose to use the design and reference it in a license application. Safety issues resolved within the scope of the design certification are not subject to litigation with respect to an individual license application. Site-specific design information and environmental impacts associated with building and operating the plant at a particular location could be litigated. The NRC has certified three other standard reactor designs and is considering certification of a fourth later this year.

General Electric submitted its application Aug. 25 and provided supplemental information several times in September and October. The application (docket number 52-010) is available both in a *Federal Register* notice to be published shortly and on the NRC Web site at this address: http://www.nrc.gov/reactors/new-licensing/design-cert/esbwr.html .

During the staff's review of the ESBWR, they will continue to request additional information, if necessary, to properly analyze the design, and then issue an initial Safety Evaluation Report, which would identify remaining technical and safety questions to be resolved. A supplemental Safety

Evaluation Report will be issued when all technical and safety issues with the design have been resolved.

Once the design has passed staff review it can then be certified through NRC's rulemaking process, which is open to public participation. The certification process is described in Title 10 of the Code of Federal Regulations, Part 52, Subpart B. The design certification process normally lasts between 42 and 60 months.

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NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION
2443 Warrenville Road
Lisle IL 60532
Web Site: http://www.nrc.gov E-mail: opa3@nrc.gov

No. III-05-045 November 29, 2005
CONTACT: Jan Strasma (630) 829-9663
Viktoria Mitlyng (630) 829-9662

NRC ISSUES CONFIRMATORY ORDER AND REDUCES FINE FOR RADIATION SAFETY VIOLATION AT LASALLE

The Nuclear Regulatory Commission staff issued an Order confirming commitments made to the NRC by Exelon Generation Co. and reduced the amount of a fine as part of a settlement agreement concerning a violation of radiation safety requirements at the company's LaSalle County Nuclear Power Station. The plant, which has two reactors, is located near Seneca, Ill.

The violation occurred when four employees of a contractor working at the LaSalle Station entered a high radiation area without authorization on Jan. 25, 2004.

NRC investigators determined that the violation was willful in that the foreman and two of the workers were aware they were not authorized to enter the high radiation area and had not received the briefing by radiation protection personnel necessary to enter the area.

The workers did not receive a significant radiation exposure. The maximum radiation exposure received was 5 millirem, which is a small fraction of the NRC limit of 5,000 millirem per year for workers at nuclear facilities.

The NRC issued Exelon a Notice of Violation and proposed a $60,000 fine on May 2, 2005 (see NRC press release issued May 4, 2005).

On May 12, 2005, Exelon announced its intention to appeal the NRC's enforcement action through the use of Alternate Dispute Resolution (ADR), a process used to help the NRC and the utility to reach agreement.

As part of this process, the utility and the NRC staff met with an independent mediator and reached a settlement agreement. The utility acknowledged that the violation had occurred and committed to carrying out extensive corrective action to address the problem.

The NRC reduced the fine from the proposed $60,000 to $10,000 as a result of the utility's commitments to improve radiation safety rules, procedures and awareness at the plant. Exelon has until December 22 to pay the fine.

"The commitments made by the utility to make sure radiation safety rules and procedures are properly understood and enforced will help the NRC gain stronger confidence that nuclear workers are protected from excessive doses of radiation at the LaSalle plant," said James Caldwell, NRC Regional

Administrator.

The Confirmatory Order to Exelon and other documents related to this case are available from the Region III Office of Public Affairs and on the NRC web site at: http://www.nrc.gov/reading-rm/adams.html. Enter docket number 05000373 as a search term.

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NRC NEWS

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NRC PROPOSES $60,000 FINE AGAINST D.C. COOK FOR FAILURE TO PROVIDE COMPLETE AND ACCURATE INFORMATION

The Nuclear Regulatory Commission staff has proposed a $60,000 fine against Indiana Michigan Power Company for failing to provide complete and accurate information and meet reporting requirements for NRC-licensed operators at Donald C. Cook Nuclear Power Plant. The plant is located near Bridgman, Mich.

The violations were identified during an NRC inspection (NRC Inspection Report No. 05000315/2005006) conducted at D.C. Cook in spring and summer 2005 to review the plant's reactor operator licensing program. The inspection also reviewed corrective actions undertaken to address a previous violation (EA-04-109 issued Sept. 29, 2004) in the same area. NRC inspectors identified three violations: (1) the utility had provided the NRC with incomplete and inaccurate information. The utility stated that a complete review of all operator medical records had been conducted and that no records that would require restrictions to operator licenses for medical reasons had been found. However, NRC inspectors identified three licensed operators who had medical conditions that would require their licenses to be restricted; (2) the utility had failed to notify the NRC about licensed operators experiencing a permanent illness within 30 days. Two NRC-licensed operators at the plant were diagnosed with potentially disqualifying medical conditions in 1998 and 2003. However, the NRC was not notified of these facts until 2005; (3) the utility also failed to provide the NRC with complete and accurate information on NRC reactor license applications. Applications submitted to the NRC for new, renewed and amended NRC licenses did not describe the individuals' recently diagnosed medical conditions that would affect the conditions of these licenses.

"Reactor operators licensed by the NRC are entrusted with safe operation of nuclear reactors and must be capable of performing their assigned duties. For that reason, their physical condition and general health are significant concerns of the NRC and are closely monitored," said NRC Regional Administrator James Caldwell. "Providing the NRC with accurate and timely information on changes in reactor operators' health that may affect their ability to perform their duties is key to the agency's ability to fulfill its mission of protecting public health and safety."

Indiana Michigan Power Company has taken such corrective actions as developing guidance for the submission of reactor operator application forms; revising administrative procedures to discuss regulatory requirements with the medical review officer prior to performing the annual medical records review; training operators on the requirements to report a change in medical condition.

The notice to the utility on the enforcement action will be available online at http://www.nrc.gov/what-we-do/regulatory/enforcement/current.html and from the NRC Region III Office of Public Affairs. D.C. Cook inspections reports are available through the NRC's online document collection, known as ADAMS, at http://www.nrc.gov.gov/reading-rm/adams.html by entering docket number 50-315 and 50-316.

The utility has until December 18 to pay the fine or to protest it. If the fine is protested and subsequently imposed by the NRC staff, the utility may request a hearing.

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Inside NRC

Volume 27 / Number 24 / November 28, 2005

Commissioners anxious about managing COL requests, regulatory stability

How NRC will handle the barrage of combined construction permit-operating license (COL) applications expected beginning in late 2007 and address industry's concerns about maintaining regulatory stability for new plant licensing reviews were two major topics the commissioners debated at a public meeting last week.

The number of COL applications likely to be filed in fiscal 2007 and 2008 has jumped from four to 10 since the staff last briefed the commissioners in April, Office of Nuclear Reactor Regulation (NRR) Director Jim Dyer said at the Nov. 21 meeting. And there could be more in 2009 and 2010, he told the commissioners.

An NRR reorganization implemented in late October, the groundwork laid to expand NRC's contractor base, and the launch of an aggressive recruitment plan are several ways the staff is preparing to meet the new plant licensing demand, Dyer and other senior NRC managers said.

NRC Chairman Nils Diaz said at one of two briefings Nov. 21 that new plant activities could rival the level of effort the agency has invested in security issues over the past four years. Utility executives and vendors were invited to speak and answer questions about new reactor issues at a morning session, and the NRC staff provided a briefing in the afternoon.

Commissioner Peter Lyons questioned electric company representatives as to whether the enactment of the Energy Policy Act, and particularly the production tax credits established in the framework, was driving the recent surge of announcements for potential future construction.

Lyons said he was concerned the act was creating an "artificial rush to be the first through the gate." Yet the "gate" hasn't even been fully defined, he stressed. Under the new law, a 1.8 cents per kilowatt-hour production tax credit would be available for up to 6,000 megawatts of capacity from advanced reactors certified by the NRC and placed in service before January 2021. DOE was directed to draft regulations establishing a certification process for the tax credits by early February. Depending on how the DOE guidance is worded, it could put additional stress on NRC's licensing review schedules, Lyons said.

The law also offers "standby support," or investment protection to offset the financial impact of NRC licensing or litigation delays for the builders of the next six new plants, and loan guarantees for up to 80% of a project's cost. DOE must finalize a rule prescribing details of the risk insurance provisions within a year of the law's Aug. 8 enactment. James "Barnie" Beasley Jr., president and CEO of Southern Nuclear Operating Co., said the industry is working with DOE through the Nuclear Energy Institute on implementing guidance for the production tax credits. He pointed to the projected growth in electricity demand in his service territory as the main driver for Southern Nuclear's interest in potentially building new nuclear baseload capacity.

Southern Nuclear plans to submit an early site permit (ESP) for its Vogtle site in August 2006, and cite the ESP work in a COL application that is expected to be filed in March 2008.

C.S. "Scotty" Hinnant, senior vice president and chief nuclear office of nuclear generation for Progress Energy, attributed three factors to the industry's focus on new nuclear plant projects: an increase in customer growth, lack of new baseload facilities since the mid-1980s, and the surge in fuel costs, particularly for natural gas. Those forces, in addition to the Energy Policy Act's financial incentives, were affecting utility executives' decisions on the range of energy options, he said.

Need for speed?

Several commissioners reminded utilities that the industry's expectations for speedy licensing reviews must be balanced against available agency resources and the demands of addressing environmental, safety, and design issues. Commissioner Gregory Jaczko said he was concerned the industry was trying to compress licensing review schedules based on its need for having new baseload generation available by 2015. Truncating the review schedule would mean increasing the staff's workload, he said. Given the choice, would the industry rather have the agency review fewer COL applications at once or deal with more applications in parallel, but at a slower review pace, he asked the utilities. Eugene Grecheck, vice president of nuclear support services for Dominion, suggested there might be a "third option." Southern Nuclear's Beasley said he would prefer to have multiple applications in review at the same time. But he said he would hope the agency could do whatever is possible to complete them as soon as possible.

Commissioner Edward McGaffigan later put the same question to the NRC staff, asking whether the agency should allow only a few applications to go through the review process or accept "all at a ponderous rate?" Senior managers had different views on the issue.

Dyer said he preferred to finish one high-quality application first. McGaffigan said that he, too, preferred this approach. McGaffigan noted at the earlier new reactors briefing that the agency had handled license renewal applications in that manner, finishing a couple before "dealing with a tidal wave."

But Executive Director for Operations Luis Reyes said he believed the agency could find another way to handle the applications. He suggested grouping them by reactor design to increase review efficiencies.

But McGaffigan said organizing the reviews in that manner might not alleviate the bulk of contentions, which tend to be site-specific issues related to the environment, emergency planning, and security.

The staff plans to submit a paper to the commissioners in January providing options for handling the anticipated flood of COL applications.

McGaffigan told the utility executives that he would not allow their internal business pressures to dictate NRC review schedules. "We will not have our schedules written by your public affairs people talking to Wall Street," he said. At the afternoon session, McGaffigan said he was taken aback by the number of industry projects currently proposed, which the staff said could total up to 15 new units.

Roughly calculating the generating capacity that would be produced if all the projects were approved, McGaffigan rhetorically asked whether there was "really a need for 22.5 gigawatts" of new baseload capacity by 2015.

Regulatory stability
The commissioners reiterated their message that the industry was responsible for submitting high-quality, complete applications for review.

Brew Barron, Duke Power chief nuclear officer, told the commissioners that developing a high-quality application means referencing "clearly established standards." Currently, the agency's standard review plan for new plant licensing has not yet been completed, he said. So the staff will have to clearly define those standards, he said.

Barron and other utility officials said the industry was coordinating closely to standardize the reactor designs and, in turn, the applications. Barron said the industry understands that the NRC "can't give blanket approval" to applications, but it wants to eliminate redundant reviews. The industry anticipates using the same methodologies and identical text, wherever possible, in applications, he said.

Randy Hutchinson, senior vice president of business development for Entergy Nuclear, suggested that the agency retract the proposed rule revisions for 10 CFR Part 52, which were recently released (INRC, 14 Nov., 8). In his presentation, he called the licensing requirements in Part 52 "workable, although not perfect." He said amendments will be needed but that now was not the time to make extensive changes because the proposed revisions would increase regulatory uncertainty. If the rulemaking proceeds and is completed in early 2007, that would give the first applicant about six months leeway to revise its application. If the industry is in such a time crunch to build new baseload plants why not use Westinghouse's AP1000, which is almost through the design certification rulemaking process, rather than choose GE Energy's ESBWR design, which is just starting the certification process, McGaffigan asked the executives. He noted that the process was already at least three months behind because the staff did not find GE's August submittal to be complete enough to begin its technical review.

McGaffigan indicated that 2010 was an "optimistic date" for COL applicants who reference the ESBWR and expect to be finished with the licensing process.

Grecheck said Dominion believes the design was worth the wait for the additional licensing effort. Marilyn Kray, vice president of project development for Exelon Nuclear and president of the NuStart Energy consortium, said many in the industry are attracted to the improved safety and economics of the design.

Steve Hucik, GE Energy's general manager of new reactor programs, spoke of the ESBWR's passive features, which he said eliminates much of the piping and pumps that are in the current LWR fleet and consequently reduces operator surveillance and inspection requirements.

Rulemakings considered

Hutchinson suggested the agency consider only making regulatory changes to conform to requirements in the Energy Policy Act. Those changes could be made under an interim final rule, he suggested, setting off a debate among the commissioners and NRC General Counsel Karen Cyr as to whether such an approach was possible or would result in time savings.

McGaffigan said he believed the agency could revive the 2003 proposed revisions to Part 52 and finalize them more quickly than if it started fresh. He said the agency would only have to address the comments it received. Starting over would require a 75-day comment period, and possibly longer, before the agency could begin addressing the feedback, he said. NRC promulgated the Part 52 improved licensing processes in 1989, but parts of the regulations have never been fully tested.

Entergy's Hutchinson agreed that resurrecting the 2003 proposal was worth investigating.

NRC's Dyer, however, said that addressing the comments filed on the 2003 rulemaking proposal meant "we'd have to go back to square one." He said that effort could take months. But he agreed to review the previous proposal, noting he had only been briefly on the job as NRR director when it was issued.

Progress Energy's Hinnant had earlier called the existing Part 52 "sound and viable regulations that we can work with."

Other reactor projects

Lyons asked the staff whether its work plan included a review of a small, liquid metal reactor proposed by officials in Galena, Alaska and an advanced nuclear power plant capable of co-producing electricity and hydrogen. The Energy Policy Act calls for constructing a demonstration cogeneration reactor at the Idaho National Laboratory, and there is authorization for additional funds to demonstrate hydrogen production at two existing nuclear plants.

Regarding the Idaho so-called Generation IV reactor project, Lyons said he has seen different dates from the Idaho delegation and White House Office of Management & Budget regarding a project schedule. Commissioner Jeffrey Merrifield skeptically said the agency should expect DOE's license application for the cogeneration project "sometime after we received the Yucca Mountain license application." Reyes said the Galena project appears to be "frozen."

City officials have not communicated any further plans to develop the reactor, he said. He promised that the staff's fiscal 2008 budget assumptions, which are due to the commission in early January, would be as "precise" as possible on all likely new reactor projects.—*Jenny Weil, Washington*

NRC staff outlines ambitious agenda in latest update to risk-informed plan

A wide-ranging regulatory and research agenda was outlined by NRC staff in the most recent update of its riskinformed regulation implementation plan (Secy 05-199),

released Nov. 17. The paper lists and prioritizes NRC's proposed activities in the area through fiscal 2007. Resources for these activities, except for a few that have been deferred and are listed as "on hold," have been budgeted in FY-06, Executive Director for Operations Luis Reyes said in the paper.

In his Sept. 12 speech to the international topical meeting on probabilistic safety analysis (PSA '05), Chairman Nils Diaz called for "full implementation" of the commission's 1995 policy statement on risk-informed regulation, which "should result in a predictable and timely regulatory approach, one that integrates and optimizes reactor safety, security, and preparedness through risk management." This approach "must use the best available information from operating experience and research, the best available techniques, including risk-informed and performance-based regulation; and it must resolve the relevant issues in the right progression and be realistic and implementable," Diaz said. Over the next two fiscal years, NRC staff plans to complete a range of activities to risk-inform its nuclear reactor safety regulations, and is close to presenting its roadmap of risk-informing regulations more generally under Part 50.

Risk-informed regulatory applications

Over the last several years, NRC staff have begun to integrate risk-informed approaches into many areas of regulation (INRC, 30 May, 1), including the agency's reactor oversight process (ROP). Reactor safety "notebooks," used in the ROP's significance determination process and originally issued in 2003, are being revised to incorporate "basic pre-solved tables" that "contain a comprehensive target set of approximately 40-50 plant-specific key components and operator actions," the staff said in its paper. Selection of these items was based in part on those "components and equipment issues typically encountered in ROP inspection activities or selected to test the notebook's model and logic." The tables are scheduled to be completed by the end of this year.

"Staff has been developing a framework to address the coherence of regulatory activities," which "will provide an approach (guidelines and criteria) to ensure that the reactor regulations, staff programs, and processes are built on a unified safety concept and are properly integrated so that they complement one another," staff said in Secy 05-199. If the implementation plan for the project is approved by the commission, the draft framework will be issued for NRC internal review in January 2006, and the final framework "issued for use" in June.

A staff plan for more comprehensive risk-informed, performance-based regulation under 10 CFR 50, including eventual revision of regulations, regulatory guides, and standard review plans, is scheduled to be sent to the commission next month (INRC, 5 Sept., 5).

"Most of the critical milestones necessary" to implement the mitigating systems performance index (MSPI) "need to be resolved by industry before January 2006." This "target date is achievable if identified milestones are met," but "completion of the milestones depends on industry resolution of implementation issues." Completion date for development of an external event assessment tool for the SDP is yet to be determined. The staff and industry, however, are no longer planning for the MSPI to be put in use at the start of the year (see story, page 1).

Under NRC's program to incorporate risk information into the high-level waste regulatory framework, the staff plans by the end of 2005 to complete "development of high-level waste inspection procedures using risk insights" and to "complete seven integrated inspection procedures." Version 5.0.1 of the total-system performance assessment code, being developed by NRC staff and contractors at the Center for Nuclear Waste Regulatory Analyses to assess projected repository performance, is now scheduled to be completed by March 2006.

NRC staff expects to complete "draft data collections" and an analysis report for use in advanced reactor PRA reviews this month. A report due in February 2006 will explore modeling of GE Energy's ESBWR reactor's passive systems in a PRA, "including an assessment of the impacts from using enhanced passive system PRA modeling as compared to the traditional PRA practice."

Over the next year, NRC's decommissioning program (INRC, 17 Oct., 8) will continue to "develop guidance for staff licensing reviews that will give further details about the risk-informed approaches to institutional controls, engineered barriers, and exposure scenarios," according to the update. The guidance is scheduled to be completed in

September 2006. Until then, "staff will continue to implement these new approaches at specific sites" and "the sitespecific lessons learned are expected to enhance the guidance development process."

Specific initiatives

The risk-informed approach is being applied to specific regulatory issues as well. Because "there are no widely accepted methods for including software failures of real-time digital systems into current generation probabilistic risk assessments (PRAs)," NRC staff is developing "methods and tools for analyzing digital systems reliability that are consistent with a riskinformed approach to decisionmaking." A letter report that "documents the development of a method to identify system state and transition rates and quantify system failure probabilities for dynamic methods" is targeted for completion by NRC's Office of Nuclear Regulatory Research (RES) in April 2006, with a draft regulatory guide on risk-informed digital system reviews to be published for public comment in June. Another letter report on software-induced failure experience is slated for November 2006.

Guidance for NRC inspectors to resume inspections of postfire safe-shutdown electrical circuits is scheduled to be completed by January (INRC, 31 Oct., 5).

The staff also has plans to complete another fire protection-related guidance document. "Complete fire PRA review guidance for NRR specialists" in accordance with regulations endorsing the National Fire Protection Association's standard 805, which details a risk-informed approach to fire protection (INRC, 17 Oct., 13), is due to be completed by the end of 2005, the staff update said.

A final report from RES on recommended changes associated with pressurized thermal shock screening criteria is scheduled to be sent to the Office of Nuclear Reactor Regulation (NRR) in December for consideration of possible rulemaking (INRC, 13 Dec. '04, 7).

"Development of a technology-neutral framework/guideline" for new plant licensing continues (INRC, 19 Sept., 12), with a draft of the framework to be released for public comment in June 2006. The final framework is scheduled to be issued in June 2007.

RES and Spent Fuel Project Office staff are developing a spent fuel storage cask PRA to "provide a methodology to quantify the risks of dry cask storage of spent nuclear fuel and apply the methodology to a specific cask design at a specific site." The draft study "determined that a stainless steel welded canister with a concrete overpack poses a very low risk to the public," the staff update said.

The Advisory Committees on Reactor Safeguards and on Nuclear Waste (ACRS and ACNW) are to be briefed on the final pilot PRA by May 2006, with the final pilot PRA issued as a Nureg report sometime next year.

PRA quality

Agency staff and industry organizations continue to pursue diverse projects in the area of PRA quality (INRC, 27 June, 5). Publication of a revision of regulatory guide 1.200, NRC's plan for improving the quality of licensee PRAs, including three appendices specifying PRA quality criteria, is scheduled for January 2006.

Final standards for low-power/shutdown PRAs being developed by the American Nuclear Society (ANS) are anticipated in December, and revision 1 of the group's external events PRA standard is expected by January 2006. An appendix detailing NRC staff's position on the low-power/shutdown standard is scheduled to be added to RG 1.200 by the end of 2006.

ANS also expects to issue its draft final internal fire standard next month and to finalize it by June 2006. NRC expects to add the staff's position on the standard to RG 1.200 by June 2007.

RES is developing models and guidelines under its risk assessment standardization project, known as RASP (INRC, 8 March '04, 14). In 2006, RASP will provide support to NRC's development of four sets of guidelines for events during power operations (internal flooding, internal fire, high-wind and seismic events), and for internal fire events during lowpower and shutdown operations. RASP will also support development of guidelines for calculation of "large early release frequency," a key risk component of PRAs, by September 2006.

A draft staff Nureg report on treatment of PRA uncertainties and use of alternate methods (INRC, 6 Sept., 4) is to be published in June 2006 for public review and comment. There are several "ongoing activities to improve the quality of human reliability analysis (HRA)" in PRAs. In fiscal 2005, staff plans to continue development of the human event repository and analysis system, known as HERA, the subject of a draft Nureg report published in September. "Beyond its primary objective of providing quality data for

HRA applications, HERA can also provide a means of obtaining an agreement among experts on the quantification of human error." Publication of a Nureg report on "the evaluation of current HRA methods with respect to HRA good practices," originally expected in December, is now scheduled for September 2006 (INRC, 2 May, 5).

Secy 05-199 is available on NRC's Web site (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2005/secy2005-0199/2005-0199scy.pdf).
—Steven Dolley, Washington

Inside NRC

Volume 27 / Number 24 / November 28, 2005

NRC preparing to launch effort to improve corrective actions

The NRC staff is preparing to implement an agency-wide corrective action program (CAP) to make sure that problems such as those that led to the Davis-Besse vessel head degradation not only are fixed but are "institutionalized" so that incidents don't recur, the head of an agency task force told the commission this month.

Loren Plisco, the Region II deputy administrator and team leader of the Institutionalizing Agency Lessons Learned Project, told the commissioners at a Nov. 1 briefing that the program is part of a larger effort on "knowledge management," dealing with the way the agency archives and conveys information.

Plisco's work was built on an earlier document, a report by the Effectiveness Review Lessons Learned Task Force (LLTF), which was part of the agency's response to Davis-Besse's discovery of severe vessel head degradation in March 2002. The plant subsequently stayed down until March 2004 to fix its problems.

One of the follow-up activities recommended by the LLTF was a review of the effectiveness of previous LLTFs. The Effectiveness LLTF completed its report in August 2004, and the commission was briefed in December. The staff requirements memorandum for the meeting said the commission "enthusiastically" supported the corrective action plan (INRC, 27 Dec. 04, 5). The effectiveness report recommended that NRC "establish an agency-wide CAP that has the fundamentals and accountability" of 10 CFR Part 50, Appendix B, the section of the regulations dealing with the CAPs that NRC licensees are required to maintain.

The effectiveness report also said the NRC needed a centralized tracking system for corrective actions, a more rigorous system for closing out corrective actions, and effectiveness reviews as part of the close-out process.

Overall, Plisco said, there will be "more rigor and formality" in tracking corrective actions. Examples of that would be a management directive and a handbook describing the process, he said. He also said there would be more management involvement in corrective actions and close-outs of these actions, as well as dedicated staff in the Office of the Executive Director for Operations (EDO).

Plisco said the schedule called for launching a pilot program in March, finalizing the management directive in April, and fully implementing the program beginning in June. But he said there is another phase of the program—in addition to ongoing program improvements—tying his team's work to the agency's broad goal of improving knowledge management.

One part of that, Plisco said, is "configuration management," which, he said, means "making sure that before changes are made, the staff understands the history of what they are changing." In a Nov. 8 interview, Plisco cited as an example the agency's emergency preparedness procedures, which were changed after Hurricane Andrew in 1992. Under the new system, the description of NRC's procedures might have a link to the memorandum of understanding between NRC and the Federal Emergency Management Agency, he said. That memorandum was a direct result of lessons NRC learned from the hurricane, Plisco said.

Emphasis on story-telling

More broadly, he said at the briefing, "we need to summarize the information in a new way, using a story-telling approach."

EDO Luis Reyes said the objective is to go beyond "the old tribal story-telling, which is mostly what we used in the past, to one that is institutionalized, where you have either a training requirement or a qualification process" or some other formal way of conveying the information.
As an example, he said, the agency has created a training session on the 1979 Three Mile Island-2 accident, since "every engineer" that the NRC is hiring out of college "was either not born or was in diapers when TMI happened." The agency also has created a video on licensing for similar reasons, he said. But there is not yet a similar tool dealing with Davis-Besse, Reyes said in response to a question from Commissioner Gregory Jaczko.

Jaczko suggested that improved knowledge management would be important also because "some of our best people" who "really dealt with a lot of the lessons learned" are likely "to want to go the new exciting area of new reactors."

Commissioner Edward McGaffigan said the knowledge management initiative, including the new CAP, probably is "the most important thing we are going to be doing in the next several years." He asked the staff to keep the commissioners informed on "where we are on resources" for the initiative. Citing recent conversations, McGaffigan said Sen. George Voinovich (R-Ohio), the chairman of a Senate subcommittee with oversight of NRC, is "terribly interested" in the issue. Voinovich and Delaware Sen. Tom Carper, the top Democrat on the subcommittee, are likely to be willing to "help us on the resource side," McGaffigan said.

In the interview, Plisco said he and other NRC managers were still "working on the details" of determining the financial and personnel resources that will be needed for the knowledge management effort. He said the CAP initiative was not the first undertaking of that effort, but he acknowledged it probably has a higher profile than the others.

At the briefing, Commissioner Peter Lyons, while strongly supporting the effort, also raised some cautions, saying it was going to be difficult "to define what are the important findings" that would be given high priority under the new system. Plisco acknowledged the difficulty, saying that LLTFs "do a great job of finding issues, but everything they find, they put in their report." The CAP will have to focus on issues that are root or contributing causes to problems, he said.

The forthcoming management directive will define the criteria for including issues in the CAP, he said. He also emphasized that items falling below the threshold will still be tracked to completion, although the process would not have the same requirements as the one for the higher-priority issues. Lyons also said he was "very concerned" that having a dedicated staff and "a package of resources" for the effort could lead other staffers to see the issue as "someone else's problem." The test of "whether we have truly learned from Davis-Besse," he said, is "whether every employee, including every new employee, truly has institutionalized the lessons from Davis-Besse and truly has developed that questioning attitude." The review of previous LLTFs was one of 49 Davis-Besse LLTF recommendations that the staff has been implementing. Jim Dyer, the director of the Office of Nuclear Reactor Regulation, said that 48 were now completed. Brian Sheron, the associate director for engineering and safety systems, said the one outstanding recommendation deals with the revision of the requirements for upper vessel head inspections. But that issue has involved protracted negotiations among the NRC, the nuclear industry, and the American Society of Mechanical Engineers (ASME). The plan has been to modify the relevant section of the ASME code and then incorporate those requirements into NRC's regulations (10 CFR 50.55a).

At the briefing and in a subsequent interview, Sheron said he expects to be able to accept the code case, with certain "conditions," or modifications. He said the proposed rule on revising the NRC inspection requirements was scheduled to be published next summer, with the final rule following about a year later.

Leak monitors not required

Sheron also said the staff had decided not to require operators to install leak-detection equipment to detect reactor coolant system (RCS) leakage below current technical specifications (typically 1 gallon per minute). Argonne National Laboratory had identified technologies that could do that (INRC, 10 Jan., 8), but the staff determined that "implementing these increased capabilities would likely result in only a small reduction in loss-of-coolant accident frequencies at best," Sheron said.

However, Sheron said, the agency and industry are continuing to work on leak detection, since even small leaks, if allowed to continue for a long time, can lead to large boron deposits. There continue to be examples of boron accumulation inside containment, although "we haven't seen examples of major problems," he said.

Operators typically have administrative leakage limits below tech spec limits, and if they see that the leakage is increasing, even if it is staying below the tech spec limit, the operators generally shut down their plants, he said. But even if there is no upward trend, such leakage is a potential concern and needs to be pursued, he said. Sheron said one option the staff is considering is imposing a time limit on such leakage.

Sheron noted that there are industry efforts under way to address leakage issues. The Westinghouse Owners Group and BWR Owners Group are developing voluntary guidelines for the monitoring and management of RCS leakage (INRC, 10 Oct., 7). Also, Sheron said, the NRC staff is working with industry groups that are developing uniform responses to such leakage.—*Daniel Horner, Washington*

Entergy Vermont Yankee
Extended Power Uprate

Presentation to the
Advisory Committee on Reactor Safeguards
December 7, 2005

Steam Dryer
Key Points

Entergy

- Acoustic Loads are Primary Source of Dryer Significant Degradation
- VY Measurement Configuration Detects Acoustic Loads
- VY Dryer:
  - No CLTP Acoustic Resonance
  - Substantial Margin to ASME Stress Limit
  - Baseline Inspection – No Structural Vulnerabilities
  - Modified to Strengthen for EPU Operation
  - Power Ascension Controlled via Monitoring Plan

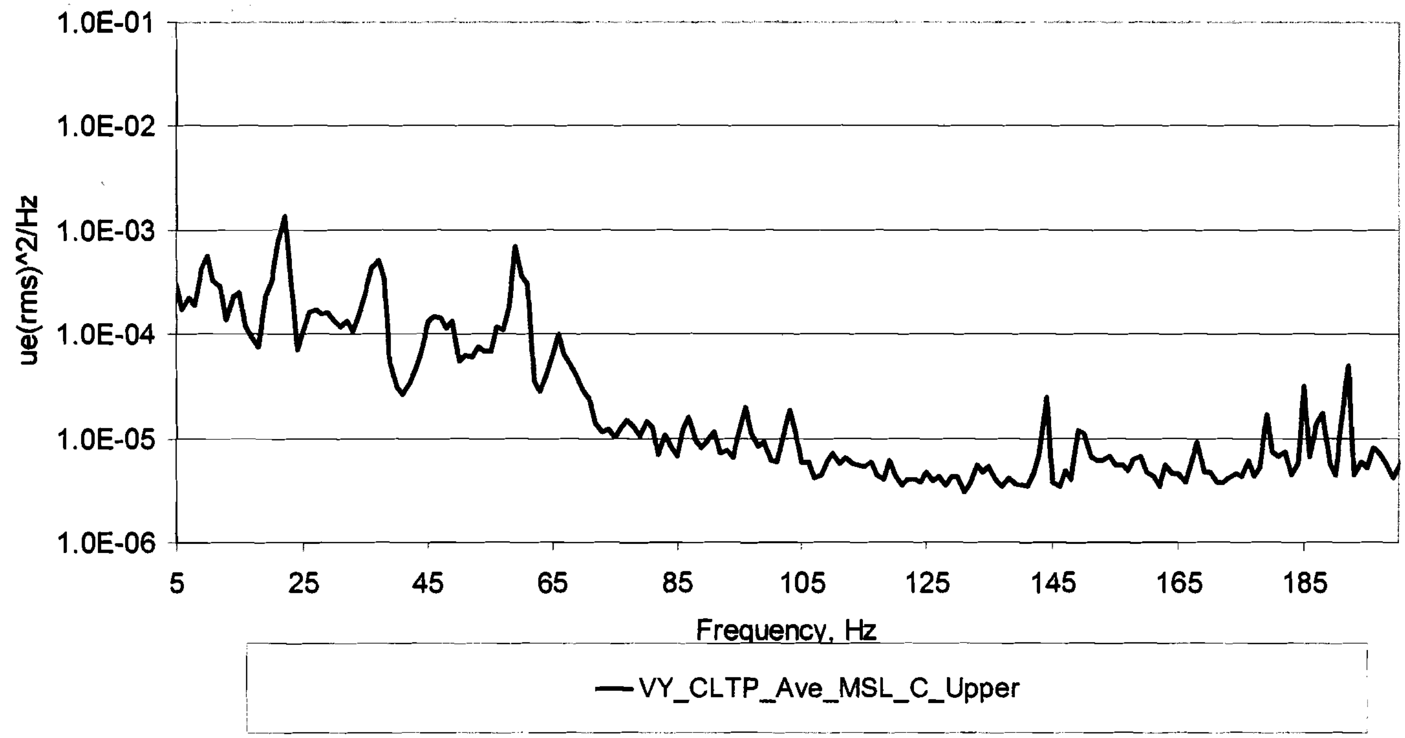
Steam Dryer
VY Main Steam Vibration

- MSL Measurements
  - Strain Gages
  - Accelerometers
- Main Steam Branch Line Potential Acoustic Resonators
- MSL Monitoring Will Detect Excitation from Sources in:
  - Main Steam Lines
  - Reactor Vessel

Steam Dryer

SG Measurements for Acoustic Monitoring

Figure 1a: VY SG Data CLTP

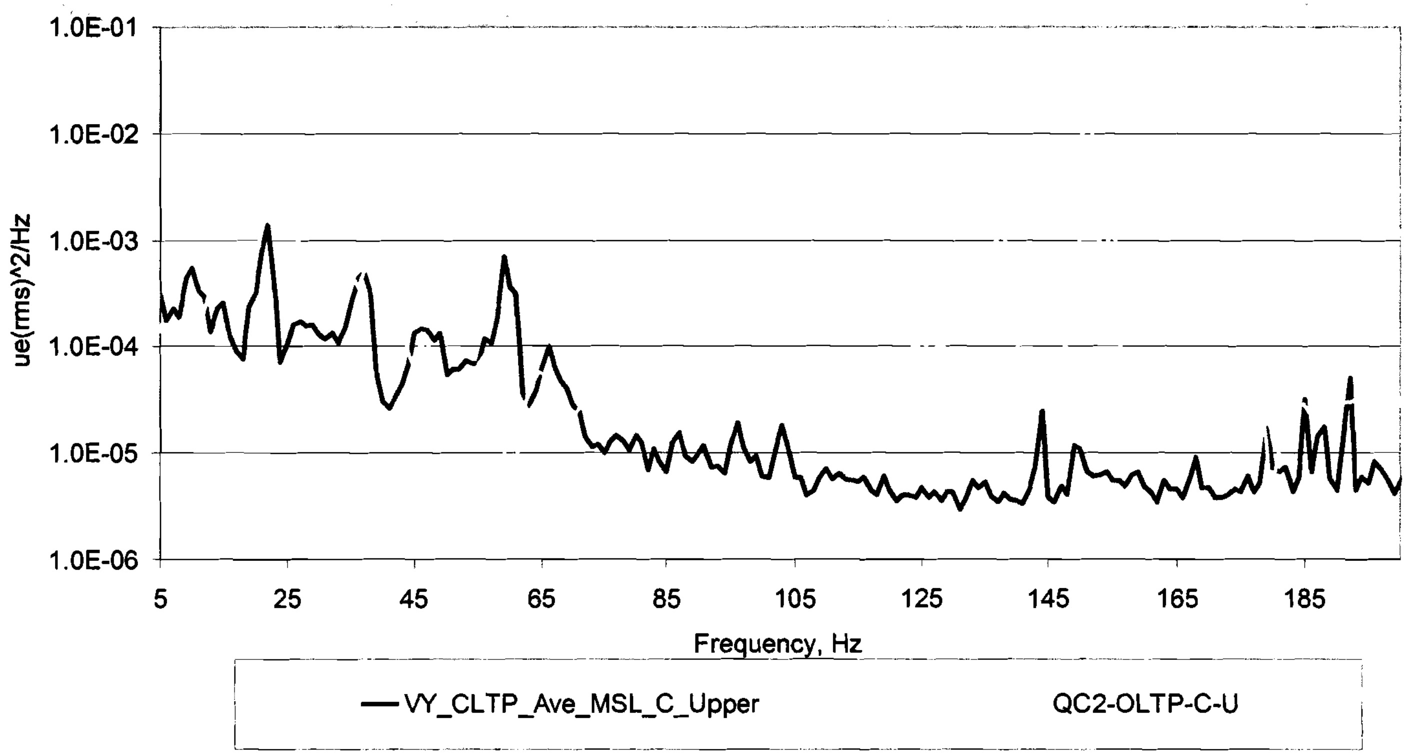


Entergy

Steam Dryer

SG Measurements for Acoustic Monitoring

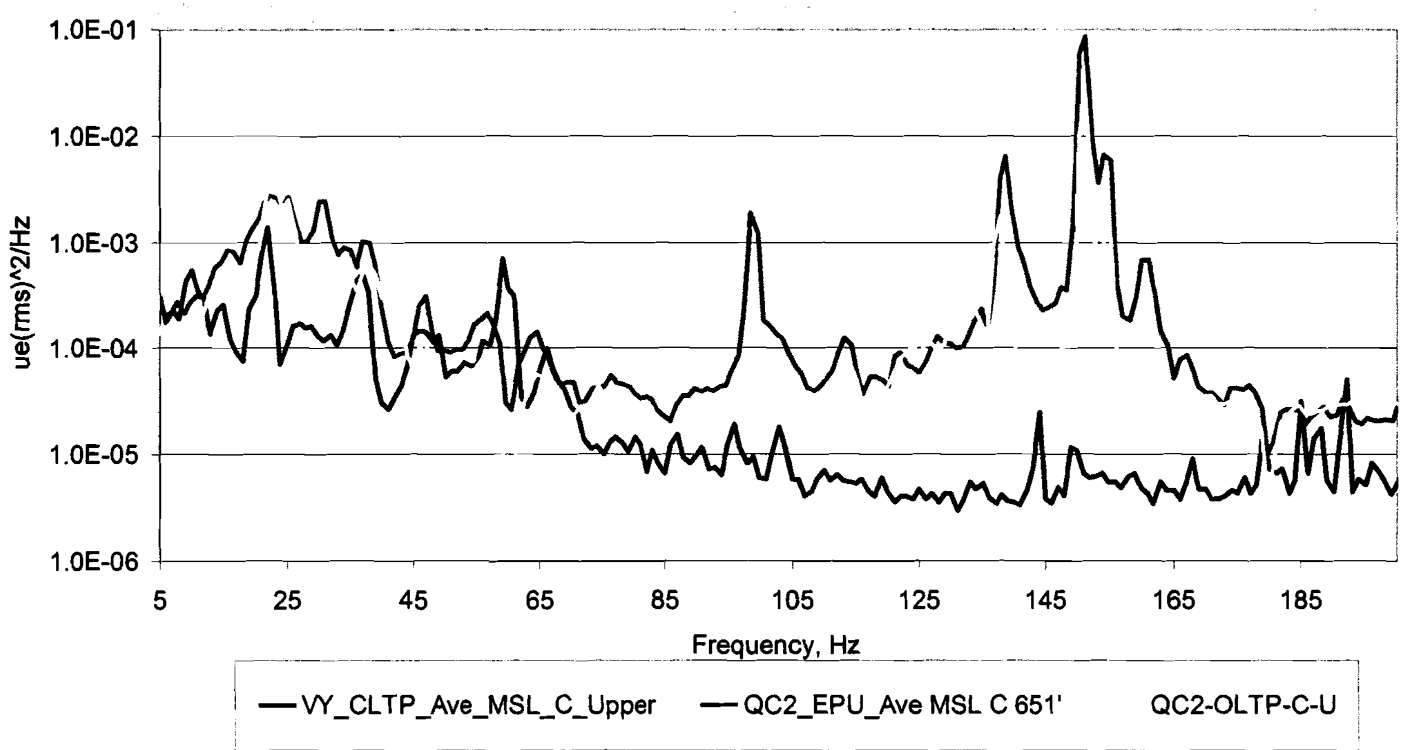
Figure 1b: VY SG Data and QC OLTP



Steam Dryer

SG Measurements for Acoustic Monitoring

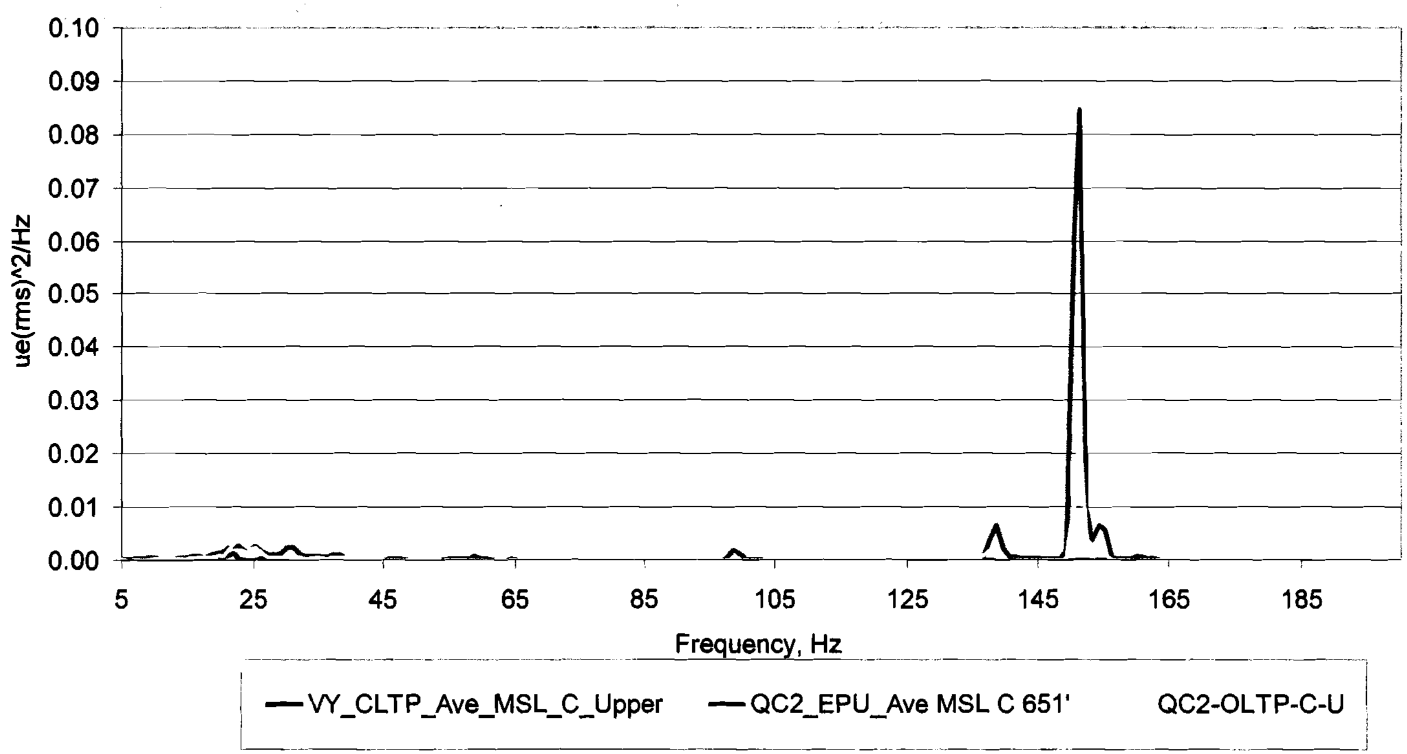
Figure 1c: VY SG Data and QC OLTP & EPU



Steam Dryer

SG Measurements for Acoustic Monitoring

Figure 1d: VY SG Data and QC OLTP & EPU (linear)



Steam Dryer

VY Structural Analysis – Load Definition

- Acoustic Loads Impact Vertical Faces of the Dryer
  - Generated from Steam Line Data
  - Transfer of Steam Line Data Benchmarked via QC Data

- Turbulent Forces Act in the Area of the Nozzle
  - Little Effect on Dryer Components

Steam Dryer

VY Structural Analysis – Peak Stress & LCF

VY Dryer Limiting Component – Vertical Face Top Weld

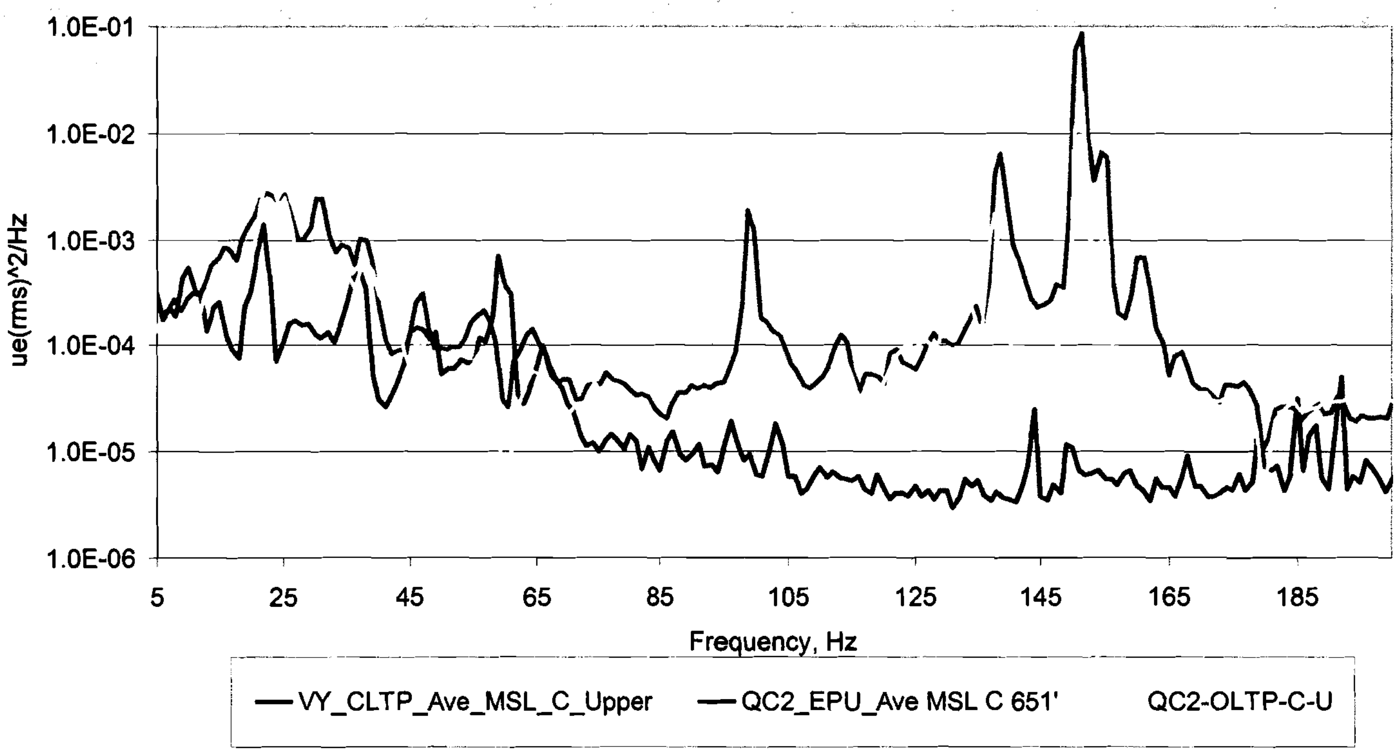
- Peak Calculated Stress 5,450 psi
- ASME Acceptance Criterion 13,600 psi
- Limit for Power Ascension 7,400 psi
  - Reflects 2.78 Limit Curve Factor

Entergy

Steam Dryer

SG Measurements & Power Ascension Limit Curve

Figure 2a: VY SG Data and QC OLTP & EPU

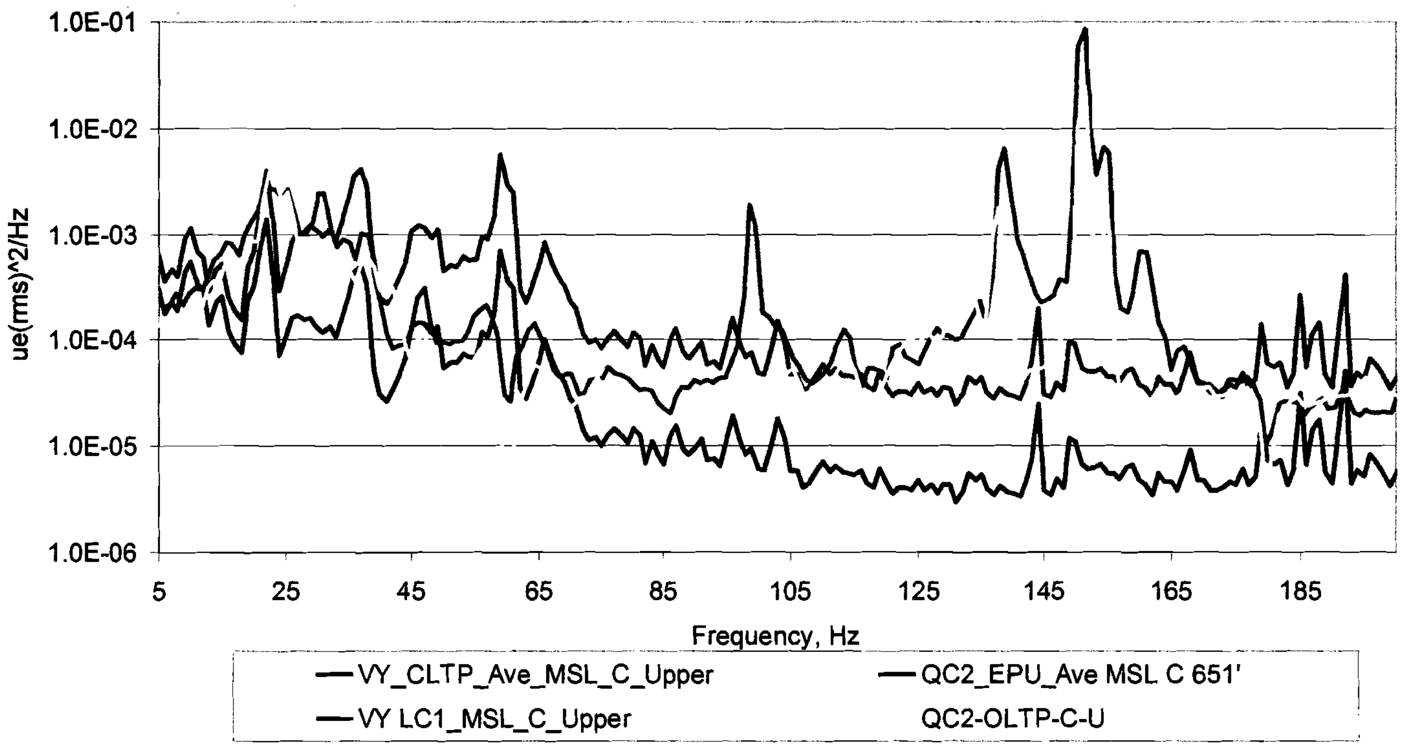


Entergy

Steam Dryer

SG Measurements & Power Ascension Limit Curve

Figure 2b: VY SG Data and QC OLTP & EPU

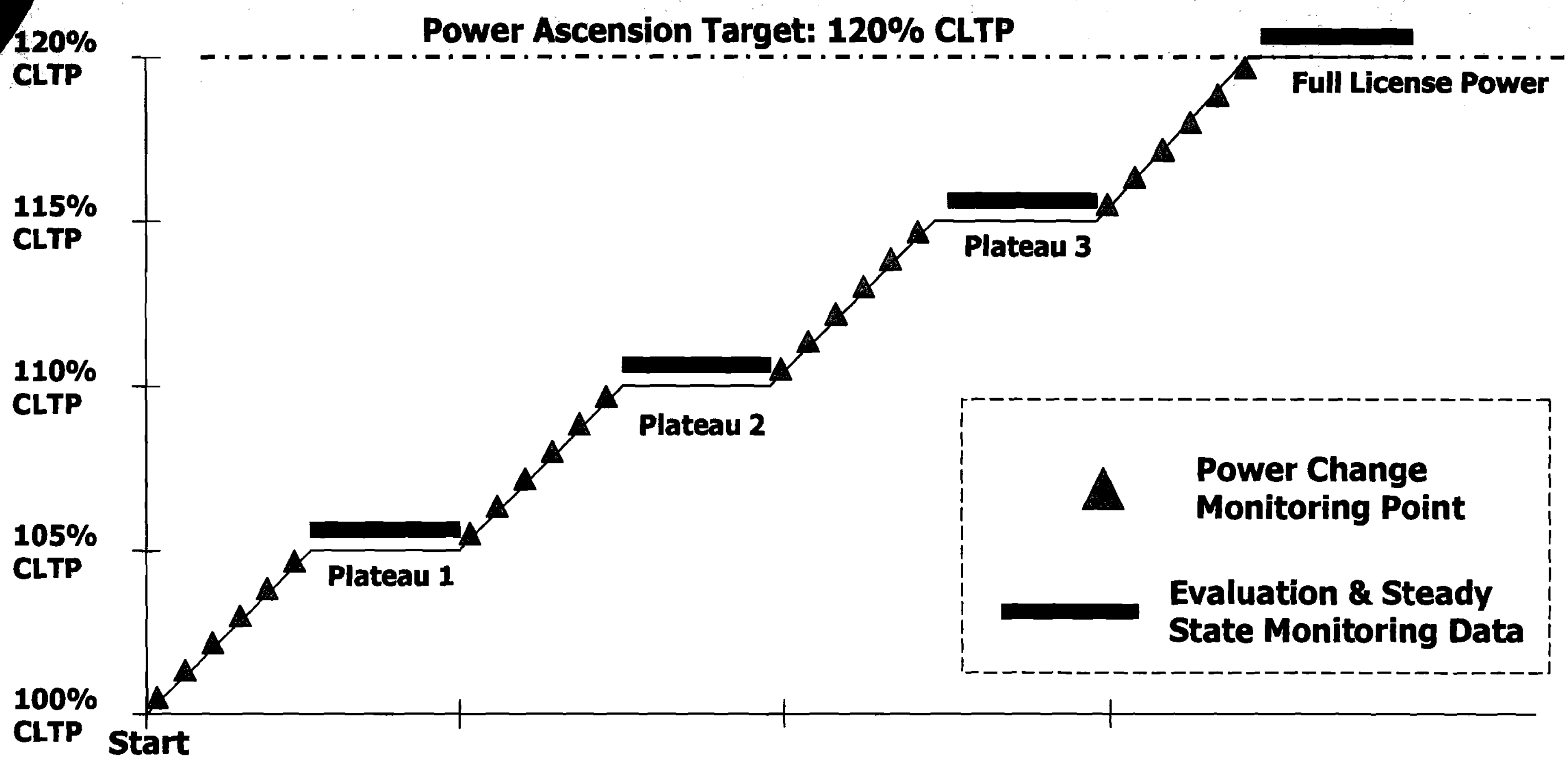


Entergy

Steam Dryer

Power Ascension Monitoring – Test Plateaus

Figure 3: Dryer Monitoring & Test Plateaus



Steam Dryer
Conclusions

- No Vulnerability at CLTP
- VY Measures Acoustic Loads
- Margin to Fatigue Stress Limit

3

ACRS Full Committee Meeting

NRC Staff Review of Proposed Extended Power Uprate
For
Vermont Yankee Nuclear Power Station

December 7, 2005

Risk Evaluation of Proposed Credit for Containment Accident Pressure

Martin A. Stutzke
Senior Reliability & Risk Analyst
Probabilistic Safety Assessment Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

In a Nutshell...

- Entergy has completed its risk evaluation of the proposed credit for containment accident pressure (CAP) to provide adequate net positive suction head (NPSH) to the emergency core cooling system (ECCS) pumps.
- Using realistic assumptions to estimate available NPSH, no CAP credit is necessary. Thus, granting the proposed CAP credit does not increase the risk associated with operation of Vermont Yankee (VY).
- The proposed CAP credit meets the five key principles of risk-informed decisionmaking.

Entergy's Risk Evaluation

- Chronology:

10/05/2005	Staff asks Entergy to provide risk evaluation that addresses the five key principles of risk-informed decisionmaking in RG 1.174
10/21/2005	Entergy provides partial risk evaluation (Supplement 38)
10/26/2005	Entergy completes risk evaluation (Supplement 39)
11/25/2005	Staff issues RAI about the risk evaluation
12/02/2005	Entergy responds to the staff RAI (Supplement 43)

- To address PRA modeling uncertainty introduced by the proposed CAP credit, Entergy performed a sensitivity analysis.

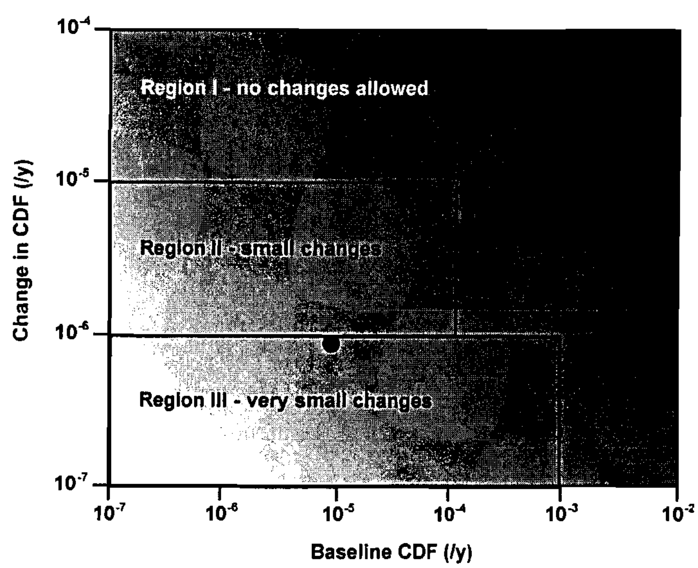
Entergy's Risk Evaluation (continued)

- Differences between the Entergy and staff sensitivity analyses:

	Entergy	Staff
Credit for alternative injection (AI) sources after loss of LPCI and CS pumps due to loss of containment integrity	yes	no
Note: No credit for AI after large LOCA for injection systems that take suction outside the suppression pool		
Credit for suppression pool cooling following loss of containment integrity	no	yes
Probability of pre-existing containment leakage		
Failure hole size	60 x La	35 x La
Failure probability	2.5E-04	1.4E-02
Data source	EPRI TR-1009325 December 2003	NEI Intermin Guidance 11/13/2001
Basis	expert elicitation	Bayesian (Jeffrey's non-information prior; no failures in 182 tests)
Change in CDF	5.8E-07	6.2E-08

Entergy's Risk Evaluation (continued)

Entergy's Sensitivity Analysis (internal events and internal floods; including proposed CAP credit and other EPU impacts)



Five Key Principles of Risk-Informed Decisionmaking in RG 1.174 and SRP 19

- The five key principles:
  - Proposed change meets the current regulations.
  - Proposed change is consistent with the defense-in-depth philosophy.
  - Proposed change maintains sufficient safety margins.
  - Increases in risk should be small and consistent with the intent of the Commission's Safety Goal Policy Statement (51 FR 30028).
  - Impact of proposed change should be monitored using performance measurement strategies.
- Acceptability of proposed change is determined by an integrated decisionmaking process.

Integrated Decisionmaking

- RG 1.174, Section 2.2.6 discusses integrated decisionmaking, and states that "None of the individual analyses is sufficient in and of itself."
- ACRS guidance:
  - ACRS letter of May 19, 1999 expressed concerns about the staff making "arbitrary appeals to defense in depth" to avoid making changes to regulations and regulatory practices that seemed appropriate in light of PRA results.
  - Joint ACNW/ACRS letter of May 25, 2000 discussed establishing limits of necessity and sufficiency on defense-in-depth within a risk-informed regulatory framework.

Defense-in-Depth Evaluation

- The proposed CAP credit is consistent with the defense-in-depth philosophy because it meets the four defense-in-depth objectives stated in SRP 19:
  - Does not result in a significant increase in the existing challenges to the integrity of barriers.
  - Does not significantly change the failure probability of any individual barrier.
  - Does not introduce new or additional failure dependencies among barriers that significantly increase the likelihood of failure compared to existing conditions.
  - Overall redundancy and diversity among barriers is sufficient to ensure compatibility with risk acceptance guidelines.

Defense-in-Depth Evaluation (continued)

- The proposed CAP credit does not affect normal plant operating conditions. So, no impact on:
  - Frequency of any initiating event, or
  - Probability of pre-existing containment leakage.
- Using realistic assumptions to estimate available NPSH, no CAP credit is necessary. Therefore, the proposed CAP credit does not:
  - Change the failure probability of the fuel barrier,
  - Increase the risk of VY operations, or
  - Significantly change the existing balance between accident prevention and mitigation.

Defense-in-Depth Evaluation (continued)

- Even if the CAP credit is assumed to change in PRA success criteria, then:
  - There must be at least four failures to cause a core-damage accident (LOCA followed by loss of containment integrity, suppression pool cooling, and alternative injection sources).
  - The change in CDF is very small and meets the RG 1.174 risk acceptance guidelines.
  - Results are robust in terms of uncertainties and sensitivities to key modeling parameters and assumptions.
  - No significant change in conditional containment failure probability (CCFP).

Performance Monitoring

- Diverse methods for detecting containment leakage:
  - Drywell/torus air space differential pressure > 1.7 psi - control room alarm; measured once per shift.
  - Low torus air space pressure - continuous.
  - Unusual nitrogen makeup - measured daily.
  - Oxygen concentration ≥ 4% - weekly measurements.
  - Integrated leak rate tests (ILRT) - Type A test done once every 15 years (temporary change).
- The fraction of time that the plant would be operated with a containment leak is small:
  - Leaks will be promptly detected, and
  - TS preclude prolonged operation with known leaks.
- Leak detection not explicitly considered in PRA.

Conclusions

- Entergy has completed its risk evaluation of the proposed credit for CAP to provide adequate NPSH to the ECCS pumps.
- Using realistic assumptions to estimate available NPSH, no CAP credit is necessary. Thus, granting the proposed CAP credit does not increase the risk associated with operation of VY.
- The proposed CAP credit meets the five key principles of risk-informed decisionmaking.

Deterministic Evaluation of Proposed Credit for Containment Accident Pressure

Richard Lobel
Senior Reactor Systems Engineer
Probabilistic Safety Assessment Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Purpose

- To discuss NRC staff review of Entergy's proposal to credit containment accident pressure in determining available net positive suction head (NPSH) for emergency core cooling system (ECCS) pumps for certain Vermont Yankee (VY) design basis accidents (DBAs).

Conclusions of NRC Review

- Need for crediting containment accident pressure for VY arises from conservative nature of design basis analyses.
- A more realistic, but still conservative, calculation would show that credit is not needed.
- A single failure resulting in loss of containment integrity will not result in a loss of NPSH margin.
- Credit for containment accident pressure has no impact on the operators.
- NRC staff finds proposed crediting of containment accident pressure for VY to be acceptable.

Regulatory Guide (RG) 1.82

- RG 1.82 is currently being revised by the NRC staff to address ACRS concerns.
- Entergy has stated as part of VY extended power uprate (EPU) submittals that it does not intend to make RG 1.82 part of the VY licensing basis.
- Methods and solutions different from those set out in RGs will be acceptable to the staff if they provide a basis for the requisite safety findings.
- Bottom line - all unresolved issues regarding revisions to RG 1.82 do not need to be resolved to find VY proposal acceptable.

Regulations

- There is no regulation prohibiting credit for containment accident pressure in determing available NPSH for safety-related pumps.

Design Basis Accidents

- Boiling Water Reactor (BWR) DBAs currently credit containment integrity and containment accident pressure for other considerations:
  - Radiological dose
  - Effectiveness of core spray cooling

Single Failure Considerations

Single Failure	Peak Suppression Pool Temp
RHR Heat Exchanger	195 F
Containment*	169 F

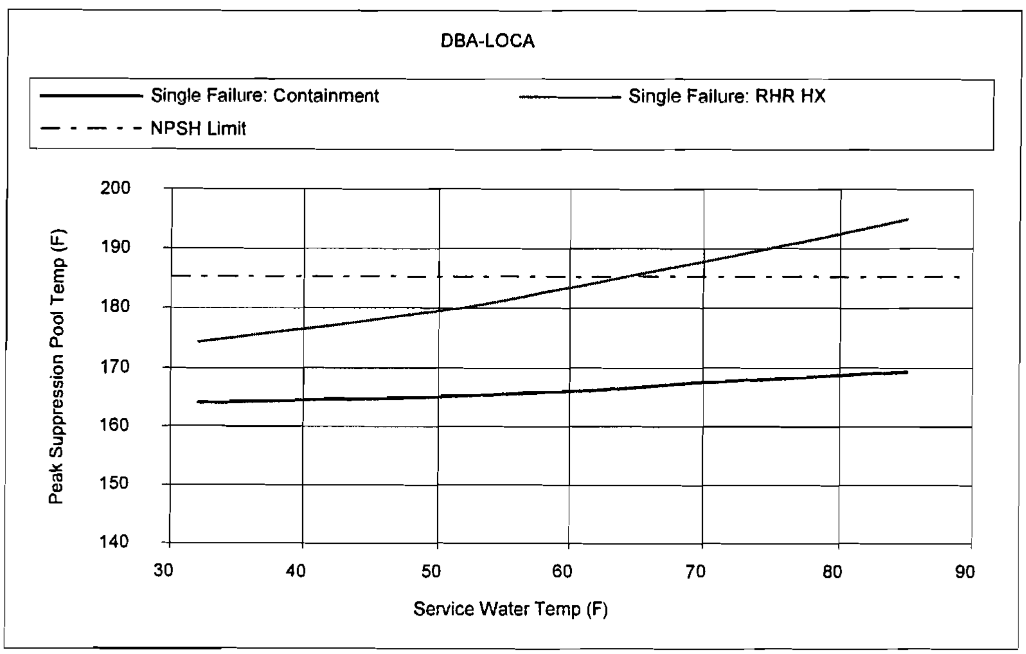
*If it is assumed that a single failure causes a loss of containment integrity, both RHR heat exchangers would be available and peak suppression pool temperature would be 169 F.

Credit for containment accident pressure is not needed if the suppression pool temperature is less than 185 F.

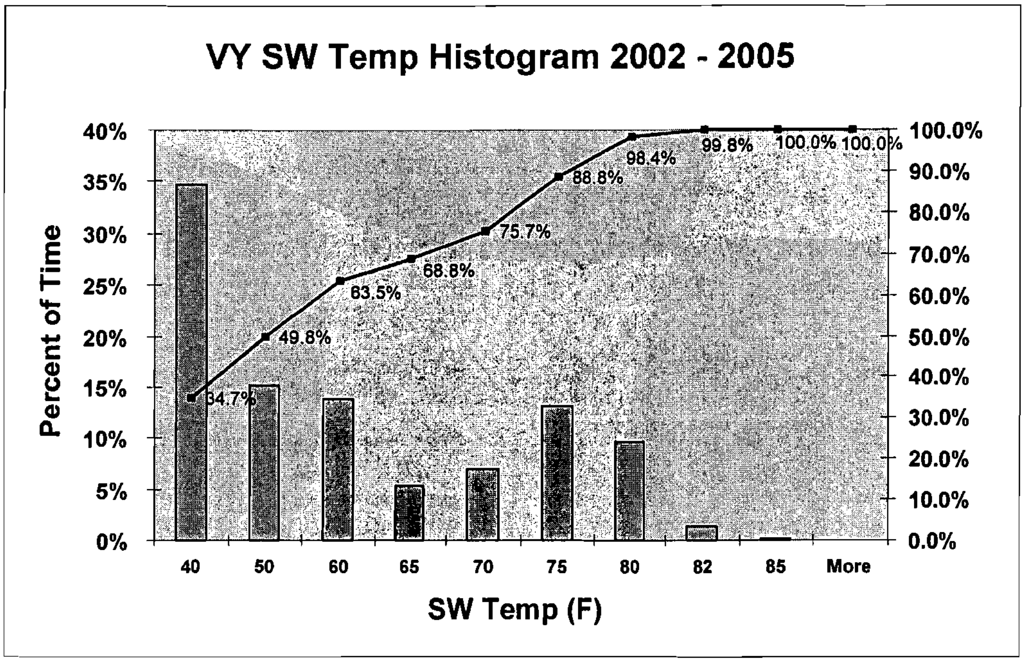
Defense-in-Depth

- Because the need to credit containment accident pressure for NPSH arises from the conservatisms in the calculations, eliminating excess conservatism eliminates the need to credit containment accident pressure.
- Dependence between barriers has been raised as an issue, however for VY based on the way calculation is done, there is no realistic physical dependence between barriers.
- Therefore, NRC staff considers defense-in-depth is maintained.

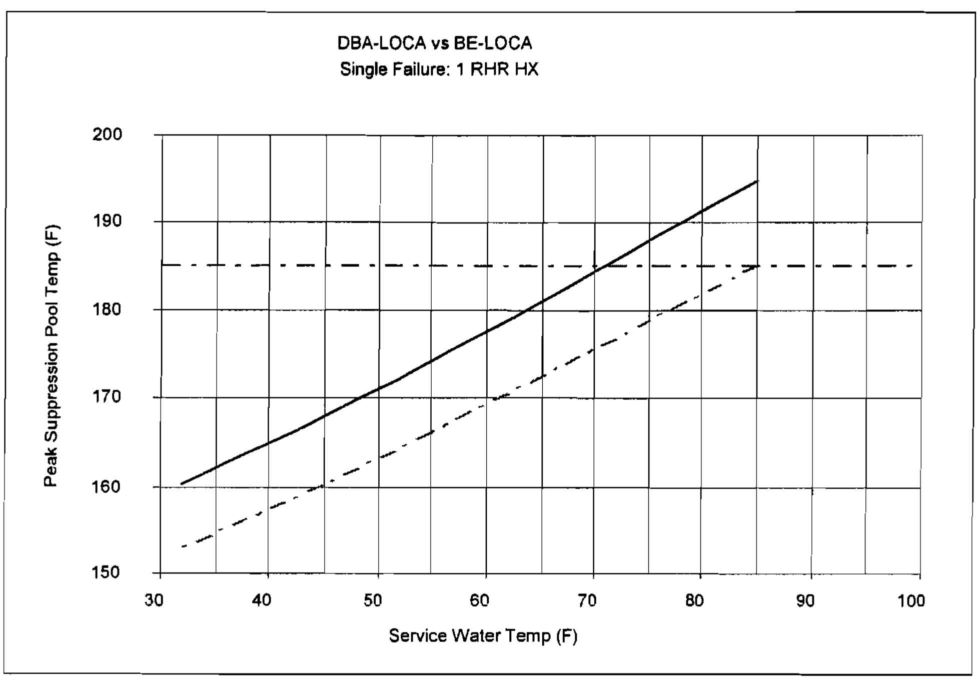
VY Sensitivity Study



VY Sensitivity Study (continued)



VY Sensitivity Study (continued)



Summary

- NRC staff finds that credit for containment accident pressure for VY is acceptable and is based on conservative calculations:
  - These calculations result in the need to credit containment accident pressure
  - A more realistic, but still conservative, calculation would show that credit is not needed
- Single failure resulting in a loss of containment integrity will not result in a loss of NPSH margin.

(4)

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
529	2	1

Title Vermont Yankee Power Uprate

List of Documents	
1. Draft Summaries (non-proprietary/proprietary sections) of the 11/29-30/05 Power Uprate Subc. Mtg. 2. VY Response to staff RIAs 3. Ltr report by G. Leitch 4. Ltr reports by S. Bannerjee 5. Ltr/emails/comments from members of public re: VY Power uprate	2

Instructions to Preparer	From Staff Person
1. Paginate Attachments 2. Punch holes 3. Place Copy in file box	Ralph Caruso

December 6, 2005

Note for: ACRS Members:

From: Ralph Caruso

Subject: Vermont Yankee Power Uprate

Attached please find the following supplemental documents related to the VY EPU, which were not available in time to be included in the meeting notebooks.

1. Draft summaries(non proprietary and proprietary sections) of the November 29-30 Power Uprate Subcommittee meeting

2. VY response to staff RIAs concerning the PRA analyses supporting the containment overpressure credit issue.

3. Letter report by consultant Graham Leitch.

4. Letter reports from consultant Sanjoy Bannerjee.

5. Letters/emails/comments from members of the public regarding the VY power uprate

I also have copies of the handouts from the November 29-30 meeting, for members who did not attend, in both paper and electronic formats.

cc: J. Larkins

Issued:11/23/2005
Certified:??/??/2005

WORKING DRAFT

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
POWER UPRATE SUBCOMMITTEE MEETING MINUTES
NOVEMBER 29-30, 2005
ROCKVILLE, MARYLAND

Introduction

The ACRS Subcommittee on Power Uprates held a meeting on November 29-30, 20055, with NRC staff members and representatives of Entergy Operations Northeast. The purpose of this meeting was for the Subcommittee to discuss the application by Entergy for a 20% Extended Power Uprate for the Vermont Yankee Nuclear Power Station. The meeting was convened at 8:30 a.m., November 29, 2005, and adjourned at 5:00 p.m. on November 30, 2005.

Attendees

ACRS Members/Staff	**NRC Staff/Consultants**	**Entergy**
Richard Denning (Chairman)	C. Holden	J. Thayer
Graham Wallis (Member)	R. Ennis	C. Nichols
Thomas S. Kress (Member)	T. Scarborough	B. Hobbs
Victor H. Ransom (Member)	C. Boyd	
Jack Sieber (Member)	S. Hambric(PSU)	
Sanjoy Bannerjee (Consultant)	M. Stutzke	
Graham Leitch (Consultant)	T. Mulcahy (ANL)	
	V, Shah (ANL)	State of Vermont
	J. Wu	W. Sherman
	M. Razzaque	
Ralph Caruso (DFO)	Z. Abdullahi	Raymond Shadis
	B. Elliott	
	R. Davis	

The presentation slides and handouts used during the meeting are attached to the Office Copy of these Minutes. The presentations to the Subcommittee are summarized below. There were several requests by members of the public to make written or oral statements. The oral statements are summarized in this record, and the written comments are attached to the Office Copy of these Minutes.

A portion of this meeting was closed to discuss proprietary information, and the minutes of that portion are provided separately.

Introduction

Dr. Denning opened the meeting and welcomed the participants. He noted that this meeting was being transmitted to listeners to a conference call line that had been established, and he hoped that everyone would speak clearly and identify themselves so that the listenrs could better understand the discussion.

1. Opening Remarks (C. Holden - NRR) (open)

Mr. Holden thanked the ACRS for meeting with the staff and the licensee regarding this uprate, and he noted that this meeting would cover a number of issues that were not discussed on Nov 15-16, in Vermont. He recalled that at that meeting, the staff was asked when they would be revising their SER to include consideration of additional information to be provided by Entergy. He recounted the proposal that was made by Dr. Sheron at the November 2005 full committee meeting, and noted that the licensee has made several submittals, the staff has issued RAIs, and expects to be able to talk about its evaluation at the December 7, 2005 full Committee meeting.

Dr. Denning asked whether it appeared that the staff is risk-informing a part of the power uprate review process. Mr. Ennis replied that the staff does not intend to risk-inform the entire uprate process, but intends to use it selectively, in accordance with the guidance in RS-001.

2. Introduction (R. Ennis - NRR) (Open)

Mr. Ennis provided an overview of the presentations to come. He described the information that will be provided, and noted that the staff does not intend to discuss containment overpressure issues during this session, but intends to hold that information until the Full Committee meeting on December 7, 2005.

3. Steam Dryer and Vessel Internals (C. Nichols,B. Hobbs, E. Beaty - Entergy) (Open portion)

Mr. Craig Nichols opened the presentation by thanking the Committee for providing the opportunity to discuss the VY uprate, and he described the overall plan for presentations today. He introduced Mr. Brian Hobbs and Mr. Enrico Beaty.

Mr. Hobbs described the operating history and the results of inspections of the steam dryers, and the results of analyses of the steam dryer response to acoustic loads. They believe that acoustic loads are the promary source of dryer degradation. It is important to monitor for acoustic loads, and acoustic circuit methodologies can be used to project dryer loads from steam line signals. They also believe that EPUs can exacerbate existing FIV vulnerabilities.

In response to a quesiton from Mr. Leitch, he reported that the main steam line steam flow at VY in the EPU condition will be roughly equal to the steam velocity at Quad Cities before their uprate. Entergy has been intensively involved in industry benchmarking and analysis efforts. He described the physical configuration of the dryers. He noted that the dryer is not safety-related, but is designed to perform without generaiton of significant loose parts.

The dryer was inspected in the spring of 2004 and the fall of 2005, and several indications have been repaired and modified. The indications that were found were determined to be caused by IGSCC. No misssing parts were found. Mr. Sieber asked about the extent of the inspection. Mr. Hobbs explained that it was an enhanced VT1 inspection. Mr. Sieber what sort of rack characterization was performed, and Mr. Hobbs described two indications that were ground out. Mr. Sieber asked why 2 indications at the steam dams were reparied, and Mr. Hobbs replied that they believe they may have been caused by high stresses, and may have been related to initial fabrication, so they decided to repair them. These inspections were consistent with the industry-standard processes for inspecting BWR vessel internals. Mr. Sieber thought that it is very important to first characterize flaws, before their future growth can be analyzed.

Mr. Hobbs described the dryer strengthening modificaitons that were made. The vertical hood, top hood, cover plate, and ties bars were strengthened, and new gussets were added. He showed photos of the modifications in place. Dr. Bannerjee asked whether this strategy has been shown to be valid, and Mr. Hobbs replied that it has worked in other plants. Dr. Bannerjee noted that if the source of the vibration is not removed, then the cracking will just move elsewhere. Mr. Beatty explained that these mods are intended to remove low frequency vibration modes by changing the natural frequency of the hood. Mr. Beatty commented that the results with this mod have been mixed, depending on the exitation frequencies. If there is another higher excitation frequency, then other cracks will appear. Mr. Hobbs noted that this mod was made at Dresden, and subsequent inspection showed that high-frequency failures had occurred. However, at Brunswick, the loads are lower, and the fix has been successful.

Mr. Beatty described how the gussets reduce the vibration, and how increasing the thickness of the plates makes them stiffer. Dr. Ransom commented that it would be best to identify the fundamental modes of vibration, so that the behavior could be understood better, and the stiffening could be focused. Dr. Bannerjee commented that the addition of the gussets creates additional vortex shedding, and therefore additional vibration. Mr. Hobbs replied that this has been considered, and will be discussed later in the presentation. Mr. Sieber commented on the difficulty of analyzing this ocmplex geometry, and Mr. Hobbs replied that as a result, they have instituted a complex monitoring program to detect the occurrence of vibration in the future.

Mr. Hobbs described the program that they have established to monitor vibration in the dryer, by measuring main steam line vibrations. Mr. Leitch asked how they dealt with vibrations from the MS control valves that are reflected back into the steam lines, and Mr. Beatty commented that this was investigated extensively by the entire industry. The solution that was agreed on was to measure the vibration as close to the reactor vessel as possible, and the methodology is able to deal with resonances that arise from downstream components. Dr. Bannerjee asked about instrumenting the dryer itself, and Mr. Hobbs explained that the new QC dryer has been instrumented, but installing sensors on an old dryer is a high-dose effort.

Mr. Hobbs presented the results of the strain gage measurements at VY compared to the gages at QC, and pointed that the QC vibrations have a much higher peak at the resonance values than VY, both at pre-EPU and post-EPU conditions. Dr. Bannerjee asked how these measurements compared to the dryer measurements, and Mr. Hobbs replied that they are correlated.

Mr. Hobbs also described evaluations that were performed to consider MSL branch line acoustic resonances. They believe that if the EPU does cause these lines to resonate, the MSL strain gages will detect it. He presented the results of the analyses with the expected resonances, and noted that they should be detecting resonance from the relief valves lines, but they are not detecting them. After EPU, they believe that they may be able to detect resonances from both the relief and safety valve branch lines.

(Closed Portion - see proprietary summary)

4. Mechanical and Civil Engineering (T. Scarborough, NRR) (Closed)

(See Proprietary Summary)

** LUNCH ** 12:40 - 1:30 pm

5. GE Methods and Reactor Issues (J. Head - Entergy, F. Bolger - GE) (Open Session)

Mr. Head described the nuclear analytical methods that were used to support to the VY power uprate. He began by recounting the Constant Pressure Power Uprate (CPPU) methodology that was developed by GE. It was approved by the staff in 2003, after review by the ACRS. He noted that it is possible to implement a power uprate in a BWR by increasing the batch fraction, or by increasing the enrichment, and for the VY uprate, they are using a combination of these techniques. Mr. Sieber commented that this increases neutron fluence on the vessel, and Mr. Head agreed that both the vessel and internals see a higher fluence.

Mr. Head recalled that the NRC was in the process of reviewing an expansion of the BWR operating domain while it was reviewing the VY EPU, andas a result, the two reviews became intertwined. The staff eventually performed an extensive review of the computer codes and methods that were used to establish the operating limits for VY at EPU conditions. As a result of these reviews, VY proposed an "alternative" approach to resolve staff concerns about the methods which involved increasing the cycle-specific SLMCPR by 0.02. Mr. Head described the relationship of the SLMCPR, the OLMCPR, the LHGR, peak pellet exposure, and the MAPLHGR.

Mr. Head described how the MCPR was originally determined and verified, and he explained that these values were thought to be relatively insensitive to the fuel lattice, but there are staff questions about whether they need to be re-validated using gamma scans of exposed fuel. He explained how OLMCPR is derived from the SLMCPR, and how the peak pellet exposure relates to the LHGR limit. Dr. Bannerjee asked about the computer codes that are used, and Mr. Bolger described the GEXL correlation and the GESTR fuel mechanical model.

Dr. Denning noted that one important part of the uprate process is the flattening of the overall reactor power shape, which would seem to have an effect because a large number of rods are coming to a higher temperature than before. Mr. Bolger replied that yes, the peak, limiting bundle does not change, but the rest of the core come up closer to the peak bundle in its performance.

Mr. Head described the design process related to core shutdown margin, and he noted that this is particularly important, but the VY cores have been robust, and they have plenty of shutdown margin. With regard to core stability, he described the core stability solution for VY, and he explained how the exclusion region and the detect&suppress methods work to ensure that instabilities will not occur. Dr. Bannerjee asked about the stability analyses, and GE explained how the ODYSY code is used to calculate fuel performance during a stability event, and establish the exclusion and buffer regions. GE described several examples of BWRs that have recently experienced instability events, and in all of those cases, the automatic systems detected the instability and scrammed the plant.

Dr. Ransom asked about the effect of flattened power shape on CCFL flow down thru the low power bundles, and Mr. Pappone replied that the water that is held up will flow down through the peripheral bundles and make it into the lower plenum. The phenomenon is self-limiting.

Mr. Head also noted that they maintain their own core physics models that are used to independently evaluate the performance of their cores, and the designs produced by the fuel vendors. They believe that they can detect problems before the vendors can, because of the continuous nature of this monitoring. He described how they use TGBLA06 and CASMO-4 to verify core performance for expected transients and accidents.

Mr. Head noted that the CPPU methodology requires evaluation of several events on a cycle-specific basis, and he explained how the licensee had evaluated stability, overpressure protection during AOOs, and the ATWS analyses. He noted that the peak pressure remains below the ASME limit, and the suppression pool temperature remains well below the limit. Dr. Bannerjee asked about analysis uncertainties, and Mr. Bolger explained that ATWS calculations do not account for uncertainties, but are done using nominal values. He explained how TRACG and ODYN are used, to evaluate core and fuel performance during a pump trip with oscillations. Containment pressure and suppression pool temperature are also calculated. He noted that there is no need to consider ATWS instability

Closed session -- 2:29 pm
(See proprietary summary)

** Break - 2:40 - 3:15 pm**

6. <u>Reactor Systems (M. Razzaque, Z. Abdullahi - NRR) (Closed)</u>

(See Proprietary Summary)

RECESS 5:30 pm

The meeting reconvened on November 30, 2005 at 8:30 am, in open session.

Introductory remarks - R. Ennis (NRR)

Mr. Ennis opened the session by noting the interest in ECCS suction strainers and proposed dropping the electrical engineering and station blackout presentations, and substituting the sump strainer issue. The Committee agreed to this change in the agenda.

7. Flow-Accelerated Corrosion and P-T Limit Curves (J. Callaghan - Entergy)

Mr. Callaghan presented a description of the FAC program at VY, which is based on guidance supplied in GL 89-08 and EPRI NSAC-202L. Dr. Wallis asked about the emperical base for the correlation, and Mr. Callaghan explained that corrosion is a function of fluid velocity, and the correlation is emperical, based on inspection results. He explained that VY has seen very little wear in the piping at VY. The extraction steam system was originally installed with FAC-resistant material, and is original to the plant. The amount of wear is within the tolerance of the UT measurement capability. Some components that have been replaced include the feedwater heater shells, the LP turbine casings, moisture separator drains downstream of the level control valves, and the turbine cross-around lines.

Mr. Callaghan noted that no additional systems have been added to the FAC program due to the EPU. He described the CHECWORKS program and the inspection program that is used, and the most susceptible locations, where turbulence exists in the flow stream and the fluid velocities are highest. The piping around the FW reg valves are inspected and monitored most closely. The WF system is carbon steel and is not inherently FAC-resistent. Dr. Bannerjee asked about flow effects on cobalt alloy corrosion and transport of crud around the system. Mr. Nichols replied that they do not anticipate that the EPU will effect radiological transport at VY.

Mr. Callaghan described the revision to the PT curve that is required to accomodate the EPU. The neutron fluence at end of life increases by about 6% compared to current rated power, but is still bounded by the current TS limits. Mr. Beatty commented that improvements in GE fluence methodology has resulted in reductions in the calculated values, and they remain within design allowances.

8. Materials and Chemical Engineering (B. Elliot, R. Davis, K. Parczewski - NRR

Mr. Elliott described the RPV integrity surveillance program, and the effects of the EPU on Charpy upper-shelf energy at the expiration of the license at EPU conditions. He noted that VY is member of an integrated surveillance program for all BWRs, which has been approved by the staff for application to VY, and it will be using data from Susquehanna surveillance coupons. Mr. Sieber asked why the staff has not used data from VY capsules, and Mr. Elliott replied that the staff has data from VY to cross-correlate to the Susquehanna capsules, and it prefers to leave the VY capsules in place as a backup to the other capsules. He noted that because of the welding technique that was used to fabricate the VY vessel, there is very little copper in the welds, and it has very good resistance to embrittlement, and can accomodate a very high fluence.

He also discussed the implementation of the BWRVIP program at VY, and explained that the staff had identified deficiencies in the inspection program for the top guide and the steam dryer. The licensee has proposed to perform periodic enhanced visual testing of a sample of the top guide grid beans to detect IASCC, and will perform additional inspections of the steam dryer, as was discussed previously.

Mr. Davis discussed the effects of the EPU on reactor coolant pressure boundary materials, loading, and performance. He noted that the flow, pressure, temperatre and mechanical loading for most RCPB piping systems do not increase for the EPU, and the plant-specific evaluation is consistent with ELTR1. VY has replaced the recirc system piping with 316L, and this meets the requirements of NUREG 0313. He noted that the MS and FW systems will experience an increase in flow, and they were evaluated for compliance with the Code of construction for EPU conditions, and found to be acceptable.

Mr. Parczewski discussed the effects of the EPU on protective coatings in the plant. He noted that there was some concern about coating failures during a LOCA, and the licensee considered failures of inorganic zinc with an epoxy top coat, and other carbon-based paint chips. The licensee determined that 85 pounds of coatings could be stripped by the post-LOCA jet, and this is consistent with the guidance in NEDO-32686. Based on testing at Alden labs, there is no change in the transport of paint chips to the screens.

Dr. Wallis and Dr. Bannerjee expressed some skepticism about the assumptions used in this analysis, asking why it was assumed that the pool is quiescent and the paint not stirred up when a suction is taken on the pool. Dr. Bannerjee also asked where the sludge in the pool comes from. Mr. Nichols commented that Entergy will be addressing this issue later today.

Mr. Parczewski described the staff review of the VY FAC program, including application of CHECWORKS. He noted that the only significant change in plant operation related tot he RWCS as a result of the EPU will be more frequent backwash of the filter demineralizer because of the increase in FW flow.

Dr. Bannerjee asked whether the EPU affects the energy in the LOCA break jet, and GE replied that the break stream does not change, because the analyzed break locations do not change, and the break flow chokes, so the jet may last longer, but will be the same magnitude.

Dr. Wallis asked about how the steam separators would maintain integrity during EPU, and he asked what the basis was for this conclusion. Mr. Dick (GE) replied that the VY separators were instrumented both at VY and at Monticello, and the predicted stress levels on the separators at VY are very low, and the prototype separators were tested at much higher flow rates than will be experienced by the VY separators at EPU conditions. Mr. Leitch asked whether VY was using hydrogen water chemistry, and Mr. Dick replied that VY has both hydrogen injection and noble metal chemistry control. The hydrogen injection rate will increase by about 22%, but VY is considered to be a "low"-hydrogen injection rate plant, and easily has the capability to accommodate the increase.

9. Station Blackout and Grid Stability (C. Nichols - Entergy)

Mr. Nichols described the SBO analyses and coping evaluation that was performed to support the EPU. SBO at VY would require loss of offsite power to the switchyard, loss of both EDGs, and a loss of the Vernon Tie AAC that would require a restart due to a regional blackout. The plant has a 2 hour coping duration for restoration of AC power from the Vernon Hydro Station. There is sufficient CST inventory, battery capacity, and heat sink capacity within the suppression pool. There is also sufficient ventilation and control air(N2) available. Dr. Denning asked what capacity would be available once the Vernon tie becomes available, and Mr. Johnson replied that they would regain one full safety train with the Vernon tie, and that can remove decay heat indefinitely. Also, this analysis starts at nominal power, not 102%.

Mr. Nichols presented the timeline for the SBO, and the notification of the necessary operators to restart the Vernon facility. Mr. Johnson noted that they believe that they have about 20% more capacity than they need to meet the SBO coping scenario. Once they recover AC power, they can continue to remove decay heat and depressurize within 8 hours from the start of the SBO event.

**Recess ** 10:00 - 10:15 am

10. Electrical Engineering (N. Trehan - NRR)

Not Presented.

11. Operations, Training, EOPs, Operator Actions, Timelines (C. Tabone, C. Wamser - Entergy)

Mr. Wamser described the operator training program, the changes to the operating procedures, and other operator-related actions that have been taken to support the EPU. He noted that the only significant operational impact is the need to run all 3 FW pumps at full power, and the need for additional rod pattern adjustments. There is a slight reduction in operator action times for certain events, but the balance of plant modifications generally improve plant performance and component reliability.

Mr. Leitch asked whether the operators have been trained in the processes associated with rod pattern changes, and Mr. Weiss discussed the training program for the operators and the reactor engineers who establish the need for core configuration changes. Mr. Leitch commented that under EPU, more bundles are operating closer to licensing limits, and he wondered whether this addition to the complexity of the workload for the reactor engineers might effect their performance. Mr. Wamser replied that the EPU itself did not add to the reactor engineer job tasks, because they perform the same tasks. Operators are trained to respond to events that require core configuration changes. Mr. Head explained that while it is possible to increase operating complexity, VY has taken the tack of trying to design the VY EPU core to include as much margin as possible, so as to reduce the stress level on operators and engineers. They verify GE's methods with their own calculations, and they don't want to add unnecessary complexity. At the end of core life, rod pattern exchanges increase so that they might occur every few weeks, in order to maintain reactor power.

Mr. Sieber asked what VY used to set the end of a cycle. Mr. Head replied that cycle length is determined mainly by target refueling dates, because they are operating a large fleet.

The operators at VY have been undergoing training for the past 2 years to support the EPU, and all crews have completed the training program. Additional training to deal with the power ascension program will be held prior to implementing the EPU.

Some abnormal operating procedures have been changed, but there are no new emergency procedure actions or strategies, and only minor revisions to the EPG graphs. The time available to perform some operator actions has been reduced because of the EPU, but they believe that they ahve demonstrated that the operators can continue to perform these actions within the allowable time windows. This assurance comes from testing in the simulators and walkthroughs in the plant.

Mr. Sieber asked about operator response to ATWS, and Mr. Wamser replied that the training program at VY requires operators to initiate SLCS immediately during an ATWS, and they are not supposed to wait until they observe oscillations to proceed. At VY, the key for the SLCS injection switch is maintained in the switch - it is not stored away from the control panel.

12. Human Performance (J. Bongarra - NRR)

Mr. Bongarra described the staff review procedure for human performance issues related tot he VY EPU. The staff considered the programs, procedures, traning and human system interface design features that are related to operator performance, in order to assure that operator performance is not adversely effected by the proposed EPU. He described the review process, based on criteria in RS-001, 10CFR 50.120, GL 82-33, and SRP 18.0.

Dr. Denning asked him to focus on the reductions in the amount of time available to the operators, and Mr. Bongarra explained that the staff considered the licensee evaluations, and determined that most of the tasks are not time-sensitive. In one case, the actions needed to be completed in 40 minutes, so adequate time continued to be available. In one case, the available time to completion was reduced from 6.2 minutes to 5.4 minutes, but the operators routinely complete the action in about 1.5 minutes. Dr. Wallis asked how long it takes for the operators to evaluate the situation so that they can determine that the action needs to be taken. Mr.Tabone replied that these times include diagnostic time. Mr. Bongarra commented that the events are not new, but just the amount of time that is available for the operators to respond. Mr. Sieber asked VY to describe operator actions during an ATWS.

Mr. Wamser described the specific steps that operators are expected to take, including the instruments that need to be checked and cross-checked. He described the manual scram, alternate rod insertion and recirc pump trip actions, inhibition of ADS, and then SLCS initiation actions that operators would take during an ATWS. The members then discussed the decision trees associated with operator emergency response. They asked about the probability of operators making incorrect decision, and Mr. Wamser explained that these actions are considered in PRAs, but operator errors do not necessarily lead to core damage. BWR operator actions are very symptom oriented, to try to guide the operators back to success paths that mitigate the event.

Mr. Bongarra summarized the staff finding that the olicensee has accounted for the effects of the proposed EPU on operator actions, and operator performance will not be adversely affected by the proposed EPU. Dr. Wallis noted that he had seen a GE document that listed operator failure probabilities for various actions and how they change as a result of EPU. He noted that some failure rates are significant, around 73%, and he wondered how the staff takes these failure rates into account. The staff agreed to discuss these questions during the PRA discussion.

13. Plant Systems (D. Reddy - NRR)

Mr. Reddy described the scope of review for the BOP portion of the plant, and the issues that were considered. The staff focused its efforts on the spent fuel pool, service water system/ultimate heat sink, aux cooling water, and the condensate and feedwater system. These systems have, in the past, been the ones that have been most affected by EPU proposals.

The staff confirmed that the SFP temperature will continue to be maintained for a batch fuel offload or a full core offload, following EPU implementation. Mr. Jones explained that the licensee is crediting the RHR system for some heat removal functions during the first few days following shutdown to meet the SFP cooling criteria.

Mr. Reddy reported that the SWS capability is sufficient for EPU conditions, and the ACS is capable of providing cooling for seven days if there is a failure of the Vernon Dam. During such an event, the cooling tower and the deep basin would serve as a heat sink. Mr. Leitch asked whether the staff had verified the calculations performed by the licensee, and Mr. Reddy replied that they had not - they just reviewed the results presented.

The staff considered the capacity of the condensate and FW system, but it identified concerns about the operation of all 3 FW pumps, and the effect of a loss of a condensate pump on FW operation. As a result, the licensee has modified the FW pump trip logic to respond to a loss of a condensate pump, and will perform a test to verify the performance of the system as intended. Mr. Leitch asked what was the safety concern associated with the condensate pump trip. Mr. Jones replied that the concern is that they don't want to unnecessarily challenge safety systems due to a condensate pump trip. Mr. Leitch asked what would happen if the reactor scrammed during this test. Mr. Jones replied that the staff would expect that the licensee would make additional modifications to be able to withstand this transient. Mr. Nichols commented that the committment is that the loss of condensate pump does not cause a loss of all FW, and the licensee will modify setpoints as necessary to achieve this goal.

Mr. Reddy described the fire protection review, and noted that the acceptance criteria come from 10 CFR 50.48, Appendix R, and the draft GDC-3 "Fire Protection". Specific rview criteria come from RS-001 and SRP 9.5.1, and intend to assure that fire will not prevent the performance of necessary plant safety functions and will not significantly increase the risk of radioactive releases.

He described the evaulation that the licensee performed, and the basis for the staff approval of that analysis. Dr. Denning commented that it is not obvious how EPU can affect fire protection analyses. Mr. Galuci replied that these analyses are plant specific, but the only general change is to required operator response times.

Dr. Bannerjee noted that a member of the public had commented about cable-try separation issues, and Mr. Ennis replied that he knows of no cable separation issues that are outstanding for VY, and

**LUNCH ** 11:45 - 12:45 pm

14. <u>RHR and Core Spray Suction Strainers (B. Hobbs - Entergy)</u>

Dr. Denning opened this session by noting that this discussionis not intended to review the current licensing basis, but to consider how it fits into the EPU proposal.

Mr. Hobbs opened the discussion of the ECCS suction strainers that are installed at VY. The strainers are some of the largest installed in a BWR, and they believe that they fully support the EPU operating conditions.

Mr. Beatty described the VY torus and how the downcomers and the strainers fit into the torus. He noted that the modules were sized to be as large as could be possibly fit into the torus. He described how the holes in the stacked discs and the internal structure of the strainer are sized to try to load the discs uniformly during an event. He presented the design approach velocities and surface areas that were used in the evaluation of the debris head loss. Dr. Bannerjee asked how the approach velocity that was presented related to the approach velocity in the bulk of the torus, approaching the strainer. Mr. Beatty replied that they had closely observed the operation of the strainer in a test, which showed that the particles had to come up to the face of the plate to stick to it, and the "total-area" approach velocity is appropriate. Dr. Banerjee was also concerned because the test just looked at the performance of s single disc, while the stack discs will behave differently. Mr. Beatty replied that EPRI had performed tests on stacks of discs to validate their performance, and VY considered this testing in developing how its strainers would react to paint chips.

Dr. Bannerjee asked if it would be possible for the Committee to see the EPRI data that was used to support this design. He thought that the validation of the Los Alamos correlations had been called into questions, and it was not clear how the testing had been performed. Dr. Wallis expressed concern about how the LANL correlation had been applied to the strainer, and how they had factored in the "thin-bed" effect. Mr. Beatty replied that tests of these strainers showed that te debris does not deposit uniformly, but instead in patches, so there are large areas that are open. They have since resisted refinements to the head loss correlations, and still use the data from the testing that was done on the strainers.

Mr. Beatty described the debris loading assumed for the VY strainers and tested at Alden Labs, and Dr. Bannerjee asked if the design analysis could be provided to the members. VY report 1924

Mr. Beatty reported that there were originally concerns about the qualification of the containment coatings, so the coatings were subsequently inspected and tested, and the torus was re-coated below the water level with a qualified coating. He also noted that in 2004, they removed 75 pounds of sludge, which is far below the value assumed in the analysis.

Mr. Sliefer discussed the formation of vortices and air entrainment in the suction strainers, and he began with testing that was documented in NUREG/CR-2772. These strainers were short, comparted to current designs, and no air-core vortexing was observed. Testing with more typical stacked disc prototypes, with the top of the strainer exposed, did not reveal any vortexing. Dr. Wallis commented that there seem to be a serious of reports that discuss this issue, but it is not clear what the final design numbers are that wer eused in the design, and how the strainer design took into account the various design parameters. It was not clear from reading the reports what was actually done. Mr. Beatty replied that the people who had participated in the process understand what was done, but it is not gathered together in one place that is transparent to someone new.

Dr. Bannerjee was not persuaded that the had appropriately considered the paint chips, because he thought that the turbulence in the torus would continue for a considerable period of time. Mr. Beatty replied that they had tested paint transport during times of moderate turbulence, and most of the paint settled. When the turbulence is high, the debris does not stick to the strainers.

Dr. Denning asked what the head loss through the screens is, compared to the available containment pressure. Mr. Beatty replied that the bounding head loss across the strainers is about 0.5 ft of head, based on the experimental results from the tests. Dr. Ransom asked whether VY had made any attempts to reduce debris sources, and Mr. Beatty relied that they have replaced tempmat with RMI, and have replaced unqualified paint. NUKON and sludge, together, remain the primary debris source. They try to keep the containment clean, and they think that the small amounts of debris that have been found in the torus show that the cleanliness programs are working.

15. Source Terms and Radiological Consequences (M. Hart - NRR)

Ms. Hart described the source term/radiological analyses that the staff considered. She used RS-001 as the review standard, and she noted that the licensee analyzed the radiation sources in the reactor coolant for operation under EPU conditions. In a separate licensing action, VY submitted a request for alternate source term, and the staff approved AST for VY on March 29, 2005. Ms. Hart described the changes that were made to the analysis methodology as part of the AST. VY will be using the SLCS tor suppression pool pH control. Mr. Perez explained that the SLCS will be activated when radiation levels in the containment reach a certain amount, which is what is expected from a DBA gap release. Iodine will be removed by activation of the drywell sprays and they have taken credit for iodine deposition in the MS lines and the condenser during the LOCA.

16. Health Physics (R. Pedersen - NRR)

Mr. Pedersen described how the licensee had calculated radiation does to plant workers and members of the public as a result of the EPU. He noted that most of the radiological issue were addressed generically during the review of the CPPU topical report, and he considered some plant-specific issues related to occupational and public doses. The CPPU review looked at plant shielding designs. He verified that the radiation zoning in the plant continued to be valid, and he did verify that these zones did not change. They assume that radiation levels are generally proportional to reactor power.

Access to plant vital areas was also considered by the licensee, and they verified that those areas can still be accessed by plant workers, and therefore meet the TMI Action Plan II.B.2 criteria.

With regard to public dose limits, he verified that the dose to the public remains below the EPA limit of 25 mrem/year. The public dose is due to sky-shine from N-16 from the turbine/condenser. Before the EPU, the public dose was calcuated to be 15 mrem/yr, with 13.4 from N-16 shine, while the EPU value is 18.6 mrem/year with 16.9 from N-16 skyshine. He noted that these values come from the licensees current licensed method, which uses measurements that were correlated to the steam line rad monitor dose. There have been some questions raised by these calculations, and the staff is continuing its review of the methodology to resolve those questions. Mr. Sieber asked about operator access to equipment, and Mr. Pedersen explained that those questions are part of the II.B.2 review.

17. <u>Probablistic Safety Assessment (V. Andersen - Erin)</u>

Mr. Andersen presented the results of the Psa that was performed to support the EPU. Internal events were considered in Level 1 and 2 PSAs, while external events followed the IPEEE study, including internal fires, seismic, and other external hazzards. The PSA reflects the current plant configuration, and is maintained and routinely updated to reflect current plant configuration and operating experience. NEI conducted a peer review of the PSA model in 2000, and all category A&B facts and observations have been resolved.

After considering EPU impacts, the PSA did not identify any new accident sequences, and there was no significant impact on IE frequencies. Only one success criteria changed, and a slight decrease in the time available for some post-initiator operator actions were identified.

Dr. Wallis asked why the human error probabilities increased so much as a result of the EPU, and Mr. Andersen agreed that for actions that have short time frames, the error rate goes up faster than the time goes down. Dr. Wallis asked whether VY used the GE values for human errors, and Mr. Nichols thought that they were the ones that were used for VY. He discussed the large error probability associated with re-opening the MSIVs during an ATWS, but he noted that because the event is such a low frequency initiator, the error rate does not have a significant effect.

Mr. Andersen commented that events such as fires and seismic are not sensitive to changes in human error probabilities, and many industry studies have proven this out. The overall change in CDF as calculated by the PSA is a small increase of about 3E-7/yr, while LERF increases by about 1E-7/yr. There is no significant risk impact from external events and shutdown. Dr. Denning asked how the increase in the source term was accounted for. He noted that without performing a level 3 PSA, we ignore the increase in latent cancer risk associated with the uprate. He thought that CDF and LERF need to be considered carefully when they are applied to power urpates.

Dr. Kress asked if their PSA was capable of performing monte-carlo uncertainty studies, and Mr. Andresen said that it does. For the PSA that was done in support of the containment overpressure credit issue, many conservative assumptions were done to show large margins.

18. Risk Evaluation (M. Stutzke - NRR)

Mr. Stutzke began by pointing out that the VY EPU submittal is not risk-informed. He used the guidance in RS-001 to determine whether "special circumstances" exist in this application. He depends on the rest of the review process to ensure that the technical parts of the design complied with regulatory requirements, and used PRA to determine whether any unusual situation existed that would not be discovered by the normal deterministic review process. He noted that the VY PRA includes a full-power level 2 analyses, which is unusually thorough.

Mr. Stutzke reviewed the inputs and assumptions in the licensees PRA, and found that he generally agreed with their assumptions. The licensee used the MAAP code to determine success criteria and they added a spring safety valve, which affects the ATWS success criteria. He has sent VY a set of additional RAIs related tot he containment overpressure credit analyses, and he hopes to receive a response in a few days.

Mr. Stutzke considered the impact of the EPU on equipment reliability, and noted that the operating ranges and limits of equipment will not be exceeded, so no change will occur in the equipment reliability. However, he did note that the SORV probability has been adjusted to reflect the installation of an additional SSV, and the increase in the probability that a valve will chatter and fail to seat.

He discussed the effect of the EPU on human reliability, and on the ability of operators to perform actions in the time required. VY performed a more detailed HRA analysis, and it considered a wider variety of performance measures, than he was used to seeing from a licensee. The licensee also re-assessed the dependencies among associated operator actions. He determined that their assessment was performed in a reasonable fashion, and modeled properly.

Mr. Stutzke explained how the CDF and LERF changed as a result of the EPU, and he noted that the changes are not generally significant. No new seismic vulnerabilities were identifies, and nothing in the EPU changed the plant's ability to resist a seismic event.

Overall, the licensee has adequately modeled the potential risk impacts of the proposed EPU, and the EPU does not create any "special circumstances" that rebut the presumption of adequate protection provided by the licensee meeting current regulatory requirements.

19. Public Comments

Mr. Jorem Hoppenfeld commented that he is concerned about the behavior of a damaged steam dryer during a DBA, possibly releasing broken parts that would interfere with the operation of safety equipment. He thinks that this needs to be addressed. He is also concerned about iodine releases and the uncertainty of calculations at EPU on the iodine concentration used in dose calculations. He did not think that iodine spiking had been addressed at all, and the database is emperical, and notbased on direct measurements. There was no presentation that described the margin available to the 5 rem limit for control room operators.

He noted that there was very little said about the head loss across the sump screens, and he did not understand how the term "conservatism" is applied to the crud and sludge that is washed down during a LOCA. He did not think that the approach described is "conservative", and he did not understand how the different parts come together. Dr. Denning noted that the LOCA loads on the dryer don't seem to change, as a result of the EPU, and he wondered what the concern was. Mr. Hoppenfeld replied that he is not sure that the LOCA will not generate additional excitation in the dryer and cause a failure.

Mr. Peter Atherton commented that he is concerned about the present state of the NRC safety culture. He did not understand why a 2.206 petition that he submitted could be denied because the NRC did not believe that the accident would occur. He thought that the NRC was not giving sufficient consideration to environmental effects, disgruntled employees, or terrorist attacks. He was also concerned about the lack of any statement of what an acceptable margin would be to established limits. He did not understand why the NRC had not developed standards about how close values could come to limits.

Mr. Atherton asked the subcommittee why there is no standard for an acceptable release of radiation to the public. He thought that there should be some sort of study to determine the amount of radiation that could be released. He also expressed some concern about the verification of the computer codes that are being used to calculate the behavior of the steam dryers and the reactors.

Mr. Atherton commented that he has seen no failure modes and effects analyses (FMEAs) related to the power uprates, but he thought that these have been well known for many years. He noted that people have stated that the probability of the TMI accident has been estimated by some to have been 1E-9, but it still happened. And we now consider risks on the order of 1E-5 to be acceptable. He also did not understand how the OBE and the SSE for VY are so much smaller than the maximum expected 2400 year earthquake. He commented that there are many parts of the plant that are not qualified to withstand these earthquakes, and he does not understand why these systems and components are not upgraded.

Mr. Atherton recalled that other BWRs have had their torus's upgraded for seismic resistance, and other BWRs have replaced their core shrouds, but there was not discussion about upgrading either the torus or core shroud for VY. He believes that the whole review seems to be constrained by time schedules, and there is no attempt to spend the appropriate amount of time to review it before moving forward. Time management seems to be more important than safety, and he hopes that the ACRS will look into this indicator of a poor safety culture within the NRC.

Mr. Raymond Shadis commented that he hopes that the committee will eventually say something about the EPU process that is moving forward. He thought that the technical review was being driven by a schedule that had to be met, even though the staff is not done with its review. He also thought that the segmentation of the licensing actions, among the EPU, the ARTS/MELLLA, and the AST actions was questionable. He thought that there may be more technical issues that are floating around in separate licensing actions, but which would benefit from an integrated review. He reminded the committee that they have been asked by the state of Vermont to perform a review of this application, and he hoped that they would recommend that a more thorough independent inspection be performed.

Mr. Shadis described the NRC response to the request from the state of Vermont, and he concluded that the pilot inspection that was performed was inadequate to meet the needs of the Vermont PSB. The state asked for a very special inspection, but the staff did not provide it. He noted that the NRR staff depends on what it receives from the licensee, and he does not think that they will ever find any problems because they do not inspect on site. He plans to submit additional written comments and will persist until he convinces the ACRS of the need for an expanded independent inspection.

Mr. Shadis noted that it seems, from the presentations, that safety margins are eroding, and therefore there is a negative trend in safety. He specifically pointed to the increase in radiation dose due to the increase in N16 shine. This is an issue that is important to the state of Vermont, which has an agreement with VY to limit the site does to 20 mrem/yr. He thought that VY and the staff are making modifications to the methodology to remain within the state limit.

****ADJOURN ** at 5:00 pm**

ACRS Subcommittee
Meeting of 11/1505, 11/16/05, 11/29/05, and 11/30/05.
Vermont Yankee Extended Power Uprate Application
by
Graham M. Leitch 12/5/05.

This report is intended to summarize my views on the Vermont Yankee (VY) Extended Power Uprate Application (EPU). The views expressed are my own and do not necessarily represent the views of the ACRS, the ACRS Subcommittee, or the NRC.

Many issues were discussed in the inspection report, the SER and at the Subcommittee Meeting. I discuss here only those issues I feel are the most significant. These issues are listed below and are then discussed in some detail.

SIGNIFICANT ISSUES:
Adequacy of NRC Engineering Inspection.
Steam Dryer Degradation.
ECCS Suction Strainers.
Credit for Containment Overpressure.
Operator Training and Response Times.
Codes and Thermal Limits.
Public Comments.

ADEQUACY OF NRC ENGINEERING INSPECTION.
As part of the review of this application, the staff performed an engineering inspection at VY. As compared to a standard inspection, this inspection involved an additional 400 hours of inspection activity. It was conducted by inspectors having no recent VY involvement and it was part of a new pilot inspection program focused on risk significant systems. The inspection revealed 8 findings all of which were of very low safety significance. This inspection report was provided to the ACRS for review. The inspection was intended to be responsive to a request by the Vermont Public Service Board for such an inspection the results of which they requested be reviewed by the ACRS. I agree with the staff conclusion that the components and systems inspected would be capable of performing their intended functions and I see no justification for a further large scale inspection.

STEAM DRYER DEGRADATION.

There have been cracking problems with some, but not all, BWR steam dryers in the industry that may be caused by. or at least exacerbated by, EPU operation at those plants. VY has carefully inspected the dryer during a recent outage and found a number of IASCC causedcrack indications. These are believed to be self limiting and not of concern. They also found some fatigue cracking adjacent to lifting lugs which they repaired. They also proactively strengthened the dryer adjacent to the steam outlets by adding stiffening gussets. Since the root cause of dryer cracking is not completely understood from a theoretical view, the staff proposed some license conditions on VY which includes monitoring of carry over and monitoring of newly installed strain gauges on steam lines at 5% increments of power between 100% and 120% of original power level. They also require that the dryer be inspected at each of the next 3 refuel outages following EPU. I believe that the modification made to the dryer by VY and the license conditions imposed the by staff are adequate to prevent, or at least detect, the development of dryer cracking well before it becomes a safety issue.

ECCS SUCTION STRAINERS

The subcommittee reviewed the arrangement and design of the ECCS Suction Strainers which were greatly enlarged in 1996 at the time most other BWR's were modified. This design appears adequate and I support the staffs conclusions that pressure drop across these screens would be sufficiently low that it would not significantly contribute to a reduction in NPSH available to the ECCS pumps.

CONTAINMENT OVER PRESSURE

Where the single failure is assumed to be loss of the RHR system credit for containment over pressure appears to be necessary to achieve adequate NPSH to the ECCS pumps. There are two possible approaches to resolve this issue. These approaches do not alter the technical facts, but are two possible treatments of regulatory issues. One approach is to remove some conservatisms in the assumptions and replace with more realistic assumptions and demonstrate that containment over pressure was not necessary after all. The other approach is to use a risk based approach and demonstrate by RG 1.174 methodology that the delta risk is acceptably low. The first approach would likely work in this case but leaves unanswered the resolution of future applications. The second approach appears to use an acceptably small delta risk as a justification for compromising the long standing defense in depth principle. I tend to favor the first approach, but this seems to be a matter of which regulatory treatment should be applied here and not a technical issue. I feel this application ought not to be rejected on the basis of this issue, but the staff and the applicant need to present to the ACRS the merits of each approach either at the December meeting or at some subsequent time prior to approval.

A number of BWRs have been approved for EPU with credit allowed for containment over pressure in the range of 5 to 10 psi for 50,000 seconds. The ACRS has recently taken a position allowing credit for containment overpressure under some limited circumstances, but when to permit this credit depends on unclear guidance as to the criteria that makes it "required". This is more than a VY issue. A clear consistent regulatory approach is needed.

LARGE SCALE TRANSIENT TESTS
The staff has imposed license conditions related to plant response on tripping a condensate pump and either or tripping or analyzing response to tripping of a RFP. I feel these tests adequately demonstrate the plant response to the new logic for these pumps and the new automatic runback of the recirculating pumps. The additional information that could be gained by a full power plant trip might be marginally useful, but in my mind does not justify the risk and cost. The testing requirements imposed by the staff appears adequate.

OPERATOR TRAINING AND RESPONSE TIMES
The simulator and class room training have been modified to reflect plant changes to support EPU. These changes have already been made in the plant and in the simulator. Just in time training will be utilized to drill the crews for the pump trip test(s) referred to earlier.

The operators use E.O.P.s and associated flow charts to operate in abnormal situations. These documents have been reviewed and except for scaling changes are unchanged by EPU.

There are several situations, most notable ATWS, where required operator response times are reduced as a result of EPU. The applicant is aware of these situations and have and will continue to drill the operators in these scenarios. Properly trained crews are capable of meeting the slightly reduced response times.

After hearing the discussion by the VP Operations Manager, I feel the operators are well aware of and prepared for EPU

CODES AND THERMAL LIMITS

Codes used were those previously approved by the NRC. I note that the PCT increased by approximately 50 degrees F to 1960 degrees F which is still well below the 2200 degree F limit.

The addition of 0.02 to the MCPR safety limit appears to be an appropriate conservatism.

PUBLIC COMMENTS

There were many public comments at the Subcommittee Meeting.
Many of the comments related to issues of economics, reliability, or the desire to have the plant closed. These issues were not the purpose of the meeting, but the public should be assured that those matters raised which related to EPU were carefully considered by the ACRS.

CONCLUSION

In preparing this SER the staff was guided by RS 001. This relatively new standard takes into account lessons learned from previous power uprates and provides a more structured way of reviewing this and future applications. I feel that this standard helps improve the quality and consistency of the review.
In conclusion, I feel that the EPU should be approved pending the satisfactory resolution of the containment overpressure issue as discussed above.

Please let me know if I can provide any additional information.

Graham M. Leitch
12/05/05.

S. BANERJEE
REPORT

ACRS Meeting of the Subcommittee on Power Uprates
Quality Inn, Brattleboro, VT
November 15-16, 2005

General Remarks

This meeting was held to consider the Vermont Yankee Nuclear Power Station Extended Power Uprate and was one in a series, with the next being November 29-30 in Washington DC, following which there is supposed to be a meeting of the full ACRS in early December. In view of the November 29-30 meeting, the remarks in this report mainly pertain to uncertainties in the uprate case that were apparent from the presentations made at the November 15-16 meeting and the requirements for additional information.

The Vermont Yankee Extended Power Uprate (VYEPU) is for a 20% increase from 1593 to 1912 megawatts thermal. One other EPU has been approved for 20% (Clinton). During the discussions and the related public comments, it became apparent that there were several major issues that required further examination. These were:

1. Evacuation plans (noted by several members of the public as being deficient)
2. Need for independent and more complete safety inspection (also noted by several members of the public as being desirable)
3. Age of the plant and potential for age-related material failures, exacerbated by the power increase, which potentially lead to higher corrosion rates and flow-induced stresses. In particular, steam dryer integrity was noted as a major issue.
4. Crediting of containment overpressure during accidents to provide adequate NPSH for ECCS pumps
5. Power ascension and large transient tests, with the issue being that the staff has recommended significant relief from the performance of LTTs to the applicant.
6. It should be noted that there was also very limited discussion of ATWS, application of the MELLA operating line to this specific plant, and analysis procedures and results for related plant instabilities.

It was clear from the public comments that those representatives of the local community that attended the meeting were broadly opposed to the proposed power uprate. While many of the negative arguments advanced related to nuclear power in general, such as the generation of nuclear spent fuel, for which long-term storage had not been finalized, as well as routine releases from nuclear plants, nonetheless, the local community did feel that increased risks would arise from the proposed uprating without concomitant benefits. In the following section, "Specific Comments", I do not consider these broad concerns, or concerns related to problems such as evacuation, regarding which I have limited expertise. Discussion therefore centers primarily on thermalhydraulic issues.

Specific Comments

ATWS

There was virtually no discussion about this subject or the operating procedures, other than to say that ODIN had been used for the analysis rather than TRACG. It is expected that this subject will need thorough examination at the November 29-30 meeting, with supporting documentation being made available in advance. In particular, during a previous visit made by the Thermalhydraulics Phenomena Subcommittee to GE, some difficulties in ATWS-related calculations with TRACG had been noted. Was this the reason that TRACG was not used?

Containment Overpressure Credit

Entergy and NRC staff presented arguments justifying the allowance for containment overpressure in handling the increased temperature of the sump water that would be expected in a LOCA due to the power uprate, and would therefore cause problems with NPSH for the ECCS pumps. The statement was made that considerable conservatisms were inherent in the calculations that showed the requirement for this overpressure and that, if these conservatisms were removed or reduced, the requirement was eliminated. Furthermore, the requirement only arose if one RHR exchanger did not operate, based on the single faulure criterion. In view of this single faulure criterion, the view was implicitly advanced that containment leakage would constitute a second failure, which had a very low probability. (Note however that only a ½ inch hole would suffice to remove the overpressure!) Be that as it may, it was not clear what assumptions had gone into the risk assessments, and the Vermont Department of Public Service suggested an alternative approach based on a top event being "pump fails due to inadequate NPSH". When questioned about aspects of the uncertainty in the calculations, e.g., the pressure loss due to debris accumulation on the strainers during LOCA and in particular the fate of paint chips, the staff and Entergy personnel deferred more detailed discussion to the November 29-30 meeting. This subject, therefore, remains open, and it would be desirable to have detailed information regarding these matters in advance of the meeting.

Steam Dryer Integrity

Entergy had, in a very recent higher resolution inspection, found a dramatic increase in the number of steam dryer cracks compared to their previous inspection. As problems had also been found in Dresden and Quad Cities, both on power uprates, potential problems with steam dryer integrity at Vermont Yankee must be examined in more detail at the November 29-30 meeting. In particular, a statement was made that a piece of the dryer had found its way to the top of the core in Quad Cities, with the potential for flow diversion. If this is true then the NRC staff contention that steam dryer integrity does not pose a safety issue needs evaluation.

Preliminary discussion also indicated that the causes for the vibration and related dryer integrity problems were not well understood. It was stated that some CFD calculations had been done but were inconclusive. In spite of this, the staff had recommended that Vermont Yankee power ascension tests be allowed with a limited placement of vibration detectors. It was not clear that these vibration detectors (strain gauges) would provide adequate information about what was actually going on in the stream dryers. In any case, the relationship between the measurements of these detectors and steam dryer vibrations must be clarified at the November 29-30 meeting. At present, the problem seems so ill understood that even if the cracked steam dryer was replaced by a new one, it is not clear how long the replacement would last in the increased flow conditions.

Recommendations

Even without considering evacuation plans and the requirements for a more complete safety inspection, subjects in which I have little expertise, there are significant issues which need investigation and discussion at the November 29-30 meeting. These are:

1. ATWS and associated uncertainties, as well as operating procedures.
2. The causes of steam dryer vibrations and whether the staff-recommended measurements during power ascension tests are adequate
3. Uncertainties in the calculations for the required ECCS pump and NPSH, with particular emphasis on pressure losses through debris accumulated on strainers
4. MELLA line operation effects on stability boundaries for this plant, and associated operating procedures.

In more general terms, there was a sense in the public comments that this was an aged plant, a fact implicitly acknowledged by the plant operators in trying to avoid large transient tests, and there could be hidden problems that would emerge only upon very detailed inspection and rigorous large-transient testing. The vulnerability of the plant to terrorist attack was brought up on several occasions. Certainly, while much of the public comment was outside the range of issues that we were charged to deal with, nonetheless there were several points of interest, including a Vermont Yankee containment safety study that had been transmitted to Mr. Denton in 1986, which we have not had access to as yet.

I felt the meeting, and especially the public comments, were a worthwhile and necessary prelude to ACRS's deliberations..

From: "banerjee" <banerjee@engineering.ucsb.edu>
To: <rxc@nrc.gov>
Date: 12/7/05 7:25AM
Subject: VY EPU Meeting Nov 29-30, REPORT

Dear Ralph and Graham,

PLease find attached my report on the Nov 29-30 meeting, which i hope will be available to thoe interested including Dr. Denning.

I strongly recommend against:

1. Allowing the containment overpressure credit in view of the more than order of magnitude nonconservative uncertainties in the debris bed/strainer pressure loss calculations- they are simply incorrect and optimistic as presented.

2. Allowing the power ascension tests with the proposed steam line instrumentation. The case for proceeding in this manner was very weak. The FLUENT calculations are useless and the ACM model has no predictive capability.

I believe that any colleague in the fluids- thermal community (Acrivos, Leal, Homsy, Denn, Hewitt, Bankoff, Sreenivasan, amongst many others) would fully support these views were they presented to them for peer review.

Best regards,

Sanjoy

CC: <g.b.wallis@dartmouth.edu>, <banerjee@engineering.ucsb.edu>

REPORT – Dr. Sanjoy Bannerjee

ACRS Subcommittee on Power Uprates
November 29-30, 2005, Rockville, Maryland

1. General Comments and Recommendations

This meeting was held at the USNRC/ACRS Offices to discuss the Vermont Yankee extended power uprate, following a meeting held in Brattleboro, VT on November 15-16. The agenda covered a broad range of topics, of which I will only comment on

1. Steam dryer and monitoring
2. NPSH of ECCS pumps (in particular the effect of debris bed pressure loss)
3. ATWS (stability and operator action times and QA)
4. Thermalhydraulics analysis methods

My recommendations, which are discussed in more detail in the following sections are as follows:

1. The VY steam dryers should be instrumented for vibrations or replaced with new dryers instrumented like the new Quad Cities II dryers. The proposed monitoring and power ascension program is based in a low and possibly wrong understanding of the processes involved, and is therefore high risk. The so-called ACM model has little or no predictive capability. The reasons for this are detailed under 'Specific Comments.' It is likely that vibrations originating in the dryers themselves, which could lead to failure, will not be detected. A failure will be a notable public relations disaster, especially as the possibility has been repeatedly discussed in the public comments that ACRS received.

2. There is large and non conservative uncertainty in the debris bed head loss calculations which would substantially reduce the NPSH available for the RHR and CS pumps. This arises from incorrect assumptions regarding approach velocity to the strainers, which is much too low, and it is probable that all the sludge and a portion of the paint chips could be entrained and clog the strainers. The experiments conducted to support the analysis are flawed, as they are for a single strainer with a much higher surface area for unit flow, than would obtain in a partially clogged stack of strainers, which corresponds closely to the real situation. In view of this uncertainty, the positive containment pressure needed to ensure proper operation for the ECCS pumps may turn out to be much more than estimated. There are sufficient uncertainities in the calculations presented that I would recommend that the positive containment pressure credit should not be allowed. The risk based arguments are flawed and would not stand up to peer review by any group of expert fluid dynamicists.

3. The ATWS operator action decision tree is extremely complex, in fact I had difficulty following the various branches and actions. It should be QA'd to ensure that adequate times are available for operator actions which could have to occur faster with the uprate. This should be done in a supplementary inspection, which is also desirable for evacuation plans. I strongly recommend that an inspection, to supplement the one already done, be carried out with objectives focused to include the two subjects mentioned above- before the EPU is approved.

4. The analysis methods for ATWS instability and CHF need to be thoroughly reviewed. Calculations with up to date codes like TRACG have not been performed, and steady state CHF correlations are used for a rapidly oscillating situation. The proposed uprating highlights the need for review, as more fuel is susceptible to damage because of the flattened core. In private conversation with NRC staff they were frankly embarrassed that they could not properly address the questions addressed during the meeting.

2. Specific Comments

A. Steam Dryer and Monitoring Program

Some, not very coherent, information was presented regarding measurements and analysis of steam dryer vibration and failure associated with power uprates. While the dryer is apparently not of direct safety significance, nonetheless failures impact plant operation and garner enormous unfavorable comment in the media. Furthermore, a failure could ultimately propagate into a safety-significant incident if broken pieces damaged turbines, causing operational transients and shut-down, as well as damaged discharge valves. Experience with other uprated PWR plants (Quad Cities II and Dresden) indicate that dryer cracking and subsequent failure are not isolated, random incidents, but are directly related to uprating and the resulting higher steam velocities. The experience is far from comforting. In spite of the problem being known from Quad Cities, the dryers in Dresden cracked, even though measures were taken to strengthen them. Furthermore, the Quad Cities dryers cracked, in spite of strengthening, and ultimately had to be replaced.

Presentations were made by Vermont Yankee staff, GE, and NRC to outline the program of experiments and analysis that has been put in place to address this problem. CFD analyses with FLUENT have been undertaken to clarify the flow structures and pressure fields arising in the dryer region, but are essentially useless. A one-dimensional model had also been constructed for the acoustic phenomena in the steam lines, coupled to a somewhat impressionistic multidimensional acoustic model made for the region of the dryer and the dome. The FLUENT calculations were used to study the so-called "hydrodynamic pressure fluctuation modes" arising from fluid motions, and the acoustic modes addressed through solution of a Helmholz equation. These 'far-field' calculations, in principle, should use the hydrodynamic near-field calculations as source terms and essentially be 'Ffowcs- Williams theory' estimates that would be too expensive to compute with usual CFD methods. This is far from what is being done!

The overall results of this program were unsatisfactory. NRC Staff concluded that the CFD calculations with FLUENT had high levels of uncertainty, and wee essentially meaningless, because of a variety of factors, including coarse nodalization. No attempts, for example, were made to determine the effect of refined nodalization. For example, based on the CFD calculations, only relatively low-frequency fluctuations and pressure were seen in the steam dome region, but whether higher-frequency fluctuations could obviously arise with higher fidelity and more resolved simulations. The comparisons between the acoustic (Helmholz equation-based) equations were also poor in comparison to experiments conducted in a scaled-down GE facility. This was attributed to the need for source and damping correlations for this so called ACM model- which in any case is of no predictive capability.

Nonetheless, based on this potpourri of models, and some monitoring of pressure fluctuations in full-scale operating plants it was suggested that the higher frequency (>100 Hz) arose from phenomena associated with draw-off lines leading to relief valves, etc., and were apparently not generated in the steam domes. It is hypothesised that the damaging fluctuations arise from vortex shedding in branches and closed ends considerably downstream from the vessel. These fluctuations are then supposed to propagate into the steam dome regions, excite natural frequencies in the dryer that ultimately lead to cracking and, potentially, failure. The evidence to support this is virtually nonexistent and would not stand up to peer review.

In spite of this uncertainty, it seems to be taken for granted that the damaging vibrations originated far downstream of the vessel and could therefore be monitored by instrumenting the steam line. There is no direct proof of this supposition. No detailed predictive model was advanced. The acoustic model proposed relied on measured inputs and appeared to be a method of extrapolating from steam line measurements to what might happen in the vessel than a predictive simulation.

The upshot of all this was that the applicant and the staff have both agreed to a program of monitoring during a gradual power ascension in Vermont Yankee that relies heavily on strain gauge measurements made on the outlet steam line. This may be all that can be done with the existing dryers as they are said to be too hot to instrument directly, but the issue remains as to whether this is satisfactory.

In summary, little was presented that indicated a clear understanding of the phenomena associated with the higher steam flows and why they cause steam dryer damage. The program of CFD calculations and GE experiments on scaled-down facilities do not appear to have been particularly useful, though the measurements in Quad Cities II with the instrumented dryer, as well as transducers on the steam line, did provide important information, which serves as input and validation for the acoustic model proposed. However, the predictive power of this model, and hence its applicability to an unknown situation such as the power uprate at Vermont Yankee, remains uncertain.

B. ATWS and Stability Issues

The applicant, GE and staff made several presentations related to operating conditions (critical power ratio), operational transients, though with only a small amount of information regarding ATWS. Since the power uprate is being achieved primarily by flattening the core, there is not much at issue with regard to critical power ratios, other than some uncertainty in reactor physics calculations and, hence, core power distributions under uprated conditions. The reactor physics uncertainties arise because of the higher void fractions to be expected in the upper regions of the core and depletion. While such calculations have been validated for earlier fuel assemblies, the 10x10 assemblies need to be gamma scanned. In view of these, some additional safety factors were added to the critical power ratios to determine the ranges of operational flexibility. There is little to take exception to here, but the situation is somewhat different with regard to stability and, in particular, instability during ATWS.

So far as could be determined, the staff had not carried out in any detail independent evaluation of the stability maps and how they change with the power uprates Nor have they evaluated in detail the ATWS-related instabilities. They were in the process of doing this for MELLA+ but were apparently having difficulty with TRACE for undertaking such calculations. The issue here is related to operator actions and how quickly they must take place to limit the severe oscillations that occur in core power and flow rate when ATWS-related instabilities arise. These issues have been addressed in the past using earlier-generation calculational methods and a variety of assumptions related to heat transfer, critical heat flux and core hydrodynamics. Calculations with improved codes, such as TRACG and TRACE, are still to be done, presented to ACRS, and subjected to scrutiny. Legalistic arguments were advanced to support the case for operating along the MELLA line, as apparently such courses of action had been approved in the past. Such arguments, while they may conform to the letter of the law, certainly do not conform to the spirit. If new and better calculations indicate more severe oscillations and greater fuel damage, then these should, in any case, be taken into account and measures taken to mitigate their effects.
This whole area, in my mind, remains an issue, as what was allowed in the past with a much less flat core is problematic with the uprate, where much more fuel is operating close to damage thresholds. Therefore, in incidents where fuel damage may only have occurred in a relatively small number of rods before the uprate, the damage in the uprated core would be expected to be much larger. The arguments were also advanced that if all the operator actions had to be taken during ATWS were done in time and perfectly, then there was no anticipated fuel damage. The fact that much more of the core was operating close to its limits did not enter the considerations. The uncertainties in the calculations are sufficiently large that such an approach is difficult to defend and seems oblivious to the uncertainties in timing and effectiveness of operator actions.

C. Adequacy of Analytical Methods

This is of concern with regard to the Vermont Yankee uprate, particularly with regard to flow instability and instability during ATWS. Independent evaluation of the methods used by GE do not appear to have been done. The staff did not appear to be in a position to assess the validity of the results with regard to codes such as TRACE, though they had done some independent evaluations for some similar thermalhydraulics transients with RELAP. The adequacy and independent assessment of vendor calculations with regard to stability and ATWS remains an open subject. This may be something of a generic issue (such codes are used for every uprate, which amounts to a situation where margins are cut down and more fuel is brought closer to various limits). A careful evaluation of vendor and utility calculations with regard to a wide variety of transients and of normal operating conditions as well as accidents is necessary, using codes like TRACG and TRACE.

D. NPSH and Screen Head Loss

Uprates lead to high temperatures in the water bodies from which the ECCS pumps draw, leading to reductions in available NPSH. Therefore, more care must be taken in evaluating various pressure losses, as well as the

potential for air entrainment due to vortex formation close to intakes. The calculations for pressure losses take into account elbows, fittings, and so on, in the piping leading into the pumps. The assumption is made that these pipes are smooth and without fitting. Is this a good assumption? If one assumes some degradation of pipe surfaces, how much are the head losses affected.
Even more important, are the assumptions that go into calculating the head losses through the screens that strain out debris. Are these adequate, and what experimental database supports such calculations? These are important issues for the Vermont Yankee uprating because the NPSH requirements for ECCS are such that containment overpressure has to be appealed to. In any case, violates the defense in depth principle, which is fundfamental to nuclear safety. This is quite independent from the single failure criterion, thus the argument that the RHR system failure, at least of one train, is necessary to require containment overpressure, and therefore that failure of the containment would constitute a second failure, is specious. This becomes a somewhat philosophical issue and, more to the point, is whether the calculations that result in requiring a certain level of containment overpressure are defensible. In my view they are not. First the wrong approach velocity is used giving extremely optimistic results on pressure losses through the screens. Second, the pressure loss correlations are base on single strainer experiments that have no bearing on the situation that would obtain with partially blocked stacked strainers. Third, the assumptions regarding the capture of the sludge (50%) and the bed porosity are incorrect, and nonconservative,. Fourth, the approach velocity in reality is high enough to entrain paint chips which could lead to severe blockage. All this suggests that uncertainties are more than an order of magnitude. To allow this type of analysis to form the basis of allowing containment overpressure credit would be imprudent.

Entergy

Entergy Nuclear Northeast
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December 2, 2005

Docket No. 50-271
BVY 05-107
TAC No. MC0761

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Vermont Yankee Nuclear Power Station
Technical Specification Proposed Change No. 263 – Supplement No. 43
Extended Power Uprate – Response to Request for Additional Information

Reference:
1) Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263, Extended Power Uprate," BVY 03-80, September 10, 2003

2) U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," November 25, 2005

This letter provides additional information regarding the application by Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy) for a license amendment (Reference 1) to increase the maximum authorized power level of the Vermont Yankee Nuclear Power Station (VYNPS) from 1593 megawatts thermal (MWt) to 1912 MWt.

By letter dated November 25, 2005 (Reference 2), the NRC staff requested additional information regarding Entergy's probabilistic risk assessment studies that support the application for extended power uprate. Attachment 1 to this letter provides responses to the specific information requested.

There are no new regulatory commitments contained in this submittal.

This supplement to the license amendment request provides additional information to clarify Entergy's application for a license amendment and does not change the scope or conclusions in the original application, nor does it change Entergy's determination of no significant hazards consideration.

If you have any questions or require additional information, please contact Mr. James DeVincentis at (802) 258-4236.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on December 2, 2005.

Sincerely,

Jay K. Thayer
Site Vice President
Vermont Yankee Nuclear Power Station

Attachments (1)

cc: Mr. Samuel J. Collins (w/o attachment)
Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. Richard B. Ennis, Project Manager
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
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USNRC Resident Inspector (w/o attachment)
Entergy Nuclear Vermont Yankee, LLC
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Mr. David O'Brien, Commissioner
VT Department of Public Service
112 State Street – Drawer 20
Montpelier, Vermont 05620-2601

Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 43

Extended Power Uprate

Response to Request for Additional Information

Total number of pages in Attachment 1 (excluding this cover sheet) is 6.

NRC RAI APLA-A-1

Supplement 38, Attachment 1, page 9: Provide the engineering assessment that shows that the residual heat removal (RHR) and core spray (CS) pumps can operate at significantly reduced net positive suction head (NPSH) compared to the design NPSH, which is based on the results of tests conducted at Browns Ferry as described in NUREG/CR-2973. Have the conclusions of this engineering assessment been discussed with the pump manufacturer (Sulzer Bingham)? If so, does the pump manufacturer concur with the conclusions?

Response to RAI APLA-A-1

The discussion of pump operation at reduced levels of available NPSH in Supplement 38 was an indication of the margins provided against severe operational and mechanical degradation inherently available in the RHR and CS pump designs. However, for the risk evaluation of containment overpressure (COP) credit, the assumption is made that loss of containment results in loss of COP, which in turn results in failure of the pumps. This was a conservative assumption used to determine the risk impact of crediting COP on available NPSH.

The original engineering assessment for pump operation at reduced NPSH was performed by Tennessee Valley Authority engineers. The applicability of the results to Vermont Yankee Nuclear Power Station (VYNPS) RHR pumps was documented in a Yankee Atomic Electric Company memo written by R. Turcotte, dated January 13, 1993. This memo refers to a discussion with a nuclear pump expert at General Electric. There is no mention in the memo of any similar discussion with the pump manufacturer. The VYNPS RHR pumps and the Browns Ferry RHR pumps were purchased by General Electric under the same purchase order from Bingham (now Sulzer Pump Co.).

An evaluation of applicability of the results to VYNPS CS pumps is documented in Supplement 38 (Attachment 1, page 10). The VYNPS applicability assessments were not discussed with the pump manufacturer. As noted above, these results are not used in VYNPS deterministic evaluations of NPSH margin.

NRC RAI APLA-A-2

Supplement 38, Attachment 1, page 19 and Supplement 39, Attachment 1, pages 12 and 13: It is stated that EPRI TR-1009325 was used to determine the probabilities of containment pre-existing leakage. The NRC staff has not yet accepted this reference as a technical basis for granting permanent 15-year integrated leak rate test (ILRT) intervals. In fact, the Nuclear Energy Institute (NEI) submitted an updated version of this document for further staff review on October 26, 2005. The staff notes that the technical basis for containment leakage probabilities used to justify the one-time 15-year ILRT interval that was granted in VYNPS Amendment No. 227, dated August 31, 2005, was EPRI TR-104285, and that the containment leakage probabilities in this report are notably higher than those provided in EPRI TR-1009325. Either justify the use of EPRI TR-1009325 as an acceptable source of containment leakage probabilities, or reassess the change in core-damage frequency (CDF) caused by crediting containment accident pressure using containment leakage probabilities that are consistent with the recently granted one-time 15-year ILRT interval.

Response to RAI APLA-A-2

The basis for the VYNPS ILRT license change (license amendment 227) assumed that, for EPRI class 3a (small containment leakage), the leakage rate is 10La with a probability of 0.027. Similarly, for EPRI class 3b (large containment leakage), the leakage rate is 35La with a probability of 0.0027. These values are taken from NEI interim guidance for performing risk impact assessments in support of one-time extensions for Containment ILRT (issued in November 2001) and are considered conservative.

Because these leakage values are conservative, an update of the original EPRI evaluation (EPRI TR-104285) to assess the risk due to a revised containment leakage rate interval considered an expert elicitation process in the development of the probability of a large pre-existing containment leak. The results of this process are found in EPRI TR-1009325, which is currently undergoing a significant revision in which the expert elicitation results will be a sensitivity case in the analysis. However, the report also includes and references supporting documentation that compares the expert elicitation results with test data with good agreement.

The updated values are a better representation of leakage rates used in evaluating the one-time extension in the change for the ILRT interval. For example, these updated values considered a closer examination of previously observed containment leakage events that had been designated as failures, the potential risk benefits associated with additional containment inspections, and potential indirect containment monitoring techniques that would provide indications of a containment leak.

It is recognized that EPRI TR-1009325 has not yet been approved by the NRC. However, these updated values represent valuable improvements in evaluating containment leakage scenarios in risk sensitivity analyses.

NRC RAI APLA-A-3

Supplement 38, Attachment 1, page 20 and Supplement 39, Attachment 1, page 22: Provide the high confidence of low probability of failure (HCLPF) values used in the Seismic Margins Analysis (SMA) of VYNPS for the following: reactor coolant system piping, reactor vessel supports, safety relief valves (SRVs), and the containment.

Response to RAI APLA-A-3

HCLPF values were not computed for the reactor coolant system piping, reactor vessel supports, safety relief valves (SRVs), and the containment, because all of those components were screened out from the analyses.

From the VYNPS Individual Plant Examination on External Events (IPEEE), major components and equipment in the nuclear steam supply system (NSSS), which are located inside containment, are excluded from the scope of the Unresolved Safety Issue (USI) A-46 review. The NSSS primary coolant system (which includes reactor coolant system piping, reactor vessel supports, and SRVs) is screened out per EPRI NP 6041-SL.

The Seismic Class I structures (which includes the drywell) were also screened out since they are designed for a safe shutdown earthquake (SSE) greater than 0.1g. Mark I containments have undergone significant strengthening to resolve dynamic loading issues. The VYNPS torus, which was originally designed for seismic loads, has thus been significantly strengthened. Because of this, the IPEEE reviewers determined that significant margin had been added to the design of the torus and that the torus should be screened out.

NRC RAI APLA-A-4

Supplement 38, Attachment 1, page 20 and Supplement 39, Attachment 1, page 22: Could a fire simultaneously cause a stuck-open relief valve and a failure of the containment isolation (CI) system?

Response to RAI APLA-A-4

There is no postulated fire that could simultaneously cause a stuck-open relief valve and a failure of containment isolation. The following information is taken from the VYNPS IPEEE, Rev. 2 on Internal Fires Analysis (2004):

> Power cables for each safety relief valve (SRV) circuit are isolated in grounded, steel raceway (conduit) from the control room floor, through the cable vault and reactor building, through the drywell electrical penetration to the associated SRV. The steel conduit is dedicated to each SRV power solenoid (SOV) circuit and no other power source cables are located within these conduits. The power cables for the SRVs are IEEE 383 qualified. Given the SRV circuit design and the grounded, steel conduit isolation in the reactor building and cable vault, the likelihood of a fire-induced hot short causing spurious opening of an SRV is judged to be very remote and is not evaluated further. Fire damage to SRV power cables in the reactor building and cable vault is assumed to fail the associated SRV in the de-energized, closed position. The power cables for SRV-71A and SRV-71B are routed separately from SRV-71C and SRV-71D cables in the reactor building.
>
> An automatic depressurization system (ADS) inhibit switch is provided in the control room for manual blocking of a postulated fire-induced ADS signal which could cause the SRVs to open. The inhibit switch interrupts both the positive and negative legs of each SRV power circuit. The inhibit switch enclosure and downstream conduit leading to the control room to cable vault penetration are protected with a 1-hour rated fire barrier. Given the inhibit switch design and the rated fire barrier protection, the likelihood of a fire-induced hot short causing a spurious opening of an SRV is judged to be very remote for control room fires. Section 4.10.3 addresses LOCA events due to SRV opening and failure to re-close in the evaluation of control room fires.

NRC RAI APLA-A-5

Supplement 39, Attachment 1, general: Is the overall intent of the risk evaluation of the proposed containment overpressure credit to provide a sensitivity analysis that investigates modeling uncertainty in the baseline post-EPU PRA? The NRC staff notes that Supplement 38 indicates no overpressure credit is required using realistic assumptions. Hence, there should be no changes between the pre-EPU and post-EPU PRA models with respect to their treatment of the proposed overpressure credit.

Response to RAI APLA-A-5

The overall intent of the risk evaluation of the proposed COP credit is to provide a very specific sensitivity analysis on this one aspect (i.e., need/no need for COP). Because no COP is required using realistic assumptions, this analysis (needing COP) represents a special case and falls outside the baseline post-EPU PSA. There is no PRA modeling change between pre-EPU and post-EPU with respect to COP. The comparison is drawn only for insight purposes and does not supersede the deterministic analyses which will become the new licensing basis for VYNPS.

NRC RAI APLA-A-6

Supplement 39, Attachment 1, general: Does the change in CDF only consider the impact of the proposed overpressure credit, or does it also include the impact of other changes resulting from the proposed EPU (e.g., shorter operator times due to higher decay heat)?

Response to RAI APLA-A-6

The change in CDF in Attachment 1 of Supplement 39 only considers the impact of the proposed COP credit. The base case was the post-EPU case with realistic inputs and does not require crediting for COP. The evaluation provided in Supplement 39 was only provided as a sensitivity analysis. It does not alter the base EPU evaluation.

NRC RAI APLA-A-7

Supplement 39, Attachment 1, page 13: It is stated that containment integrity (Event IP) is considered when the hardened torus vent is being used (Event VT) to prevent over-pressurization failure of the containment following a loss of torus cooling (Event TC). It is difficult to interpret the event tree logic (e.g., the large loss-of-coolant accident (LOCA) event tree) in the context of this statement since Event IP appears before Event VT. To help clarify the NRC staff's understanding of the modeling approach taken, provide a narrative explanation of each core-damage sequence in the large LOCA event tree.

Response to RAI APLA-A-7

In conjunction with development of the COP risk assessment (i.e., Supplement 39), additional thermal-hydraulic analyses were performed using the Modular Accident Analysis Program (MAAP) computer code in order to evaluate operator response timing. An error was discovered when a review of MAAP computer runs determined that operator action to control torus venting is ineffective in controlling containment pressure to preclude NPSH concerns for low pressure coolant injection (LPCI) and core spray pumps. Therefore, operator action AINPSH, "Operator

Fails to Control Vent and LP Fails due to Loss of NPSH", which involved the potential failure by the operator to adequately control torus venting such that NPSH is lost and ECCS pump failure is assumed to occur. In conclusion, containment integrity is not considered in relation to operation of the hardened torus vent (Event VT) to prevent over-pressurization failure of the containment following a loss of torus cooling (Event TC). The engineering report (i.e., Supplement 39, Attachment 1) has been revised to reflect this model change, and the referenced statement was deleted.

NRC RAI APLA-A-8

Supplement 39, Attachment 1, page 14: If the containment is not intact (Event IP occurs), why is it possible to credit alternative injection and containment overpressure (COP) control (Event AI)?

Response to RAI APLA-A-8

The statement on page 14 (Supplement 39, Attachment 1) was in error. As stated above in response to RAI APLA-A-7, the VYNPS PRA model was changed, deleting the use of COP control in top event AI (Altemate Injection). The engineering report provided in Supplement 39 has been revised in this regard.

NRC RAI APLA-A-9

Supplement 39, Attachment 1, page 18 and Tables 3.2A and 3.3: On page 18, it is stated that CONFIG#1 represents the risk when the COP is not available and CONFIG#2 represents the risk when the COP is available. However in Tables 3.2A and 3.3, the CDF associated with CONFIG#1 is lower than for CONFIG#2. Please clarify. Also, note that in Table 3.3, the total CDF for CONFIG#1 is incorrect (typographical error).

Response to RAI APLA-A-9

The statement on page 18 of the engineering report provided in Supplement 39 was revised to more clearly state that CONFIG#1 represents the risk when COP is not necessary to satisfy RHR and CS pump NPSH requirements, and that CONFIG#2 represents the risk when the COP is necessary to satisfy RHR and CS pump NPSH requirements.

The typographical error in Table 3.3 (the total CDF for CONFIG#1) has been corrected in the revised engineering report.

NRC RAI APLA-A-10

Supplement 38, Attachment 1, page 18: It is stated that the only difference between the model cases lies in End state Bin IIV. However, Table 3.2A indicates that End state Bins ID, IIIC, IVA, and IC also change. Please clarify.

Response to RAI APLA-A-10

The statement on page 18 of the engineering report provided in Supplement 39 is incorrect. The engineering report has been revised, deleting the statement which referred to end state bin IIV.

The minor decrease in end state bins IIV and IVA (~2E-11) was due to capture of some existing sequences for CONFIG#1 in end state bins ID and IC for CONFIG#2. This occurred when the end state binning rules were revised to reflect the change in success criteria used for CONFIG#2 (i.e., COP is necessary to satisfy RHR and CS pump NPSH requirements). End state bins ID, IIIC and IC reflect the increase in CDF between CONFIG#1 and CONFIG#2. The engineering report provided in Supplement 39 has been revised for completeness and now includes split fraction (SF) sequence information relative to end state bin ID for CONFIG#1 and CONFIG#2 (i.e., Tables 4.3.C and 4.4.C, respectively), in addition to the existing SF sequence information for end state bins IC and IIIC.

112 STATE STREET
DRAWER 20
MONTPELIER VT 05620-2601
TEL: (802) 828-2811

FAX: (802) 828-2342
TTY (VT): 1-800-734-8390
e-mail: vtdps@psd.state.vt.us
Internet: http://www.state.vt.us/psd

STATE OF VERMONT
DEPARTMENT OF PUBLIC SERVICE

November 28, 2005

Mr. John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

RE: ACRS Power Uprate Subcommittee in Vermont, November 15-16, 2005

Dear Mr. Larkins:

By letter of September 17, 2004, I requested that the ACRS conduct part of its deliberations on Vermont Yankee power uprate in Vermont. On behalf of the Douglas Administration and the people of Vermont, I would like to thank the Advisory Committee for its Power Uprate Subcommittee meetings in Brattleboro, Vermont, on November 15 and 16, 2005.

The Subcommittee meetings provided an opportunity for Vermonters to observe first-hand the consideration that the ACRS gives for power uprates, as well as giving members of the public an opportunity to express their views on the subject. I appreciated greatly the Subcommittee's creating this opportunity, and the careful listening to public comments over the ample time set aside. I would like to specifically note the helpful coordination efforts of Mr. Ralph Caruso of your staff.

We look forward to your continued consideration of the power uprate request from Vermont Yankee.

Sincerely yours,

David O'Brien, Commissioner

cc: Ralph Caruso, ACRS staff

I:\VYPowerUprateNRCHearing\ACRS Presentations\ACRS - DPS Letter.wpd

From: "janet conover" <janetconover@msn.com>
To: <RXC@nrc.gov>
Date: 11/29/05 3:49PM
Subject: Vermont Yankee Relicence

Dear Sir,

We need electricity and oil is controlled by off shore sources. For economic growth and national security, I strongly support increasing the energy output of the Vermont Yankee plant to the largest reasonablel, safe, and techically able level.

France, China and many other countries have rested their power future on Nuclear. Don't let the whiney greens get you down.

My understanding is that Yankee has been running at significantly under its potential output for years. Please turn up the fires. It is in the nation's best interest.

Jan Conover

66 Orange Road
Warwick MA 01378

From: "Wyn Cooper" <wcooper@sover.net>
To: <rxc@nrc.gov>
Date: 11/29/05 6:26PM
Subject: VT Yankee uprate

Dear Ralph Caruso and members of the ACRS,

I am writing to express my strong disapproval of the proposed uprate at Vermont Yankee. I live in Halifax, VT, 11 miles from the reactor, and have followed the news and attended several meetings regarding the uprate. The plant is old, and has had numerous problems as you are well aware. Please take my opinion into consideration.

Thank You.

Sincerely,

Wyn Cooper

From: <diatom@nenetworks.com>
To: rxc@nrc.gov <rxc@nrc.gov>
Date: 11/29/05 7:33PM
Subject: Vermont Yankee Uprate

Please do not allow this proposed uprate to occur. The potential for accidents is too high, the plant too old, and our communities here too wonderful to take the risk.
OUr need for energy can be met by current levels of production at the plant, combined with conservation and alternative sources.
Thank you - Joan Deely, Northfield, MA (within the 10 mile radius)

From: "cherbill" <cherbill@svcable.net>
To: <rxc@nrc.gov>
Date: 11/29/05 9:24PM
Subject: no Vermont Yankee uprate

Can you really turn up the heat by 20% on a 33 year-old tea kettle?
It does not sound safe to me.

Cheryl Wilfong
314 Partridge Rd
E Dummerston VT 05346

From: Peter Alexander <cristobl@tds.net>
To: <rxc@nrc.gov>
Date: 11/29/05 9:41PM
Subject: Vermont Yankee Uprate

Dear Mr. Caruso:

Thank you for the opportunity to comment about the proposed Vermont Yankee Extended Power Uprate. I have followed this issue very closely for nearly three years and am quite familiar with the technical concerns. The most simplistic questions are sometimes the very best, so I will ask the NRC, "Why are you allowing Entergy to perform a nuclear experiment in Vermont that should be performed at Brookhaven or Los Alamos? What moral, legal, and scientific basis is there for allowing this profit-driven company to test an unproven, high risk technology on an unsuspecting public?

Please inform the NRC staff, commissioners, and the ACRS that they should only allow the uprate if they can in clear conscience, as individuals willing to be held accountable by their grandchildren, answer these questions to the satisfaction to the people of Vermont, New Hampshire, and Massachusetts.

Sincerely,

Peter Alexander
cristobl@tds.net
cell: (802) 380-3080

From: "Emily Koester" <emilyjk@mtdata.com>
To: <RXC@nrc.gov>
Date: 11/29/05 10:17PM
Subject: concerned

I am concerned about Vermont Yankee's proposed upgrade. I feel that my health and the health of my 3 young children may be at stake. There are much safer alternative energy sources in the world, and I would be happy to pay more for my energy if it were renewable and safe (I already pay more for green electricity).
Thank you for listening to (reading) my comments.
Sincerely,
Emily J. Koester
245 Gale Road
Warwick, MA 01378

From: Scott Ainslie <ainslie1@ix.netcom.com>
To: <rxc@nrc.gov>
Date: 11/29/05 11:06PM
Subject: Vermont Yankee Uprate

Dear Mr. Caruso:

I know that you do not live within ten miles of Vermont Yankee Nuclear Plant. And I know that you have the capacity to feel empathy for those of us who do. I call on you to exercise extreme caution as if your children and grand children lived right here. I know you are capable of that.

Vermont Yankee is a poorly maintained, badly run, dangerous plant. We have nuclear engineers who are in favor of nuclear power who have repeatedly taken stand against this plant and its current owners and managers. Vermont Yankee is a plant that could not be licensed today, if it came in with the safety issues that are present and documented. There is no justification for approving an uprate or, God forbid, license extension in the absence of an Independent Safety Assessment of the scope and depth specified by the Vermont Public Safety Board and performed at the Maine plant back in the 1990s.

Our children have elevated levels of Strontium 90 in their baby teeth. We already have a nuclear burden that will persist long after a national nuclear waste repository is established. And at this point, that seems impossibly far in the future, as Yucca Mountain continues to fail critical tests.

You are our last line of defense. Please be diligent and compassionate as you do your work. Our lives, if not yours, depend on it.

Thank you,

Scott Ainslie
101 Washington St.
Brattleboro, VT 05301
http://cattailmusic.com
802/257-7391

"The first duty of government is to protect the powerless from the powerful."

–The Code of Hammurabi
The oldest extant legal code, Babylon, 1780 BC.

From: warrenell@msn.com
To: <rxc@nrc.gov>
Date: 11/29/05 11:35PM
Subject: Re: [c380] Portable Generator

OK Rex--
A few more:
Some have built small wood platform for gen to sit on, which solves problem of gen's black feet messing up deck. Instead I use small pieces of white "antiskid" material [rubber mesh]...same as used in galley to keep stuff from moving around...under each foot.

For security, I also tie gen to mast with short cable & lock--use same code as used for companionway hatch.

While on security....& perhaps aesthetics. buy silver weather cover for gen but cover it's bold/red-colored "HONDA" with silver duct tape.

Arrange gen athwartships, with plug outlets on stb side. This allows shortest cable run to transom [25 ft cable just makes it], and provides convenient location for pull/start cord.

When ready to use, simply pull up bottom elastic edge of weather cover until it's at/over the top [bungee and security ties to mast prevent complete cover removal, but no matter].

If dink stored on foredeck, be aware of exhaust heat exiting gen on port side. Move dink away, angle gen somewhat away [towards aft] and/or add heat protection to dink tube.

Close all downwind hatches/ portlights and dodger window while gvn operating.

There's a few more.

Warren

----- Original Message -----
From: Rex Langley<mailto:slangley999@cableone.net>
To: warrenell@msn.com<mailto:warrenell@msn.com>
Sent: Tuesday, November 29, 2005 6:15 PM
Subject: Re: [c380] Portable Generator

•
Warren,

Thank you for the informative reply. I would appreciate more 'pearls of wisdom' for securing the generator to the mast as you imply or suggest.

Rex
----- Original Message -----
From: Warren Elliott<mailto:warrenell@msn.com>
To: c380<mailto:c380@list.sailnet.net>
Sent: Monday, November 28, 2005 8:15 AM
Subject: Re: [c380] Portable Generator

Rex--Still more suggestions--

Definitely tie AC output of gen to your AC system, via plug at transom is easiest, as 95%+ of Honda's available output is 120VAC.

I and several in our All-Catalina group have these great generators. Mine is "installed" in front of mast, where it sits snugly between aft tubes of 10 ft dink when cruising. Gen has weather cover, and is tied to mast with heavy bungee. You can use std adapter and 50 ft dock cable to run power to transom plug. Simpler alternative is 25 ft, 20 amp cable [Home Depot $15] and adapter--much easier to handle. Also easy to lift cover, plug in cable, and start/run in-place. One downside: you do have to go forward to start/stop--in possibly poor weather.

Others store Honda in lazarette after draining gas or letting it run dry. Then lift [heavy] it out and set it under captains seat, w/short adapter lead to transom plug, and use it there.

Can suggest few further tips for mast location if you want.

Regards,
WArren
----- Original Message -----
From: Rex Langley<mailto:slangley999@cableone.net>
To: c380<mailto:c380@list.sailnet.net>
Sent: Sunday, November 27, 2005 9:47 PM
Subject: Re: [c380] Portable Generator

Completing my inquiry:
What is the easiest way to attach the generator to the electrical system? D.C. leads directly to battery terminals? or to cables coming from battery charger in lazarette?

Rex Langley
Sal Lee II 003
----- Original Message -----
From: "Rex Langley" <slangley999@cableone.net<mailto:slangley999@cableone.net>>
To: "c380" <c380@list.sailnet.net<mailto:c380@list.sailnet.net>>
Sent: Sunday, November 27, 2005 7:34 PM
Subject: [c380] Portable Generator

> For making an extended passage from Seattle, WA. to Ketchikan, AK., I want to following the advice of 380 owner, Tom Lincoln and place a Honda 2000 generator on board.
>
>
> ---
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ID: 446027 : rxc@nrc.gov : 20050124 12:42 : 20050124 12:43

From: "KATHY TORREY" <mstorrey@verizon.net>
To: <rxc@nrc.gov>
Date: 11/30/05 6:32AM
Subject: Vt Yankee neighbor

Dear Ralph,

I am a neighbor of the Vt Yankee plant in NH, in a town which was recently in national news because of a freak flooding accident. The way Alstead was affected during the flood made me even MORE sure that no evacuation plan could be foolproof enough to protect citizens from the hazards which could be possible with an uprate on an old system. I am getting older myself, and I can't suddenly run a marathon. I am adamantly opposed to increasing the workload of the Vt Yankee plant, and I appreciate your concern for the welfare of the neighbors- Kathy Torrey, Alstead, N.H. 03602

From: "Bargeron, Richard" <rbarger@entergy.com>
To: <rxc@nrc.gov>
Date: 11/30/05 8:22AM
Subject: Vermont Yankee Uprate

I favor of the approval of the VY Uprate.

Career anti-nuclear activists who insist on redundant testing, analysis and inspection, are simply trying to bleed the industry.

Unfortunately, the average person does not know much about how a nuclear plant operates but does have a vague notion that it might be dangerous. These people are easily stampeded by half-truths and misleading statements freely spewed by the likes of the New England Coalition. Fear is easily exploited.

But the carbon content of the atmosphere is increasing. Mankind's long-term existence on earth is in question unless we take responsibility for our situation and manage it with vigor and honesty.

One way to help our situation is to actively embrace nuclear power and devote our best efforts to making it succeed. The good people of Entergy Vermont Yankee have managed the operation, maintenance and renewal of the station safely for over 30 years. No one should doubt that this will continue in the future.

Thanks for taking time to read this.

Cordially,

Richard J. Bargeron
24 Rand Road
Shelburne Falls, MA 01370

Jonathan Crowell
POB 56
W. Dummerston, VT 05357

Dear Richard S. Denning,

My name is Jonathan Crowell and I'm from W. Dummerston, VT.

I'm writing to express my **disapproval for the uprate** of the Entergy Nuclear Vermont Yankee power facility. An uprate will increase the amount of waste and radiation and increase the threat to the safety of the workers and of the community.

I also demand a **truly Independent Safety Assessment** of the entire facility immediately.

Unfortunately, in my experience, short-term profit and free markets fundamentalism wins out over moral reasoning. In the absence of a responsible moral decision to shut down this facility, **I urge you to fully consider the true economic costs** of operating and expanding it. Let's look beyond the bottom line of shareholders and rate payers.

What are the true economic costs of **displacing indigenous people** and communities to make way for uranium mining and waste disposal?

What are the true costs of **increased cancer rates**, increased birth defects and depleted immune systems, etc.? What is the true cost of being exposed to depleted uranium?

What are the costs upon **future generations** who are forced to manage these dangerous materials for 250,000 years?

What are the costs of doing irreversible **damage to the earth** and its vital ecological life support systems?

What are the opportunity costs of not transitioning to healthy and **sustainable energy** sources now? I assure you that the costs will be enormously higher if we wait until the point of desperation.

What are the costs of **defending these dangerous materials** from natural disasters and terrorist plots? How much have taxpayers lost in subsidizing the nuclear industry through research, security, maintenance, treatment of waste, treatment of water, etc.?

What are the costs of **losing a few fuel rods**? With obvious incompetence on the federal level, what are the costs of being ill-equipped to deal with a potential disaster? What are the costs of a possible **refugee scenario** in New England?

With the increase in the price of oil and the cost of shipping the **waste** across the country, what is the true cost of potentially getting stuck with this waste here in Vermont?

What are the psychological costs of Vermont Yankee **terrorizing** thousands of people in the Brattleboro area? What is the cost of **living in fear** of losing your life, family and home?

What is the cost of railroading this **scheme for profit**, and of the disempowering impacts upon the people of this region? What is the cost to **democracy** and to the **sovereignty** of the people of New England?

What are the economic effects of tolerating and perpetuating this type of **corporate abuse** and **environmental racism**? What are the true consequences of an entire community being **manipulated** by the government and huge energy companies? What is the cost of losing faith in our leaders/decision-makers?

What are the costs of making poor choices- choices that put profit before people?

I don't believe these costs are factored into the costs of electricity. At times, we pay twice- as rate payers and as tax payers. **It is time to take into consideration all of the externalized costs.** The cost of a thorough and independent safety assessment really seems insignificant, so make it happen. Please use your full capacity to make history and deny the uprate!

Thanks and I look forward to a timely response,
Jonathan Crowell
J Crowell

PS- Shut down the Entergy Nuclear Vermont Yankee power facility, now!

From: <jbrianc@adelphia.net>
To: <rxc@nrc.gov>
Date: 12/2/05 11:54AM
Subject: Support Vermont Yankee's 20% Power Uprate!

December 2, 2005

Mr. Ralph Caruso

,

Dear Mr. Caruso:

As a native Vermonter and a Brattleboro resident for the past eight years, I want to write in support of the proposed 20% power uprate at Vermont Yankee.

This plant has been a safe and reliable source of power for the citizens of Vermont since 1972 and will become an even more vital source of electricity in the future.

Growing usage and the increasing difficulty in siting new sources of generation make existing assets like Vermont Yankee more valuable to the public with every year that passes. The additional 100 megawatts that the propose uprate will provide will be enough to serve 100,000 homes here in Vermont and New England.

The uprate has undergone a thorough analysis by the Vermont Public Service Board and by the NRC over the past two years. Now is the time to act.

Brian Cosgrove

Sincerely,

Brian Cosgrove
60 Brookside Dr Unit 5
Brattleboro, VT 05301-4282

New England Coalition

VT . NH . ME . MA .

RI . CT . NY -

POST OFFICE BOX 545, BRATTLEBORO, VERMONT o5302

November 21, 2005
By E-mail & U.S. Mail

Dr. Graham Wallis, Chairman,
Committee Members
Advisory Committee On Reactor Safeguards
U.S. Nuclear Regulatory Commission
Mail Stop T2E26
Washington, DC

Dear Dr. Wallis,

This letter is to follow on comments that I made on behalf of the New England Coalition November 16, 2005 before the Sub-Committee on Thermal-Hydraulic Phenomena in Brattleboro, Vermont.

In my comments I touched two main topic areas with respect to the proposed Vermont Yankee (Docket 50-271) extended power uprate:

- credit for containment over-pressure in order to maintain emergency core cooling, and
- the comparative qualities of the NRC team engineering inspection v. the deep-slice (diagnostic) assessment requested by the Vermont Public Service Board.

With respect to the first topic area, as promised I am express mailing a copy of the 1986 Vermont Yankee Containment Safety Study, transmitted from VYNPC to NRC (J.G. Weigland to Mr. Harold Denton), September 2, 1986. I am providing below (and in advance via e-mail) excerpted portions of the study regarding the functional interdependence of safety systems:

- containment, vents, and emergency pumps,
- containment, cooling spray, and emergency pumps.

I'm sorry that your committee was not provided a copy of this report sooner. As you know, I referenced the study in my comments to your committee meeting on Regulatory Guide 1.82 (proposed Revision 4) on July 19, 2005. I received a telephone call from the NRC VY Project manager, Rick Ennis, on July 20, 2005. He told me that he had a copy of the study in hand and asked that I give specific citations on pump NPSH and containment pressure, which I did. I am not sure why I thought that he would pass the study and the citations on to the ACRS. It is clear now that he did not and that NRC staff has not acted on the information provided.

As I stated in my comments of November 16, 2005, while the licensee has now provided information that is contradictory of the study (i.e., claiming both that containment over-pressure should be credited and that it is, in realistic space, unnecessary), we can find no licensee submittal directly addressing the study and refuting its assumptions, underlying calculations or conclusions.

The following study excerpts clearly show the precariousness of the reliability and availability of various safety systems when consideration of amending containment pressure is in play.

The first excerpt is somewhat of a departure, but it is included to show that operators must weigh numerous competing considerations, including secondary v. primary containment pressure. Radiological considerations dictate early rather than late venting.

However, if venting is overdone and/or accompanied by spray and/or secondary containment pressure is elevated (due, say, to a steam line break), the study appears to indicate that only a relatively small negative pressure (2 psig) or a slightly larger pressure differential is needed to implode or crush the primary containment.

If operators wait until late in an accident when emergency pumps are needed, then it appears that use of venting and containment spray are precluded.

Excerpts/ Vermont Yankee Containment Safety Study – 1986

5.3 Drywell Spray Capability

5.3.3 Identified Issues

> ...The first issue is the chance of containment implosion...Design negative pressure of 2 psig will not be exceeded provided that vacuum breakers operate as designed.
>
> Secondly, ECCS pump NPSH is a concern, as is the case with containment venting (Section 5.4). If sprays are utilized when containment is pressurized and torus water temperature is elevated, the resultant depressurization could impact the available NPSH of pumps taking suction from the torus.

5.4 Severe Accident Containment Failure

> NRC believes that containment venting should be available to avert uncontrolled over-pressure failure of the containment in certain severe accident scenarios.

5.4.3.2 ATWS Venting

...postulated to relieve pressure and preclude failure of the drywell shell leading to an ATWS success path. However, containment venting may also jeopardize continued core cooling. In this scenario the pressure suppression pool would quickly become saturated and would boil if pressures were significantly reduced.

The operability of the reactor vessel injection system pumps that take suction from the pressure suppression pool would be compromised due to inadequate net positive suction head and resultant pump cavitation. If these injection systems are the only ones available, the degradation or failure of pumping capability could lead to core uncovery and core melt might actually be caused by wetwell venting.

5.6.3.1 Competing Safety Requirements

...competing requirements that pertain to safe operation of plant systems. For example, use of containment spray to preclude over pressurization of the primary containment may be indicated symptomatically. However, under certain conditions, this action could itself precipitate containment failure.

5.6.5.1.3 Potential Negative aspects Relative to Overall Safety

Containment venting (regardless of the offsite radioactive release rate) at decay heat mass flow rates, might also result in a loss of NPSH for ECCS pumps taking suction from the suppression pool and excessive control room radiation levels. Therefore, under certain circumstances, such procedure changes may have the effect of worsening the consequences of a severe accident.

The licensee should not be permitted to proceed with uprate, nor should the staff be permitted to approve the uprate license amendment until the licensee statements in the containment safety study and those in the application, together with those given in testimony before the ACRS, are reconciled.

The licensee should not be permitted to proceed with uprate, nor should the staff be permitted to approve the uprate license amendment until adequate assurance of public health and safety can be restored through independent confirmation that reliable, available, independent, and redundant safety systems are not compromised by the additional heat, flow, and volumes inherent in an extended power uprate.

In our view, NRC staff should now be required to explain why, as evidenced in this report, it permitted Vermont Yankee to operate with identified, but unanalyzed safety issues for many years. To that end, New England Coalition intends to file in the near term a petition for NRC enforcement action under 10 CFR 2.206.

Further, I plan to travel to NRC headquarters for the ACRS Subcommittee meetings on November 28 and 29, 2005 and there I hope to provide further comment on the engineering team inspection, the need for full transient testing, and on the NRC draft Safety Evaluation Report, generally.

Additional written material regarding the engineering team inspection question is in preparation to be forwarded to ACRS within the next few days.

New England Coalition notes with some dismay that as of November 16, 2005, the ACRS had not received the licensee's most recent application supplements and yet NRC apparently expected ACRS to complete review by December 7, 2005. It is most inappropriate and unprofessional that a unilaterally placed (and we believe, unduly influenced) deadline should drive the schedule of a technical nuclear safety review. Please do not allow NRC commission and management to put a rush on an ACRS review to the detriment of assuring public health and safety when so many basic and important issues are at stake.

On behalf on New England Coalition, thank you for your attention and attentiveness in hearing the concerns of area residents during the recent Brattleboro meetings.

Sincerely,

Raymond Shadis
Staff Technical Advisor
New England Coalition
Post Office Box 98
Edgecomb, Maine 04556
207-882-7801
shadis@prexar,com

From: T. Milne <tm11n5x@crocker.com>
To: <RXC@nrc.gov>
Date: 11/30/05 3:15PM
Subject: licensing

Vermont Yankee nuclear power plant has had its day and now needs to be retired and cleaned up . Relicensing and upgrading the plant is dangerous policy. Why are we looking toward the past -- nuclear power --instead of the future -- solar and wind and other alternative sources of power which are far safer? T. Milne

****

"The Pharaoh's Fingerprint"a strange coincidence singles out a forgotten orphan boy; one man thinks the coincidence too small to share, another sees it as an opportunity to improve the boy's life -- and his actions have unforeseen consequences for himself, as well.. "London for the Season," a rollicking Regency romance, plus the sequel, "A Christmas Wedding" and last in the series, "What About Prudence?" "Richard Rearranged" is a rewrite of Shakespeare's "Richard III" in four acts. Shakespeare based his play on Tudor propaganda. Richard was not a villain.

Pittenbruach Press "Take Heed, Dear Friends"
PO Box 553 music (to remember them by)
Northampton MA 01061 for the Advices & Queries.
"Thumbs Up" hitchhiking
adventures in Europe, 1950s
Email us for a complete listing of books. Or see us at www.teddymilne.us

Ralph Caruso,
U.S. Nuclear Regulatory Commission
Advisory Committee on Reactor Safeguards

December 3, 2005

Dear Mr. Caruso

The proposal to increase power by 20 percent at Vermont Yankee will have a strong and positive economic impact on the state of Vermont, thus I lend my support to this request.

As a participant in the process of building economic development opportunity, I recognize and understand the critical role energy plays in maintaining productivity, efficiency, and revenue for businesses. Without affordable power and electricity, the challenge to compete becomes very difficult.

From a business perspective, we need more power to meet the needs of consumers, and a more diversified energy source portfolio to provide stability and security.

Economically, the state of Vermont will gain from the uprate; money of which can be used for any number of programs that will help foster, promote and sustain a high quality of life for Vermonters.

This power increase is a good thing for Vermont and I lend my complete support to the approval of this project.

Thank you.

Tim Smith
Executive Director
Franklin County Industrial Development Corporation

From: "Brad Ferland" <bferland@together.net>
To: <rxc@nrc.gov>
Date: 12/6/05 8:31AM
Subject: Vermont Yankee Power uprate

Ralph Caruso,

U.S. Nuclear Regulatory Commission

Advisory Committee on Reactor Safeguards December 4, 2005

Dear Mr. Caruso

I am writing to unequivocally support the uprate at Vermont Yankee. This is a key power increase in a region desperate for the addition.

ISO - New England has stated the potential for blackouts this winter, as well as the pressing need for the region to increase its power production to meet growing demand in the coming years. The added power at Vermont Yankee will provide stability to the grid which will deter the necessity of implementing blackouts in Vermont and throughout New England.

This region relies heavily on natural gas and the supply shortages have made it imperative that we utilize all possible resources. Right now an increase of power at the Vermont Yankee nuclear plant is the most sensible and feasible option there is. As a native Vermonter, I for one, appreciate having Vermont Yankee as a valuable resource for our State.

I appreciate your diligent review process and hope you support the uprate at Vermont Yankee, so it can help New England meet its growing demands in the immediate and near future.

Sincerely,

Brad Ferland

Vermont Energy Partnership

Board Member

Ralph Caruso,
U.S. Nuclear Regulatory Commission
Advisory Committee on Reactor Safeguards

December 2, 2005

Dear Mr. Caruso

I am strongly in favor of the Vermont Yankee uprate, because of the pressing need for energy sources in New England that are not harmful to the environment.

Vermonters and other residents of New England take pride that their environments are clean and beautiful, and it is no secret that the region requires more power to meet its ever growing demand. However we must not compromise one for the other.

If we are to maintain our pristine landscapes and increase power production, then we must turn to plants capable of achieving both conditions.

Vermont Yankee is such a plant. It provides base-load power (crucial for meeting peak demand periods), and is non-emitting.

We must face the truth the region is not developing new major generating plants that would be able to meet demand. Our only logical solution is to increase production from those existing plants that are able to do so safely, while maintaining the values New Englanders place on the environment.

I support the uprate at Vermont Yankee, and trust the necessity of this is seen.

Sincerely,

George Nelson
Vermont Energy Partnership member
Champlain Water District

ACRS Full Committee

Vermont Yankee Extended Power Uprate Containment Overpressure Credit

December 7, 2005

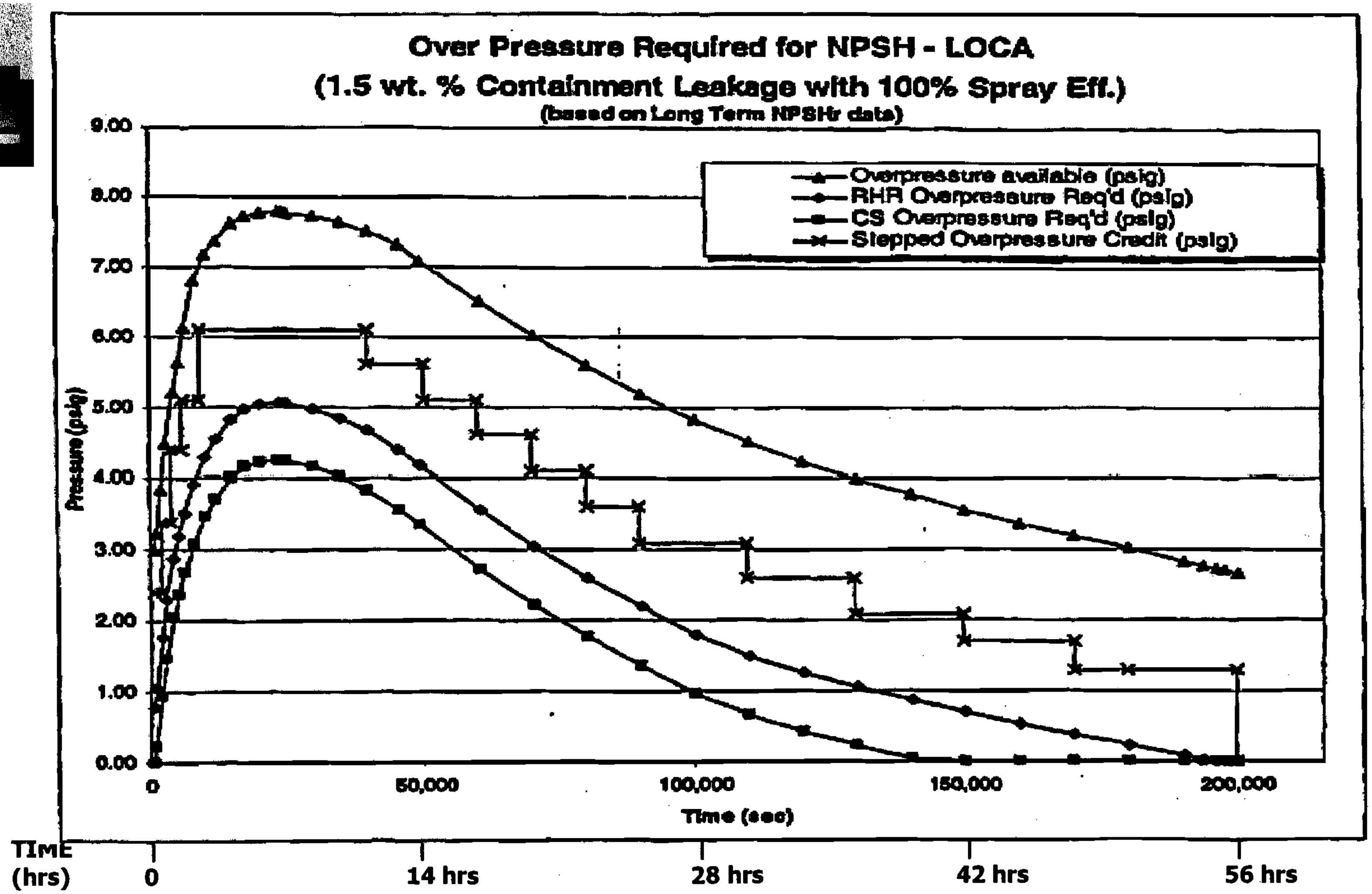
Bill Sherman – VT Dept of Public Service

5

Proposal Does Not Appear to Meet ACRS Letter September 20, 2005

- Longer than a few hours
- There are practical alternatives
- There is not full positive indication of containment integrity
- Containment integrity has not been demonstrated for the credited time period

Figure 4.2 LOCA – Long Term (1.5 wt. % Containment Leakage & 100% Spray Efficiency)



Staff Response to ACRS Letter

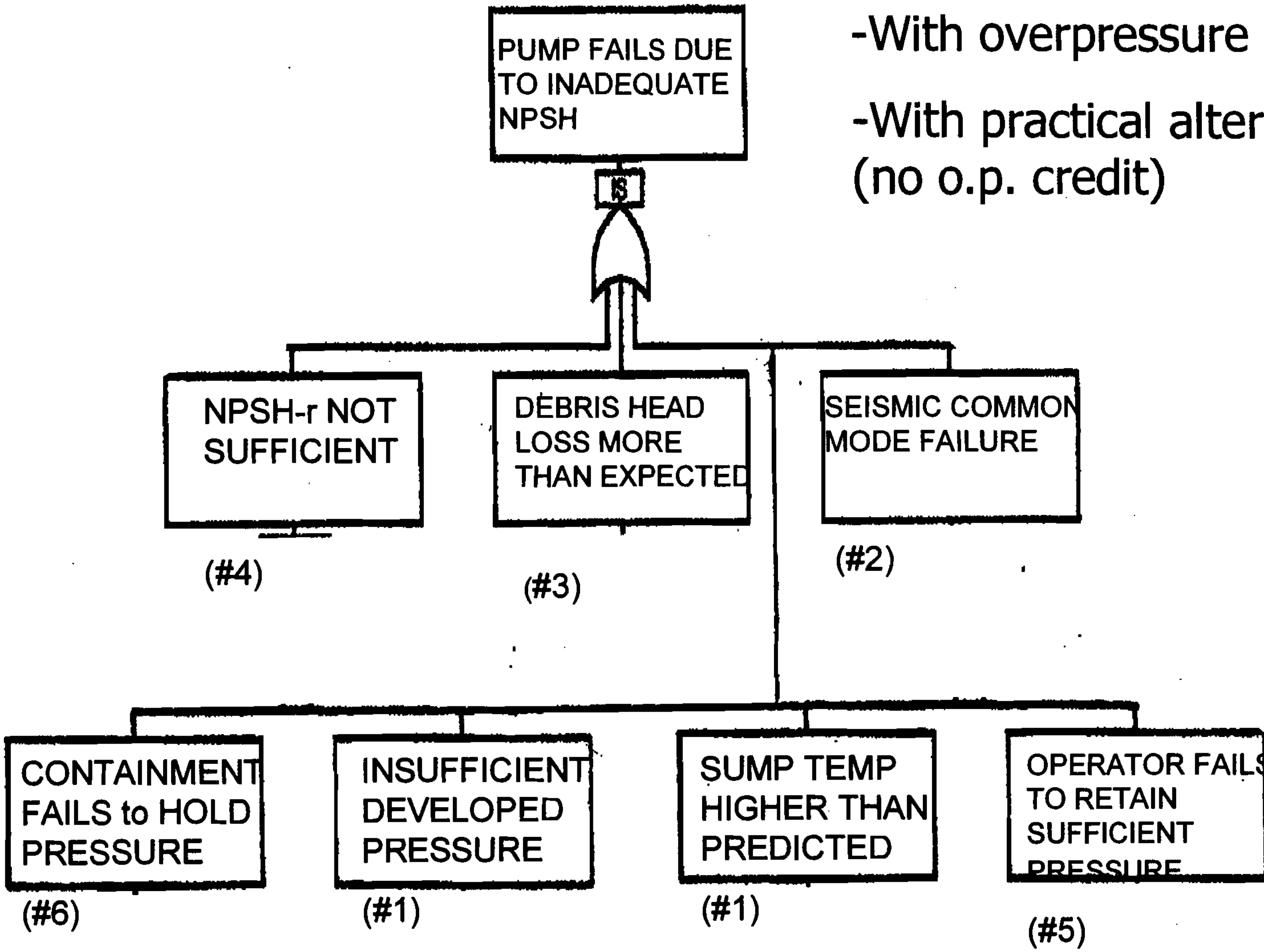
- On Oct 7, 2005, Dr. Brian Sheron proposed a risk based (RG 1.174) approach in lieu of implementing practical alternatives
- We believe this approach may have promise
- Entergy analyzed part of the problem, but not the whole problem. An analysis of the whole problem would shed light on the risk of the overpressure proposal

NEW TOP EVENT

Two Cases:

-With overpressure credit

-With practical alternative (no o.p. credit)



NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007

Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
VERMONT YANKEE NUCLEAR POWER STATION (VY)	05000 271	1 OF 3

4. TITLE
Primary Containment Leak Rate Testing Program Second Barrier Isolation Valve Found Miss-Positioned

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									N/A	
10	04	2005	2005	- 002 -	00	12	01	2005	N/A	

9. OPERATING MODE: N

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

CONTACT NAME	TELEPHONE NUMBER (Include Area Code)
William F. Maguire, General Manager Plant Operations	(802) 257-7711

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
N/A					N/A				

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On 10/04/05, with the reactor at full power, a sample line isolation valve, V10-198A on the Residual Heat Removal (RHR) "A" Loop was found open. V10-198A is a second barrier that supports Primary Containment Integrity and is required to be closed per the Primary Containment Leakage Rate Testing Program (PCLRTP). Upon discovery, Operators closed V10-198A and placed it under administrative control. This condition was identified while reviewing a Safety Classification Worksheet for a different valve in the RHR system sample line. The RHR System procedure valve line-up listed V10-198A as "open" and the RHR System Piping and Instrumentation Diagram (P&ID) displayed it as "closed". The open valve provided a potential flow path of water from Primary Containment to Secondary Containment. Two air operated valves and a manual sample valve located downstream of V10-198A provided reasonable assurance that effective isolation for this flow path was maintained during plant operation. The cause of this condition was the application of an insufficient change process in 1996 during implementation of the Qualified Closed Loop Outside Primary Containment modification that lacked sufficient documentation and reviews to effectively implement the change. There was no significant increase in radiological risk to plant workers or the public as a result of this condition.

NRC FORM 366 (6-2004) PRINTED ON RECYCLED PAPER

Example this week of a containment isolation valve left open for 10 years

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007
LICENSEE EVENT REPORT (LER)	Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
VERMONT YANKEE NUCLEAR POWER STATION (VY)	05000 271	1 OF 3

4. TITLE
Primary Containment Leak Rate Testing Program Second Barrier Isolation Valve Found Miss-Positioned

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME: N/A	DOCKET NUMBER
10	04	2005	2005	- 002 -	00	12	01	2005	FACILITY NAME: N/A	DOCKET NUMBER

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
N	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
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10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
100	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A
	☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

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N/A					N/A				

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ◉ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On 10/04/05, with the reactor at full power, a sample line isolation valve, V10-198A on the Residual Heat Removal (RHR) "A" Loop was found open. V10-198A is a second barrier that supports Primary Containment integrity and is required to be closed per the Primary Containment Leakage Rate Testing Program (PCLRTP). Upon discovery, Operators closed V10-198A and placed it under administrative control. This condition was identified while reviewing a Safety Classification Worksheet for a different valve in the RHR system sample line. The RHR System procedure valve line-up listed V10-198A as "open" and the RHR System Piping and Instrumentation Diagram (P&ID) displayed it as "closed". The open valve provided a potential flow path of water from Primary Containment to Secondary Containment. Two air operated valves and a manual sample valve located downstream of V10-198A provided reasonable assurance that effective isolation for this flow path was maintained during plant operation. The cause of this condition was the application of an insufficient change process in 1996 during implementation of the Qualified Closed Loop Outside Primary Containment modification that lacked sufficient documentation and reviews to effectively implement the change. There was no significant increase in radiological risk to plant workers or the public as a result of this condition.

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
VERMONT YANKEE NUCLEAR POWER STATION (VY)	05000 271	2005	– 002 –	00	2 OF 3

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

DESCRIPTION

On 10/04/05, with the reactor at full power, a 3/4" manual globe valve for the Residual Heat Removal (RHR) "A" Loop sample line (V10-198A) was found open. V10-198A is a second barrier to Primary Containment and is required to be closed as necessary to maintain the RHR system water seal during plant operation per the Primary Containment Leakage Rate Testing Program (PCLRTP). This condition was discovered while reviewing a Safety Classification Worksheet for a different valve in the RHR sample line. The RHR System procedure valve line-up listed V10-198A as "open" and the RHR System Piping and Instrumentation Diagram (P&ID) displayed the valve as "closed". The open valve provided a potential flow path of water from Primary Containment to Secondary Containment for the water seal that serves as part of the second barrier for Primary Containment, during a Design Basis Loss of Coolant Accident with a concurrent seismic event.

Upon discovery of this condition, Operators closed V10-198A and placed it under administrative control by tagging the valve "SHUT". Two normally closed air operated valves (AOV) and a normally closed manual sample valve located downstream of V10-198A provided reasonable assurance that effective isolation for this flow path was maintained during plant operation. The RHR "keep fill" line maintains system pressure during normal operation to continuously demonstrate that Primary Containment Integrity is maintained. Any leakage through the series of closed valves would have been into the Reactor Building Sample Sink which is within the envelope of Secondary Containment and would be detected by Operations or Chemistry personnel.

The three valves located downstream of V10-198A are not credited as Primary Containment Isolation valves within the Program Procedure for the PCLRTP. However, both of the in-line AOVs close on a Primary Containment Isolation System (PCIS) signal and are designed with a fail-safe feature to close on a loss of instrument air. The manual sample valve located downstream of the PCIS valves is also maintained in the closed position. Additionally, the first AOV in the series, FCV10-160, is designed to perform during and after a design bases seismic event.

This condition was determined to be reportable to the NRC as a Condition Prohibited by Technical Specifications in accordance with 10CFR50.73(a)(2)(i)(B). VY Technical Specification 3.7.A.2 states that Primary Containment integrity shall be maintained at all times when the reactor is critical. Technical Specification 4.7.A.2 provides a surveillance requirement to ensure that this is accomplished by stating that Primary Containment integrity shall be demonstrated by the PCLRTP. Also, Technical Specification Definition 1.0.N. for Primary Containment Integrity states that all manual containment isolation valves that are not required to be open during accident conditions, are closed, and may be opened intermittently under administrative controls. V10-198A was not in the required closed position prior to discovery of this condition and was not administratively controlled open by a dedicated operator.

CAUSE

The root cause of this condition was determined to be the application of an insufficient change process (Job Order File process) that was utilized in October of 1996 during implementation of the Qualified Closed Loop Outside Primary Containment modification. The process that was used lacked sufficient documentation and reviews to effectively implement the change.

Contributing Causes included the following;

1) The inter-relationships between P&ID valve position, operating procedure valve position, and locked valve criteria were not well understood when the event occurred.

2) Thirty five successive revisions up to 1989 to the subject P&ID reduced the sharpness of the image quality for V10-198A causing the valve's normal "open" position to appear as "closed".

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
VERMONT YANKEE NUCLEAR POWER STATION (VY)	05000 271	2005	-- 002 --	00	3 OF 3

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

ASSESSMENT OF SAFETY CONSEQUENCES

The subject valve is a second barrier for Primary Containment per the PCLRTP. The first barrier valves remained operable and closed as required. There are two AOVs located downstream of V10-198A that are designed to automatically close on a PCIS signal or on a loss of instrument air. A third manually operated sampling isolation valve located downstream of the two AOVs is maintained in the closed position. Additionally, incidental leakage from the system past these three valves would be detected by Operations or Chemistry personnel at the Reactor Building Sample Sink. Therefore, reasonable assurance existed that Primary Containment Integrity was maintained. This condition did not result in a significant increase in radiological risk or industrial risk to plant workers or the general public in the event of a design bases accident.

CORRECTIVE ACTIONS

The Job Order File process that was used when this condition occurred was superseded by an improved design control process. The procedures that implement the current design control process provide clearer and more concise direction that would likely have prevented this condition from occurring if utilized in 1996.

Immediate Actions

1) Upon discovery and confirmation of this condition, V10-198A was closed and administratively tagged "SHUT".

Interim Actions

1) V10-198A was added to the "Current System Valve and Breaker Line Up and Identification" procedure controlled population.
2) A drawing change was submitted for the subject P&ID to indicate V10-198A/B normal positions as locked closed.
3) The RHR System procedure's appendix for normal system line up was changed to control V10-198A as closed.
4) The RHR and Core Spray system procedures were verified to ensure that the valve line-ups contained within them are in agreement with the procedure for the PCLRTP. No additional discrepancies were noted.
5) On November 10, 2005, the Vice President of Engineering distributed a memo to all Vermont Yankee site employees titled "Configuration Control at Vermont Yankee". This correspondence described the event, expectations for configuration control, current design control processes employed within the Entergy Fleet, provided a list reference materials and described the relevant points from the reference materials that need to be reinforced to prevent this type of event from recurring.
6) Radiation Protection containment sampling procedures were reviewed to ensure compliance with TS 1.0.N.1 and the PCLRTP Procedure administrative controls for manual containment isolation valves. No discrepancies were noted.

Long Term Actions

1) A review of other Job Order File changes from the same time frame will be performed to assess the potential for similar conditions.
2) Evaluate the need to review and as necessary correct the image quality and valve positions for the Control Room P&IDs referenced in the PCLRTP procedure.

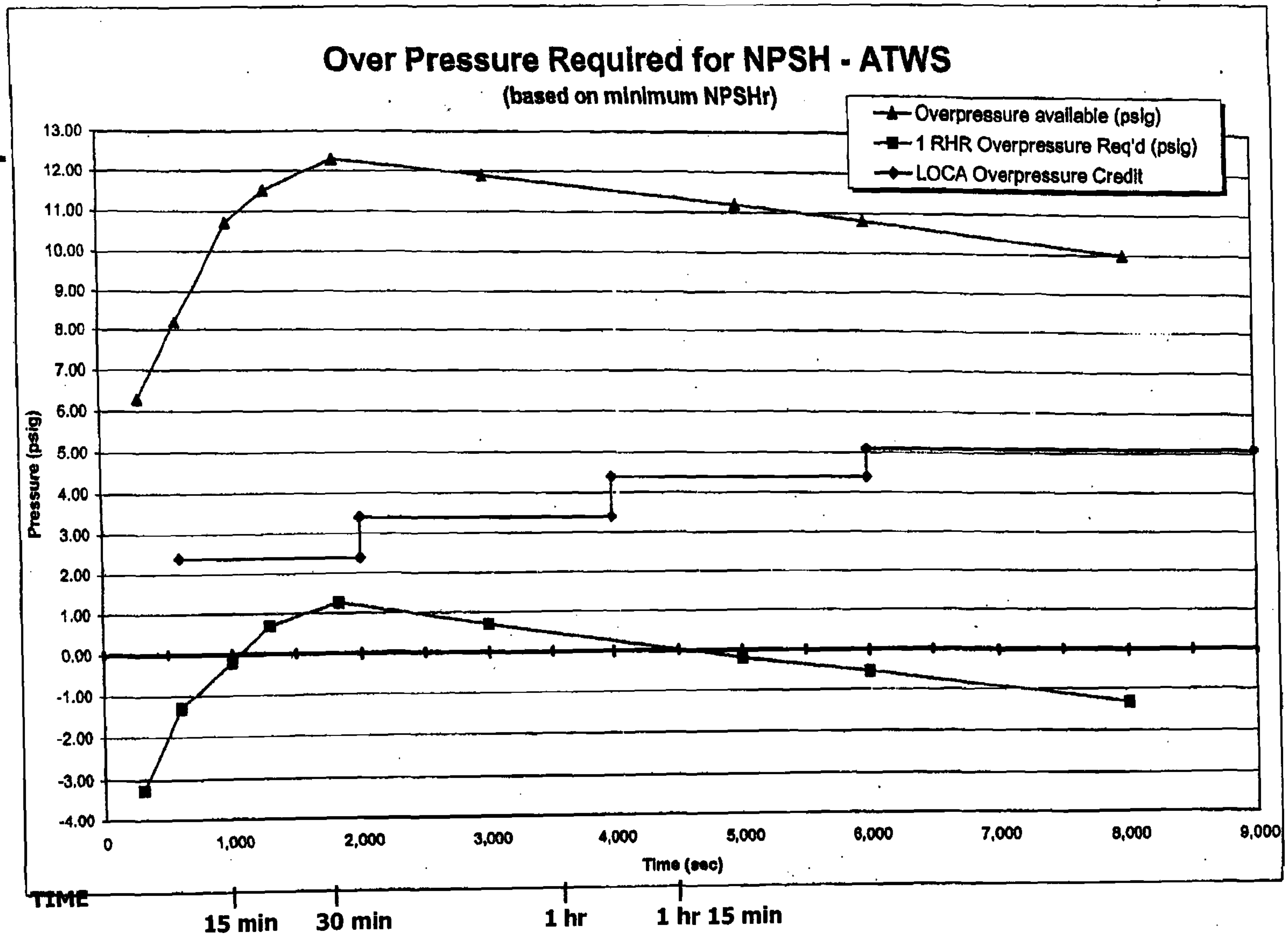
ADDITIONAL INFORMATION

No similar events have occurred at Vermont Yankee within the past ten years.

ATWS NPSH Evaluation Deserves More Questions

- Asking overpressure credit for 1.25 hours
- More energy in ATWS than LOCA
- Pressure developed differently
- The type of conservative assumptions used for LOCA are not employed for ATWS

Figure 4.3-1


Over Pressure Required for NPSH - ATWS
(based on minimum NPSHr)
Overpressure available (psig)
1 RHR Overpressure Req'd (psig)
LOCA Overpressure Credit
Pressure (psig)
13.00
12.00
11.00
10.00
9.00
8.00
7.00
6.00
5.00
4.00
3.00
2.00
1.00
0.00
-1.00
-2.00
-3.00
-4.00
0
1,000
2,000
3,000
4,000
5,000
6,000
7,000
8,000
9,000
Time (sec)
TIME
15 min
30 min
1 hr
1 hr 15 min

SUMMARY

- Under the ACRS Letter of 9-20-05, overpressure credit should not be granted.
- Under Dr. Sheron's proposal:

1. Modification of the defense-in-depth concept is troublesome
2. Entergy analyzed part of the problem, but not the whole problem. An analysis of the whole problem would shed light on the risk of the overpressure proposal.

Pilot Engineering and Design Inspection at Vermont Yankee

Raymond Shadis

New England Coalition

BEFORE A MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

December 7, 2005

Rockville, Maryland

6

COMPARING NRC INSPECTION AND IEA

VPSB REQUESTED IEA

- deep vertical slice of 4 systems
- 2 safety systems
- 2 maintenance rule systems
- implies EOC review
- level of effort – 4 persons/4 weeks/40 hours or 640 dedicated hours
- independent

NRC DID PILOT ETI

- 45 components and operations of high risk/low margin
- eight issues were found
- limited extent of condition review
- 910 hours-500 routine + 410 pilot
- independent (2 yr)

COMPARING GOALS AND RESULTS

VPSB

- INDICATORS OF RELIABILITY
- ADDRESS PUBLIC CONCERN FOR SAFETY AND CALLS FOR AN ISA

NRC

- QUALITY OF ENGINEERING
- CONFIRM DESIGN AND OPERATION
- FIND AND FIX
- DID NOT PROBE AREAS WHERE PROBLEMS HAVE OCCURRED AT VY & OTHER EPU PLANTS

Staff Conclusions – Secy-05-0118

- NRC staff reported- pilot inspection to be improvement over ROP periodic inspection -found 8 issues per 1000 hours , old program found 4.5 per 1000 hours.
- Staff recommends engineering and design basis inspection be added to EPU review
- Staff found eight safety issues in 45 components/actions examined-small sample, large yield
- Staff stated that VY issues would not have been found in routine inspection

Comparing Pilot Inspection and Diagnostic Evaluation Team or ISA

PILOT INSPECTION

- small biased sample-high risk/low margin
- some focus on epu
- team members unfamiliar with epu and/or vy
- limited extentof condition review

DIAGNOSTIC EVAL.

- large sample by system type
- team members experienced in diagnostic evaluation
- large horizontal component- extent of condition/cause

Comparing Pilot Inspection and Diagnostic Evaluation Team or ISA...continued

PILOT INSPECTION

- Independence was marked by 2 year seperation from Vermont Yankee and Entergy

DET or ISA

- Maine Yankee ISA excluded NRC personnel from Region I and the Office of Nuclear Reactor Regulation

Entergy

Presentation to the Advisory Committee on Reactor Safeguards

Early Site Permit Application for Grand Gulf Nuclear Station

December 8, 2005

Entergy / System Energy Resources, Inc.

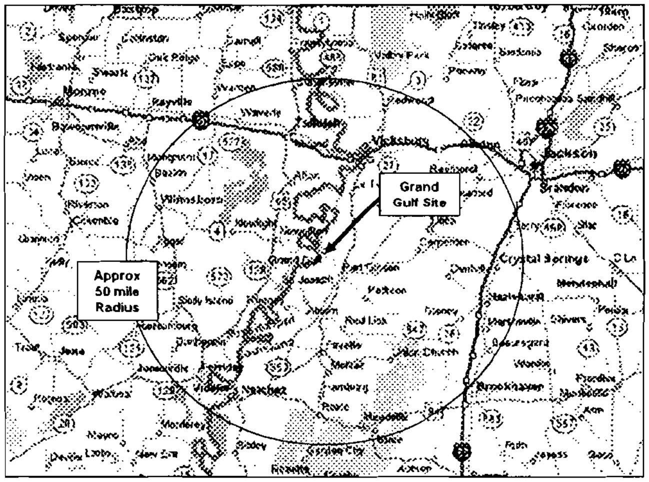
Entergy

Entergy ESP Team

- Entergy
  - C R Hutchinson, Vice President, Business Development
  - Kenneth Hughey, Michael Bourgeois, George Zinke
- Enercon Services
  - Al Schneider, Guy Cesare
- William Lettis & Associates
  - William Lettis

Entergy

General Information



Rev 5 - 12/05/05 5

Entergy

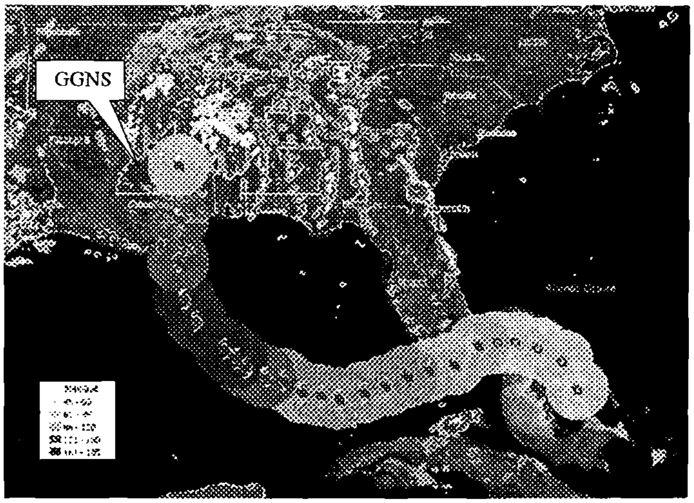
General Information

- Proposed ESP facility, located on existing Grand Gulf Nuclear Station Site
- GGNS Unit 1 licensed June 1982
- Site Owner and ESP Applicant - System Energy Resources, Inc (SERI)
- GGNS Unit 1 Operator - Entergy Operations
- ESP Application Preparer - Entergy Nuclear Potomac, Inc.

Rev 5 - 12/05/05 6

Entergy

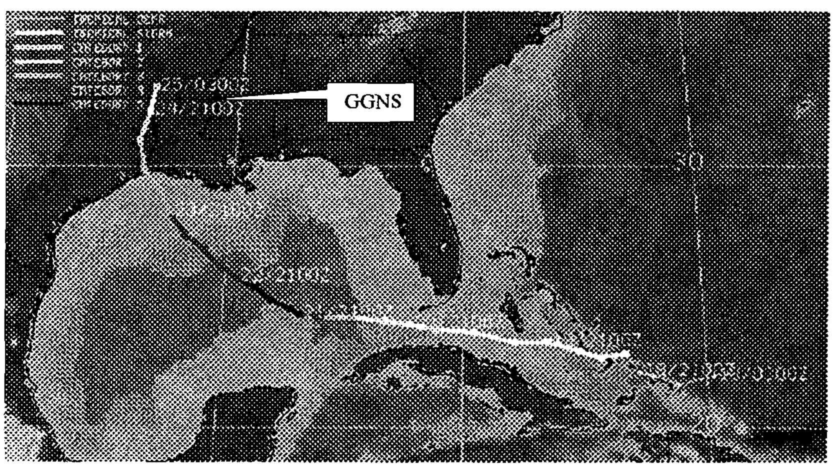
Hurricane Katrina



Rev 5 - 12/05/05 REF: http://gos2.geodata.gov/wps/portal/gos/communities/katrina 13

Entergy

Hurricane Rita



Rev 5 - 12/05/05 REF: http://www.ncdc.noaa.gov/oa/climate/research/2005/rita.html 14

Entergy

Recent Hurricane Experience

- SUMMARY
  - Site values based on industry standards for region
  - Recent hurricane experience (wind gusts and precipitation rates) appear bounded by established site values
  - Process at COLA development will evaluate new and significant information
  - Consideration of selected reactor design and margins

Rev 5 - 12/05/05 17

Entergy

ACRS Interim Letter Items

- Hazards
  - Ground & River Transportation Hazards
- Local Onsite Flooding
  - Permit Condition

Rev 5 - 12/05/05 18

Presentation to the Advisory Committee on Reactor Safeguards on the Early Site Permit Application for the Grand Gulf Site

U.S. Nuclear Regulatory Commission

December 8, 2005

Presented by: Raj Anand

NRPB/DNRL/NRR

8

Purpose

To provide the ACRS an overview of the Grand Gulf early site permit (ESP) safety review

Meeting Agenda

Grand Gulf ESP Safety Review	5 min
Key Review Areas	5 min
Permit Conditions and COL Action Items	5 min
Project Milestones and Summary	5 min
Questions or Comments	5 min

Grand Gulf ESP Safety Review

- The Final SER documents the staff's technical review of the applicant's site safety analysis report and emergency planning information

- SERI requests ESP site be approved for total nuclear generating capacity of up to 8600 MWt, with max 4300 MWt per unit

- SERI has chosen not to submit a specific design but instead has submitted a plant parameter envelope (PPE) based on a number of current and future reactor design

Grand Gulf ESP Safety Review

- The staff's review concluded that:
  - Subpart A of 10 CFR Part 52 and 10 CFR Part 100 have met
  - The exclusion area is acceptable and meets Part 100
  - The proposed site is acceptable for constructing a plant falling within the plant parameter envelop with respect to:
    - Radiological effluent release dose consequences from normal operation
    - Aircraft hazards
    - Physical security plans
  - No physical characteristics unique to the site that could pose a significant impediment to the development of emergency plans

Grand Gulf ESP Safety Review

- The staff's review also concluded that:
  - An acceptable description of current and projected population densities in and around the site
  - Potential hazards associated with nearby transportation routes, industrial and military facilities pose no undue risk to the facility
  - Site characteristics related to climatology and the methodologies used to determine the severity of the weather phenomena reflected in these site characteristics are acceptable and contain sufficient margin

Grand Gulf ESP Safety Review

- The staff's review also concluded that:

  - Applicant's proposed site characteristics related to hydrology are acceptable with the noted permit condition and COL action items

  - The site is acceptable from a geology and seismology standpoint and meets the requirements of 10 CFR 100.23

  - The applicant has provided appropriate quality assurance measures equivalent to those required by 10 CFR Part 50 Appendix B

Key Review Areas

- **The staff completed its review in the following areas:**
  - **Exclusion Area Authority and Control (1)**
  - **Population Distribution (1)**
  - Nearby Industrial, Transportation, and Military Facilities
  - **Meteorology (5)**
  - **Hydrology (7)**
  - **Seismology and Geology (5)**
  - Radiological Effluent Release Dose Consequences from Normal Operations

Key Review Areas (Continued)

- Aircraft Hazards
- **Emergency Planning (4)**
- Industrial Security
- Accident Analyses
- Quality Assurance

> **There were 23 Open Items in the Draft SER**
> **Resolution of all Open Items discussed in the Final SER**

Proposed Permit Conditions and COL Action Items

- There are 3 proposed Permit Conditions in the Final SER (10 in the Draft SER)

- There are 26 COL Action Items in the Final SER (18 COL Action Items in the Draft SER)

Proposed Permit Conditions

1. Obtain and execute agreements providing for shared control of the Grand Gulf ESP exclusion area, including State approvals before construction begins under a CP or COL referencing the ESP

2. Requires the new units radwaste systems be designed with features to preclude any and all accidental releases of radionuclides into any potential liquid pathway

3. Perform geologic mapping of future excavations for safety-related facilities

COL Action Items

- COL Action Items included to:
  - Ensure that significant issues are tracked and considered during the COL phase
  - Identify issues that shall be addressed by an applicant who submits an application referencing the Grand Gulf ESP

Project Milestones

- Received Grand Gulf ESP application – October 16, 2003
- Draft SER issued – April 7, 2005
- ACRS Subcommittee on Draft SER – May 16, 2005
- ACRS Full Committee on Draft SER – June 2, 2005
- ACRS Interim letter to the EDO – June 14, 2005
- Staff Response to ACRS letter – August 12, 2005

Project Milestones

- Final SER Issued – October 21, 2005
- ACRS Meeting on Final SER – December 8, 2005
- ACRS Letter to the EDO – December 22, 2005
- Final SER Issue as NUREG – January 28, 2006

Grand Gulf ESP Summary

- Final SER is issued on October 21, 2005
- The Grand Gulf ESP site characteristics with the limitations and conditions proposed by the staff comply with Part 100 requirements
- Reactor(s) having characteristics that fall within the parameters identified in the ESP, and which meet the terms and conditions proposed in the final SER, can be constructed and operated without undue risk to the health and safety of the public
- Issuance of the Grand Gulf ESP will not be inimical to the common defense and security or to the health and safety of the public
- Questions or comments?

Open Item No.	DSER Section	Open Item Subject	DSER Open Item Resolution
2.1-1	2.1.2.3	Demonstrate that the applicant has control over the exclusion area or has a right to obtain such control.	This is **Permit Condition 1**.
2.1-2	2.1.3.3	Include weighted transient population data in Tables 2.1-1 and 2.1-2 of the SSAR.	Additional information submitted; DSER Open Item closed.
2.3-1	2.3.1.3	Provide acceptable 100-year return period maximum and minimum dry-bulb temperatures.	Additional information submitted; DSER Open Item closed.
2.3-2	2.3.1.3	Provide the 48-hour probable maximum winter precipitation (PMWP) that can be used with the 100-year snowpack to define the extreme winter precipitation load site characteristics.	Additional information submitted; DSER Open Item closed.
2.3-3	2.3.1.3	Identify an additional ultimate heat sink (UHS) meteorological site characteristic for use in evaluating the potential for water to freeze in the UHS water storage facility.	Additional information submitted; DSER Open Item closed.
2.3-4	2.3.1.3	Identify a 3-second gust wind speed that represents a 100-year return period for the ESP site.	Additional information submitted; DSER Open Item closed.
2.3-5	2.3.5.3	Identify x/Q and D/Q values for the nearest milk cow and meat cow.	Additional information submitted; DSER Open Item closed.
2.4-1	2.4.1	Provide corrected UTM coordinates of the center of the proposed power block and/or revise Figure 2.1-1 in the SSAR to show the correct location and coordinates.	Additional information submitted; DSER Open Item closed.
2.4-2	2.4.1	Provide information on the elevation (depth) of the zone that could be disturbed by the construction of the new facility, such that the local subsurface environment and its alignment with the existing hydrogeological environment could be altered.	Additional information submitted; DSER Open Item closed.
2.4-3	2.4.1	Provide more details regarding dewatering wells to allow the staff to determine whether ground surface subsidence could affect safety-related	This is **COL Action Item 2.4-2.**

Open Item No.	DSER Section	Open Item Subject	DSER Open Item Resolution
		frequency dependence of Q in the eastern North American than in the western North American.	
2.5-3	2.5.2	Provide an explanation why the magnitude and distance bin corresponding to the SRSZ makes no contribution to the hazard deaggregation.	Additional information submitted; DSER Open Item closed.
2.5-4	2.5.2 & 2.5.4	Provide justification on applying the generic shear wave velocity profile derived from Memphis area to the ESP site and on its applying kappa value derived from ground motion observation on the Mississippi embayment in the sensitivity test.	Additional information submitted; DSER Open Item closed.
2.5-5	2.5.4	Provide the basis for the selection of values of BE, UB, and LB and other parameters for the base case profile.	Additional information submitted; DSER Open Item closed.
13.3-1		Provide responses to the following issues related to State and local emergency plans:	NA
a	13.3.3.7	Describe the communications arrangements with fixed and mobile medical support for the State of Mississippi and with mobile medical support for Claiborne County.	The staff will review during a COL or OL review.
b	13.3.3.8	Describe the dissemination of information regarding the special needs of the handicapped to the general public in the State of Louisiana on a periodic basis.	The staff will review during a COL or OL review.
c	13.3.3.11	Describe the means for the use of radioprotective drugs for emergency workers and institutionalized persons within the plume exposure pathway EPZ in the States of Louisiana and Mississippi whose immediate evacuation may be infeasible or very difficult.	The staff will review during a COL or OL review.
d	13.3.3.12	Describe the State of Mississippi's guidance related to bioassay or whole body counting for determining offsite emergency worker doses from the uptake of radioactive material (e.g., ingestion)	The staff will review during a COL or OL review.
e	13.3.3.13	Clarify the apparent inconsistencies between the	The staff will review during a COL or OL review.

Open Item No.	DSER Section	Open Item Subject	DSER Open Item Resolution
		instrumentation, data system equipment, power supplies, technical data and data systems, and record availability and management.	The staff concluded that the proposed major feature H is unacceptable.
13.3-4	13.3.3.11	Address whether discussions on results of the 2003 ETE study were held with officials from the States of Mississippi and Louisiana involved in implementing traffic management plans, according to Appendix 4 to NUREG-0654/FEMA-REP-1 and NUREG/CR-4831, or provide confirmation that State reviews were not required based on discussions with appropriate officials.	SERI responded that it had provided sufficient information regarding emergency plans in accordance with 10 CFR 52.17, and that this issue would be more appropriately addressed in the context of full and integrated emergency plans, which would be submitted with a COL application, rather than this ESP application. Subsequently, the GGNS licensee, in response to a RAI, stated the following in its letter dated June 28, 2005: All agencies in the States of Louisiana and Mississippi agreed that the 2003 ETE results support the conclusion in the 1986 ETE study, that the entire EPZ can be evacuated in any time of day or weather conditions in less than 3 hours and remains valid. The staff reviewed the applicant's response, as supplemented by a letter from the GGNS licensee dated June 28, 2005, and found that the results of the 2003 ETE study were subsequently reviewed and concurred on by the appropriate State officials. Therefore, Open Item 13.3-4 is resolved.

Final Draft Generic Letter on Hemyc and MT Fire Barriers

Sunil Weerakkody, Chief
Fire Protection Branch
Division of Risk Assessment
Office of Nuclear Reactor Regulation

ACRS Meeting
Rockville, MD
December 8, 2005

Purpose of Meeting

- To present the final draft Generic Letter 2006-XX: “Impact of Potentially Degraded Hemyc and MT Fire Barriers on Compliance with Fire Protection Programs”
- To obtain ACRS endorsement on the proposed generic letter

Generic Letter – Purpose

- To request that addressees identify whether Hemyc/MT is relied on for separation and/or safe shutdown
- To request that affected addressees provide a --
  - Description of the installation
  - Discussion of whether installation is in compliance, in light of new information
  - Description of compensatory measures
  - Corrective action schedule

Final Draft Generic Letter on Hemyc and MT Fire Barriers

Presenters,
Angie Lavretta, 301-415-3285
Daniel Frumkin, 301-415-2280

ACRS Meeting
Rockville, MD
December 8, 2005

Presentation Summary

- History
- Current Status
- Generic Letter Contents
- Public Comments & Comment Resolution
- Risk Assessment
- Conclusion

History

- Fire barrier issues raised in the 1980's
- Generic Letter 92-08 issued: called for re-assessment of all fire barrier types
- Action Plan implemented to resolve Thermo-Lag/Fire Protection issues, upgraded fire protection program inspections
- Recent NRC inspections of Hemyc raised NRC concern
- NRC initiated Hemyc and MT confirmatory tests

Recent Background

- NRC Hemyc and MT tests revealed previously-unidentified failure mode
- Information Notice 2005-07
- Public petitions filed (2)
- Plant-specific assessments needed/ draft generic letter published for comment
- Public meeting held
- Comments incorporated into final draft GL

  (Note: At least two of the 14 units affected have already begun fixes)

Generic Letter – Purpose

- To request that addressees identify whether Hemyc/MT is relied on for separation and/or safe shutdown
- To request that affected addressees provide a --
  - Description of the installation
  - Discussion of whether installation is in compliance, in light of new information
  - Description of compensatory measures
  - Corrective action schedule
- To require a written response in accordance with 10 CFR 50.54(f)

Generic Letter – Requested Action

- Within 60 days, provide the following:
  - A statement on whether Hemyc or MT fire barrier material is used at their NPP and whether it is relied on for separation and/or safe shutdown purposes in accordance with 10 CFR 50.48 or other regulatory commitments, including whether Hemyc or MT is credited in other analyses (e.g., exemptions, license amendments, GL 86-10 analyses)
  - A description of programmatic controls in place to ensure other fire barrier types will be assessed for potential degradation, in light of new information

Generic Letter – Requested Information

- Within 60 days, affected licensees are requested to address the following:
  - Whether the Hemyc and/or MT is degraded, in light of new findings of potential degradation. And plans for compensatory measures and corrective actions.
  - Justification for no corrective actions
  - Detailed description of Hemyc and/or MT installation
  - Detailed description of compensatory measures
  - Corrective actions implementation schedule, including intended licensing actions or exemptions

Generic Letter – Requested Information

- After implementing corrective actions, but no later than December 1, 2007, affected licensees are requested to provide the following:
  - Confirmation of compliance via corrective actions
  - A summary of the evaluation used for the "safety assessment"

Public Comments

- PCI Promatec
- Progress Energy
- Nuclear Energy Institute
- Duke Power
- STARS
- Exelon/AmerGen

Public Comments

Bin #	Description	# Rec'd
1	Comment on Backfit Determination	4
2	Comment on Schedule	4
3	Comment on Hemyc Testing	5
4	Comment on Risk-informing	3
5	Comment on GL 86-10, Supp. 1	5
6	Miscellaneous Comment	3
7	Comment on Details–e.g., wording, refs.	11
8	Comment on Burden Estimate	0

CRGR Comments

- CRGR Review/Questions
  - Backfit determination
    - Information request is not a backfit
    - Application of GL 86-10, Supplement 1
  - Change in impact determination
- Resolution
  - Removed incorrect reference to backfit
  - Clarified use of NFPA 251 and GL 86-10, Supp. 1
  - Explained minor change in impact determination

Simplified Risk Analysis

- Assumptions
  - Hemyc failure probability models generalized from test results, including sensitivity cases
  - Conservatism from the FPSDP
  - Typical, but not necessarily bounding, NPP fires and room layouts (combustibles, etc.)
- Determination
  - We <u>do not</u> expect any "high risk" situation

GL Issuance will Accomplish

- Compliance
- Plant-specific issues resolution
- Compensatory measures
- Corrective actions

Questions?

(12)

G:PPHandout

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
528	13	13.1

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

Authors:
JOHN T. LARKINS

List of Documents Attached

PLANNING & PROCEDURES MINUTES

13

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

From Staff Person
JOHN T. LARKINS

December 8, 2005 (5:20pm)
G:\PlanPro(ACRS)\ppmins.528.wpd

INTERNAL USE ONLY

SCHEDULE AND OUTLINE FOR DISCUSSION
ACRS PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
December 7, 2005

The ACRS Subcommittee on Planning and Procedures held a meeting on December 7, 2005, in Room T2B-3, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 10:00 a.m. and adjourned at 11:05 a.m.

ATTENDEES
G. Wallis
W. Shack
J. Sieber

ACRS STAFF
J. T. Larkins
A. Thadani
S. Duraiswamy
H. Nourbakhsh
M. Snodderly
M. Scott
J. Gallo
M. Afshar-Tous
M. El-Zeftawy
J. Lamb
G. Taylor
R. Caruso
J. Flack
C. Santos
E. Thornsbury
R. Savio
S. Meador

NRC STAFF
J. Mitchell

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

Member assignments and priorities for ACRS reports and letters for the December ACRS meeting are attached (pp. 5). Reports and letters that would benefit from additional consideration at a future ACRS meeting were discussed.

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the December ACRS meeting be as shown in the attachment (pp. 5).

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through March 2006 is attached (pp. 6-7). The objectives are to:

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

During this session, the Subcommittee also discussed and developed recommendations on items requiring Committee action (pp. 8-9).

RECOMMENDATION

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

3) Meeting with the NRC Commissioners

The ACRS met with the NRC Commissioners between 1:00 and 3:00 p.m. on Thursday, December 8, 2005 to discuss the following topics:

- Overview (GBW)
  - Major Accomplishments
  - License Renewal
  - Early Site Permits
  - Future ACRS Activities
- Issues Related to New Plant Licensing (including technology Neutral Framework) (TSK/MME)
- Proposed Alternative Embrittlement Criteria in 10 CFR 50.46 (DAP/RC)
- Fire Protection Matters (GEA/JGL)
- Power Uprate Technical Issues (RSD/RC)

Any follow-up items resulting from this meeting and a course of action for addressing them should be discussed by the Committee.

RECOMMENDATION

The Subcommittee recommends that the Committee discuss any follow-up items resulting from the meeting with the Commissioners and propose a course of action for addressing them.

4) ACRS Retreat in 2006

During its November 2005 meeting, the Committee approved a list of topics, and lead member assignments, for discussion during the retreat scheduled to be held on January 26-27, 2006. A proposed schedule for the retreat is attached (pp. 10-11). It is suggested that this meeting be held at the Marriott Hotel in Rockville.

RECOMMENDATION

The Subcommittee recommends that the Committee provide feedback on the proposed schedule and also decide on the proposal to hold this meeting at the Marriott Hotel in Rockville.

5) Candidates for Potential Membership on the ACRS (Closed)

The ACRS Member Candidate Screening Panel sent its recommendations to the Commission in November 2005 for appointment of two ACRS members to fill the vacancies on the Committee in the areas of Materials and Metallurgy and Plant Operations. Recently, the Commission has approved two candidates to fill these vacancies. Names of these candidates cannot be revealed until they are notified.

Additionally, there are two potential candidates for membership on the ACRS, with expertise in thermal-hydraulics and other areas. The Panel is seeking additional candidates.

RECOMMENDATION

The Subcommittee recommends that the ACRS Executive Director keep the Committee informed of the Commission decision on the appointment of two new members in the areas of Materials and Metallurgy and Plant Operations. The Commission is in the process of notifying these individuals.

6) Election of Officers for CY 2006

During its December 2005 meeting, the Committee will elect Chairman and Vice Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee. During the November ACRS meeting, it was requested that those members who do not wish to be considered for all or any of these Offices, should inform the ACRS Executive Director in writing by November 21, 2005. So far, two members have responded to this request.

7) Quadripartite Meeting Status

On December 5, 2005, a planning meeting among the Quadripartite members (ACRS, GPR, NSC, RSK) was held in Germany to discuss and finalize logistical and technical issues of interest, including the format of the abstracts which are due on February 28, 2006. The ACRS Executive Director and Mugeh Afshar-Tous attended this meeting. As a result, few changes have been made to the agenda to address the requests of all Quadripartite members. The revised agenda (pp. 12-15) reflects these changes. Also, the members agreed to extend an invitation to a representative from Finland to attend the Quadripartite Meeting.

RECOMMENDATION

The Subcommittee recommends that the ACRS Executive Director and Mugeh provide a summary of the agreements reached during the planning meeting. The members should provide feedback on the changes to the agenda.

8) Worksheets on Skill Set

During the November ACRS meeting, worksheets were provided to the members and staff requesting that they complete the worksheets as soon as possible. Most members and staff completed worksheets that help identify their technical expertise. The following modifications were made to the worksheets based on comments received: emergency planning has replaced evacuation planning; safeguards & security was added to the list of specific expertise; plant operating experience was subsumed into reactor operating experience. The worksheets will now be used to develop a strawman on ACRS technical expert needs and options to address gaps in technical expertise. The strawman will be distributed to members and staff prior to the January 2006 retreat.

RECOMMENDATION

The Subcommittee recommends that the strawman that identifies gaps in ACRS technical skill set and associated options to fill those gaps be distributed to the members and staff for feedback prior to the retreat. The Subcommittee also recommends that the strawman be presented and discussed at the January 2006 ACRS retreat.

9) Other Matters

Jocelyn Mitchell of RES informed the Subcommittee about a proposed RES plan for ranking the NRC research projects and how ACRS input on qualify review will be factored into the overall ranking.

The Subcommittee recommended that Jocelyn Mitchell brief the full Committee regarding this matter. Also, RES should provide a copy of the report to the ACRS prior to sending it to Congress.

ANTICIPATED WORKLOAD
DECEMBER 7 (1:00) - 10, 2005

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Bonaca	Apostolakis	Flack	Staff's Activities Associated with Responding to the Commission's SRM Related to Safety Conscious Work Environment and Safety Culture **[INFORMATION BRIEFING]**	—	—	—
Denning	—	Lamb	Draft Final Generic Letter, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance With Fire Protection Regulations"	A	To support staff schedule	**Draft**
	Wallis	Caruso	Vermont Yankee Extended Power Uprate Application and the Final SER	A	To support staff schedule	**Draft**
Powers	—	El-Zeftawy	Final Review of the Grand Gulf Early Site Permit Application and the Final SER	A	To support staff schedule	—
	All Members	Nourbakhsh/ Duraiswamy	NRC Safety Research Program Report	A	To respond to SRM. Due date March 15, 2006	**Draft**
Shack	Apostolakis	Snodderly/ Thornsbury	Proposed Program Plan and ANPR for Risk-Informing 10 CFR Part 50	A	To support staff schedule	—
Wallis	All Members	Larkins/Thadani/ Scott	Meeting with the NRC Commissioners [1:00 - 3:00, December 8, 2005]	—	—	—

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ANTICIPATED WORKLOAD
FEBRUARY 9-11, 2006

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	—	Thornsbury	Evaluation of HRA Methods with Respect to HRA Good Practices in NUREG-1792	A	To support staff schedule	—
Bonaca	—	Santos	Draft Final Generic Letter 2005-xx, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems"	A	To support staff schedule	—
	Apostolakis	Flack	Safety Conscious Work Environment/ Safety Culture **[NO BRIEFING]**	A	To support staff schedule	—
Powers	All Members	Nourbakhsh/ Duraiswamy	Final ACRS Report to the Commission on the NRC Safety Research Program	A	To respond to SRM. Due date March 15, 2006	—
	—	Snodderly/ Thornsbury	Draft Final Revision 1 to Reg. Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis"	Report as needed	—	—
Ransom	Wallis	Caruso	Proposed Revision 4 to Reg. Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a LOCA" **[TENTATIVE]**	A	To support staff schedule	—
	Wallis	Caruso	Application of TRAC-G Code for Analyzing ESBWR Stability	A	To support staff schedule	—
Sieber	Bonaca	Lamb/Santos	**SUBCOMMITTEE REPORT** - Interim Review of the Brunswick License Renewal Application [Subc. Mtg. 2/8/06]	—	—	—

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ANTICIPATED WORKLOAD
MARCH 9-11, 2006

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Bonaca	—	Santos	Final Review of the License Renewal Application and the Final SER for Browns Ferry Units 1, 2, and 3	A	To support staff schedule	—
		Thornsbury	Safeguard and Security Matters **[TENTATIVE] [CLOSED]**	A	To provide ACRS views	—
Denning	—	Caruso	FERRET Reactor Vessel Fluence Methodology **[INFORMATION BRIEFING]**	—	—	—
		Lamb	Proposed Revision to SRP Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs"	A	To support staff schedule	—
Powers	—	Taylor/Snodderly	Final Review of the Clinton Early Site Permit Application and the Final SER	A	—	—
Shack	—	Santos	Review of 1994 Addenda for Class 1, 2, and 3 Piping Systems to the ASME Code Section III and the Resolution of the Differences Between the Staff and ASME **[INFORMATION BRIEFING]**	—	To support staff schedule	—
Sieber	—	Thornsbury	Evaluation of Precursor Data to Identify Significant Operating Events **[INFORMATION BRIEFING]**	—	—	—
Wallis	—	Caruso	Chemical Effects Test Results/ Industry Responses to the Generic Letter on PWR Sumps	Report as needed	—	—

ACRS Items Requiring Committee Action

1 **Proposed Revision to R.G 1.76, "Design Basis Tornado and Tornado Missiles for Nuclear Power Plants", and SRP Sections 2.3.1 and 3.5.1.4** (Open)

Member: Thomas Kress **Engineer:** Med El-Zeftawy

Estimated Time:

Purpose: Determine a Course of Action

Priority:

Requested by: NRR B. Harvey

Regulatory Guide 1.76 is being revised to provide new guidance concerning the selection of the design basis tornado parameters such as maximum wind speed, pressure drop, and rate of pressure drop. This update is based on more realistic methodology and more recently available tornado data. In addition Standard Review Plan (SRP) Section 2.3.1., "Regional Climatology" is being revised to address a number of emergent issues related to climatic site characteristics resulting from the staff's review of early site permit applications. Also, SRP Section 3.5.1.4, "Missiles Generated by Tornado and Extree Winds" is being revised to eliminate presentation of tornado-generated missile spectrum guidance, which is being moved to R.G 1.76. All three documents are being revised to address 10 CFR Part 52 licensing actions such as standard design certifications, early site permits, and combined licenses. The NRR staff is requesting that the ACRS defer its review until after the public comment period.

Dr. Kress recommends that the Committee review this matter after reconciliation of public comments.

2 **SRP Section 17.5, "Quality Assurance Program Description- Design Certification, Early Site Permit and New License Applicants"** (Open)

Member: Thomas Kress **Engineer:** Med El-Zeftawy

Estimated Time:

Purpose: Determine a Course of Action

Priority:

Requested by: NRR P. Prescott

In accordance with implementation of a Staff Requirements Memorandum dated October 31, 2003, NRR has developed SRP Section 17.5. This new SRP section was prepared to address quality assurance programs for design certifications, early site permits, and new licenses (combined operating license and Part 50). There is no new guidance is included in this new SRP section, rather existing guidance has been consolidated. The staff is requesting that the ACRS defer its review until after the public comment period.

Dr. Kress recommends that the Committee review this matter after reconciliation of public comments.

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3 **Draft RG DG-1120 and SRP Section 15.0.2 concerning NRC Reviews of Transient and Accident Methods**

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

Estimated Time:

Purpose: Determine a Course of Action

Priority: High

Requested by: S. Marshall

Staff has received and resolved public comments related to DG-1120 and SRP 15.0.2, and it wants to issue these guidance documents for use by the staff and licensees. Committee authorized a Larkinsgram on October 28, 2005, informing staff that it wished to hear a staff presentation regarding the public comment resolution. Staff does not want to come back to the ACRS, and is extremely anxious to issue these documents by end of 2005. Dr. Wallis has reviewed the SRP and DG, and believes that the Committee does not need to review them. The Committee needs to determined whether to continue to move forward with review of these regulatory guidance documents, or change its mind and decide not to do so.

Dr. Wallis recommends that the Committee not review this matter.

9.

DRAFT

SD/bjw (Filed: P&P Agenda)

December 8, 2005

PROPOSED
SCHEDULE AND OUTLINE FOR DISCUSSION
ACRS SUBCOMMITTEE MEETING ON PLANNING AND PROCEDURES
JANUARY 26-27, 2006

<u>THURSDAY, JANUARY 26, 2006, BETHESDA NORTH MARRIOTT HOTEL AND CONFERENCE CENTER, 5701 MARINELLI ROAD, NORTH BETHESDA, MARYLAND</u>

1) 8:30 - 8:40 A.M. — <u>Opening Remarks by the ACRS Chairman</u> **(Open)** (GBW/JTL)
   - 1.1) Objectives of the meeting
   - 1.2) Anticipated goals, outcomes, and other matters

2) 8:40 - 10:40 A.M. — <u>Anticipated Workload/Technical Expertise</u> **(Open/Closed)** (DAP/ACT)
   - 2.1) Anticipated Workload for CY 2006 - 2008
   - 2.2) Anticipated Workload for each Subcommittee
   - 2.3) Technical Expertise Needed on the ACRS/ACRS staff/consultants in the future

10:40 - 11:00 A.M. — ***BREAK***

3) 11:00 - 12:00 Noon — <u>Strategy for Handling Anticipated Heavy Workload in 2006-2008</u> **(Open)** (GBW/JTL)
   - 3.1) Increasing the ACRS membership
   - 3.2) Establishing new subcommittees
   - 3.3) Expanding the meeting days
   - 3.4) Increasing the number of meetings from 10 to 12
   - 3.5) Increasing the number of staff engineers

12:00 - 1:00 P.M. — ***LUNCH***

4) 1:00 - 1:30 P.M. — <u>Strategy for Seeking Candidates for Future Membership on the ACRS</u> **(Open)** (GBW/JTL)
   - 4.1) Establishing an Ad Hoc Subcommittee to seek potential candidates through interaction with industry and other sources
   - 4.2) Maintaining a pool of candidates with expertise in different areas

5) 1:30 - 2:15 P.M. — <u>Proactive Initiatives</u> **(Open)** (GEA/ACT/JHF)
   - 5.1) Status of Proactive Initiatives Identified in the ACRS Action Plan
     - a. Advanced reactor design for hydrogen production
     - b. PRA model uncertainties
     - c. Impact of power uprates on the safety of nuclear plants in light of other ongoing regulatory activities
     - d. International activities (differences in regulatory approaches)
   - 5.2) New Proactive Initiatives

10.

2:15 - 2:30 P.M. — ***BREAK***

6) 2:30 - 3:00 P.M. Status of Implementing Commitments Made to Address Stakeholders' Comments Received During ACRS Self-Assessment Survey **(Open)** (WJS/RPS)

- 6.1) Early interaction by the ACRS with the EDO and the NRC staff on regulatory significance of complex technical issues
- 6.2) More ACRS members with industry and plant operating experience
- 6.3) Frequent interruption by ACRS members during NRC staff presentation and the need for enhanced understanding of regulatory issues and process

7) 3:00 - 4:30 P.M. Knowledge Management **(Closed)** (ACT/MA)

- 7.1`) Proposed options for ACRS/staff knowledge management

8) 4:30 - 5:00 P.M. Quadripartite Meeting **(Open)** (JTL/MA)

- 8.1) Status of arrangements for the Quadripartite meeting
- 8.2) Status of preparing abstracts and technical papers
- 8.3) Planned events

5:00 P.M. RECESS

FRIDAY, JANUARY 27, 2006, BETHESDA NORTH MARRIOTT HOTEL AND CONFERENCE CENTER, 5701 MARINELLI ROAD, NORTH BETHESDA, MARYLAND

9) 8:30 - 9:00 A.M. Review of Outcomes and Commitments from Day One **(Open)** (GBW/JTL)

10) 9:00 - 12:30 P.M. Significant Issues **(Open/Closed)**
(10:00-10:15 ***BREAK*)**

- 10.1) Advanced reactor designs (TSK) -significant technical challenges
- 10.2) Early site permit issues (DAP)
- 10.3) License renewal issues (MVB)
- 10.4) Extended power uprate issues (RSD)
- 10.5) Risk-informing 10 CFR Part 50 (WJS)
- 10.6) Safety Conscious Work Environment and Safety Culture (MVB/GEA)

11) 12:30 - 1:00 P.M. Summary of Commitments/Follow-up Items **(Open)** (GBW/JTL/ACT)

1:00 P.M. ADJOURN

11.

Technical Program Agenda
2006 Quadripartite Meeting
Day 1 – October 18, 2006

Plenary Session I: 8:00-11:30

8:00-8:15 ***Welcome*** – ACRS Chairman

8:15-9:00 Keynote Speaker – NRC Chairman

9:00 – 11:30 ***Technical Topics of Interest to Members in the Past Four Years - NSC Lead***
(25 min presentation + 5 min Q&A)

9:00 – 9:30 NSC
9:30 – 10:00 ACRS - Dr. Powers

10:00-10:30 Morning Break

10:30 – 11:00 GPR
11:00 – 11:30 RSK

11:30-1:00 Lunch

Discussion Session I: 1:00-3:00 p.m.

Safety Trends in Member Countries ***(each Country has 10 min for presentation + 5 min for Q&A)***

1:00-2:00 Use of PRA/PSA in Safety Assessment - ***NSC Lead*** (ACRS - Dr. Apostolakis)

2:00-3:00 Safety Management and Organizational Factors – ***RSK Lead*** (ACRS -Dr. Bonaca)

3:00-3:30 Afternoon Break

Discussion Session II: 3:30-5:30 p.m.

Licensing of Advanced Reactors ***(each Country has 10 min for presentation + 5 min for Q&A)***

3:30-4:30 Deterministic/Probabilistic Acceptance Metrics - ***ACRS Lead*** (**Dr. Kress** - to bring up issue of a standard multi-national design)

4:30-5:30 EPR Safety Assessment – ***GPR Lead*** (ACRS - Dr. Denning)

(12)

2006 Quadripartite Meeting
Day 2 - October 19, 2006

Plenary Session II: 8:00-11:30

8:00-8:45 Keynote Speaker - TBD

8:45 -11:30 ***Technology Advances and Changes to Regulatory Approach (RSK LEAD)*** *(25 min presentation + 10 min Q&A)*

8:45 -9:20 RSK
9:20 -9:55 ACRS – Dr. Denning

09:55-10:20 Morning Break

10:20 -10:55 GPR
10:55 -11:30 NSC

11:30-1:00 Lunch

Breakout Sessions will begin with a 5 minute presentation by the lead country followed by a 5 minute presentation by the other countries (if desired) including Switzerland and Sweden. However, the last 20-25 minutes should be dedicated to a discussion session among all participants.

Breakout Session I : 1:00-1:50

Sump Screen Blockage Issue – ***RSK***

(ACRS - Dr. Wallis)

Breakout Session II : 1:00-1:50

Fuel Operating Experience (includes high burnup & MOX Fuel) — ***GPR***

(ACRS-Dr. Powers)

Breakout Session III : 2:00-2:50

Evaluation of Materials Degradation Research – ***RSN***

(ACRS- Dr. Shack)

Breakout Session IV : 2:00-2:50

Seismic Design Guideline – NSC

(ACRS – Dr. Apostolakis)

2:50-3:10 Afternoon Break

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2006 Quadripartite Meeting
Day 2 - October 19, 2006 (...continued)

Breakout Session V:
3:10-4:00

Regulatory Treatment of Digital Instrumentation and Control (I & C) Systems Safety Criteria – ***RSK***

(ACRS - Mr. Sieber)

Breakout Session VI:
3:10-4:00

Methodologies for Accident Analyses – ***GPR***

(ACRS – Dr. Wallis)

Breakout Session VII:
4:10-5:00

Criteria for Emergency Core Cooling – ***ACRS – Dr. Shack***

Breakout Session VIII:
4:10-5:00

Phenomenology of Spent Fuel Pool Accidents – ***KSA***

(ACRS - Dr. Kress)

2006 Quadripartite Meeting
Day 3 – October 20, 2006

Plenary Session III: 8:00-10:00

Response to Significant Operating Events (GPR Lead) *(25 min presentation + 5 min Q&A)*

8:00-8:30 NPP Operating Experience in France - ***GPR***
8:30-9:00 NPP Operating Experience in US including Davis - Besse Nuclear Power Plant – ***ACRS – Mr. Sieber***
9:00-9:30 Events in Japan - ***NSC***
9:30-10:00 Events in Germany - ***RSK***

10:00-10:30 Morning Break

Discussion Session III: 10:30-11:30

Technical Issues for Operating Plants *(each Country has 10 min for presentation + 5 min for Q&A)*

10:30-11:30 Plant Aging, Life Extension & Periodic Safety Reviews – ***NSC Lead*** (ACRS – Dr. Bonaca)

Summary: 11:30-12:30

Conclusions and Recommendations - ACRS Chairman

1:00 p.m. - Conference Ends

(15)